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See Proposed Change to Tech Specs

SUBJECT: Responds to 970110 RAI re TS change request to convert Improved Standard TS pertaining to new requirements associated w/Low Temp Overpressure Protection Sys.

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10 CFR 50.90

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

RNP File No: 13510HA
Serial: RNP-RA/97-0021

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United States Nuclear Regulatory Commission
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE TECHNICAL SPECIFICATIONS CHANGE REQUEST TO CONVERT TO THE IMPROVED STANDARD TECHNICAL SPECIFICATIONS

Gentlemen:

By letter dated August 27, 1996, Carolina Power & Light (CP&L) Company submitted a request for a change to the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 Technical Specifications (TS) to convert the HBRSEP, Unit No. 2 TS to be consistent with NUREG-1431, "Standard Technical Specifications-Westinghouse Plants," Revision 1. By letter dated January 10, 1997, the NRC issued a request for additional information regarding the CP&L ITS submittal that pertained to new requirements associated with the Low Temperature Overpressure Protection System. In order to support the NRC review schedule for this submittal, the NRC has requested that the response to their request be submitted within 30 days of receipt of their letter (i.e., February 18, 1997).

The response to the NRC's request for additional information is provided as an Attachment to this letter.

Enclosure 1 provides an affidavit as required by 10 CFR 50.30(b).

Enclosure 2 provides marked up pages from the current TS, proposed ITS 3.4.12, and ITS 3.4.12 Bases pertaining to LTOP requirements.

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Enclosure 3 provides a marked up and a retyped page 3.10-9 from the current TS pertaining to removal of an allowance for a single inoperable control rod.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of this letter with the enclosures and attachment.

If you have any questions concerning this matter, please contact me at (803) 857-1437.

Very truly yours,

H. K. Chernoff
Supervisor - Licensing/Regulatory Programs

ALG/alg
Enclosures:

1. Affidavit
2. Marked up pages from the current Technical Specifications, Proposed ITS 3.4.12, and ITS 3.4.12 Bases pertaining to the Low Temperature Overpressure Protection (LTOP) requirements
3. Marked up and retyped page 3.10-9 from current Technical Specifications pertaining to removal of allowance for a single inoperable control rod

Attachment

c: Mr. Max K. Batavia, Chief, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, Regional Administrator, USNRC, Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP (4 copies)
Mr. B. B. Desai, USNRC Resident Inspector, HBRSEP
Attorney General (SC) (w/out Enclosures)
Lockheed Idaho Technology, Inc.

Affidavit

State of South Carolina
County of Darlington

C. S. Hinnant, having been first duly sworn, did depose and say that the information contained in letter RNP-RA/97-0021 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

CSHinnant

Sworn to and subscribed before me

this 18th day of February 1997

(Seal) Albert L Garron
Notary Public for South Carolina

My commission expires: March 22nd 2005

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE TECHNICAL SPECIFICATIONS CHANGE REQUEST TO CONVERT TO THE IMPROVED STANDARD TECHNICAL SPECIFICATIONS

Questions 1 through 8

Question 1.

Provide the power-operated relief valve (PORV) lift setpoint calculations for the entire range of LTOP. Include supporting discussions and material for: dynamic head effect, static head effect, overshoot, and instrument uncertainty. Indicate whether the values in Table 2.1, "LTOP System Analysis Results," include the above four effects.

Response.

The Power-Operated Relief Valve (PORV) lift setpoint calculation for the Low Temperature Overpressure Protection (LTOP) System was provided to the NRC by letter dated December 15, 1977. A single LTOP setpoint of 400 psig is used and applies to Reactor Coolant System (RCS) temperatures of 350°F and below. The analyses are presented in Table 2.1, "LTOP System Analysis Results," of Enclosure 5 to our letter dated August 27, 1996. The analyses accounted for the static head from the LTOP transmitter location at the top of the pressurizer to the reactor vessel beltline location and the dynamic head from the Reactor Coolant Pumps (RCPs) or Residual Heat Removal (RHR) pumps. An instrument uncertainty of 30 psi was assumed in the analyses, and the instrument uncertainty was evaluated in 1993 to be 24.2 psig. The evaluated instrument uncertainty was developed utilizing the square root of the sum of the squares method for an eighteen month calibration frequency utilizing the instrument accuracy for the Rosemount Model 1154 GP9 pressure transmitter, Hagan Model 118 Comparator, and Foxboro Model 66N Function Generator. The resulting uncertainty was ± 0.808% of span (0 to 3000 psig), which resulted in a total instrument loop error of ± 24.2 psig.

The limiting mass input case from Table 2.1 of Enclosure 5 to our letter dated August 27, 1996, is for an RCS pressure of 275 psig at a RCS temperature of 175°F. The calculated overshoot reaches 562.4 psig utilizing the 400 psig setpoint and taking into account static and dynamic effects, and conservatively assuming an instrument uncertainty of 30 psig. The zero degree cooldown rate pressure limit from current Technical Specifications Figure 3.1-2 (i.e., Improved Technical Specifications (ITS) Figure 3.4.3-2) is 540 psig at an RCS temperature of 175°F.

Adding 60 psi that is built into the curve for instrument uncertainties results in a pressure limit of 600 psig at an RCS temperature of 175°F. The net margin in the analysis from a 400 psig setpoint is then 37.6 psig, without instrument uncertainty. Assuming an instrument uncertainty of 30 psig, the net margin in the analysis is 7.6 psig.

Question 2.

Discuss how the restriction on the number of reactor coolant pumps (RCPs) operating (i.e., only two RCPs may be operating when the RCS temperature is between 350°F and 175°F) is controlled. Describe the acceptable configurations of operating RCPs below 175°F and provide the supporting analyses. Also, with regard to the dynamic head, describe the effect of operating residual heat removal pumps and include, as appropriate, an evaluation or analysis of this effect on the overall setpoint determination. Submit the proposed and current TS sections affected by this request.

Response.

The number of operating RCPs will be restricted to two in the plant condition defined by RCS temperature ≥ 175°F and < 350°F by the use of restrictions in operating procedures and by removing the control power fuses to the power supply breaker to at least one RCP, or by maintaining the power supply to at least one RCP in the racked out position. At RCS temperatures below 175°F no restriction is placed on operation of RCPs since, at these temperatures, SI pumps are required to be rendered incapable of injecting into the RCS when the reactor vessel is not vented to the minimum requirements.

The supporting analyses for operation at an RCS temperature below 175°F are the LTOP analyses in existence prior to our letter dated August 27, 1996.

By letter dated August 11, 1976, the NRC requested an evaluation of the HBRSEP, Unit No. 2 system designs to determine susceptibility to overpressurization events, an analysis of the possible events and proposed interim and permanent modifications of systems and procedures to reduce the likelihood and consequences of such events. By letters dated July 28, 1977, October 31, 1977, December 15, 1977, and January 25, 1978, CP&L provided analyses of overpressurization events including the expected response of the LTOP system in response to overpressurization events. By letter dated December 22, 1977, CP&L requested a change to Technical Specifications to provide operability and surveillance requirements for the LTOP System. By letter dated September 14, 1979, Amendment No. 42 was issued to the Facility Operating License for HBRSEP, Unit No. 2, that changed Technical Specifications (TS) to incorporate the LTOP System requirements. The accompanying Safety Evaluation (SE) determined that the limiting events identified in the SE formed an acceptable basis for analyses of the performance of the HBRSEP, Unit No. 2 LTOP System.

The proposed HBRSEP, Unit No. 2 ITS Section 3.4.12 maintains the current licensing basis for operation with RCS temperature below 175°F. Below this temperature, in accordance with the previously approved LTOP analyses, three RCPs and three charging pumps may be capable of operating, and no SI pumps may be capable of injecting into the RCS when the reactor vessel is not vented to the minimum requirements. The SI pumps are rendered incapable of injecting into the RCS by maintaining the breakers in the racked out condition in accordance with operating procedures.

The RHR flow rate was included in the dynamic head at the reactor vessel beltline. The RHR system takes suction from the RCS Loop B Hot Leg. The normal flow path is through the RHR pumps and heat exchangers, then through the Emergency Core Cooling System (ECCS) accumulator lines to each of the cold legs. The minimum RHR flow to be considered was 2,800 gpm. The density of water at 400 psig and 175°F is 60.76 lbm/ft^3, and the equivalent mass flow is 379 lbm/sec.

The inlet to the RHR system was modeled in ANF-RELAP by attaching a time dependent junction to the RCS Loop B (i.e., Loop 2) Hot Leg which was connected to a time dependent boundary condition volume. The flow rate through the junction was a constant 379 lbm/sec. The return of RHR flow to the cold legs was modeled with another time dependent junction connecting a time dependent boundary condition volume to the ECCS connecting header to the ECCS accumulator injection lines. This junction distributed the RHR flow to each cold leg. The conditions of the time dependent volume were adjusted to correspond to the desired conditions of pressure and temperature for each case.

By modeling the RHR system in this way, the dynamic head was sensed at the limiting location in the reactor vessel and the operation of the RHR pumps was accounted for in the verification of the setpoint.

In order to allow ease of review, marked up pages of the Current Technical Specifications (CTS) pertaining only to the LTOP requirements and proposed ITS Section 3.4.12 and Bases are provided in Enclosure 2. The information provided in the CTS markup is taken from Enclosure 12 to our letter dated August 27, 1996.

Question 3.

You requested to allow for an operable safety injection (SI) pump in Mode 4. Please discuss acceptable configurations for other Modes. Indicate when the SI pump is disabled and how it is made inoperable. Submit your proposed and current TS sections affected by this request.

Response

The operating configuration for the SI pumps, RCPs and charging pumps are outlined in Table 1 below.

	RCPs	SI Pumps	Charging Pumps
MODE 1	3	2	3
MODE 2	3	2	3
MODE 3	3	2	3
MODE 4	2	1	1
MODE 5, RCS ≥ 175°F	2	1	1
MODE 5, RCS < 175°F	3	0	3
MODE 6, Vessel Vented[1]	0	0	3
MODE 6, Vessel Not Vented[1]	0	2	3

Table 1, "Acceptable Configuration of RCPs, SI Pumps and Charging Pumps

The SI pump is disabled by maintaining the SI pump breaker in the racked out condition.

1 The minimum required vent cross section area is three (3) square inches.

The information provided in Table 1 is based upon the restrictions contained in the proposed ITS, and therefore, no changes to our submittal are required.

In order to allow ease of review, marked up pages of the Current Technical Specifications (CTS) pertaining only to the LTOP requirements and proposed ITS Section 3.4.12 and Bases are provided in Enclosure 2. The information provided in the CTS markup is taken from Enclosure 12 to our letter dated August 27, 1996.

Question 4.

In the request dated August 27, 1996, you stated:

> "The existing analysis for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, was based on generic vessel cooldown limits and the more restrictive curves, which resulted in the current technical specifications that require the SI pump be disabled when LTOP is placed into service."

You also stated,

> "The pressurization limits correspond to the 0°F cooldown limitation curve in Fig. 3.1-2 of Ref. 1."

Explain the first statement and any changes to it. Regarding the second statement, describe to what the limits correspond currently. Indicate what is required by the current methodology approved for HBR. Submit the current LTOP methodology (in its entirety) , and provide an itemized discussion of any changes proposed.

Response

The additional embrittlement margin gained in the development of P-T curves from the reactor vessel material surveillance program has enabled the necessary margin in the P-T limits to permit a single train of Safety Injection to be OPERABLE in MODE 4 as an analyzed condition.

The Pressure-Temperature (P-T) curves, which resulted in the current TS requirement to disable the SI pumps, were incorporated into TS by Amendment 26, issued by NRC letter dated February 11, 1977. The P-T curves in Amendment 26 were developed utilizing the upper circumferential weld material as most limiting at End-of-Life (EOL) based on Regulatory Guide 1.99, Revision 1, without taking credit for surveillance data. Consequently, the P-T curves in Amendment 26 were more restrictive than the current P-T curves. By letter dated December 22, 1988, CP&L responded to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials & Its Impact on Plant Operations," and stated that with the embrittlement margin gained for the upper circumferential beltline weld through surveillance results, the lower circumferential beltline weld was most limiting for EOL. By letter dated March 9, 1990, the NRC concurred with the use of reactor vessel capsule surveillance data for the development of HBRSEP, Unit No. 2 P-T curves. By letter dated September 15, 1993, CP&L requested changes to TS incorporating new P-T curves utilizing the additional embrittlement margin covering plant operations through 24 Effective Full Power Years (EFPYs).

By letter dated July 29, 1994, the NRC issued Amendment No. 149 to TS incorporating the new P-T curves. The NRC found the new curves acceptable to 24 EFPYs by performing an independent calculation. In its response to Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," dated November 20, 1995, CP&L provided revised chemistry and initial Reference Temperature - Nil Ductility Transition (RT_{NDT}) values for Reactor Pressure Vessel materials in some cases, and stated that none of the changes would result in projections exceeding the screening criteria specified in 10 CFR 50.61 for Pressurized Thermal Shock (PTS) considerations through EOL, nor would the changes necessitate a change to the LTOP or P-T limits.

With regard to the second question, as stated in response to Question 1, the zero degree cooldown rate pressure limit from current Technical Specifications Figure 3.1-2 (i.e., ITS Figure 3.4.3-2) is 540 psig. Adding 60 psi that is built into the curve for instrument uncertainties results in a pressure limit of 600 psig (i.e., considered the analytical limit) at an RCS temperature of 175°F. The net margin in the analysis from a 400 psig setpoint is then 37.6 psig, without instrument uncertainty. Assuming an instrument uncertainty of 30 psig, the net margin in the analysis is 7.6 psig. An RCS temperature of 175°F is the lower and most conservative limit of the analyses in Table 2.1 of Enclosure 5 to our letter dated August 27, 1996.

The LTOP methodology is contained in the report, "Pressure Mitigating Systems Transient Analysis Results," Westinghouse Electric Corporation, submitted to the NRC by letter dated July 28, 1977, and in the report supplement, "Supplement to the July, 1977 Report on Pressure Mitigating Systems Transient Analysis Results," Westinghouse Electric Corporation, submitted to the NRC by letter dated October 31, 1977.

No changes to the existing LTOP methodology are proposed for RCS temperatures < 175°F or for the heat input analyses which cover the entire range of LTOP. The only change to the LTOP methodology is the new analyses provided in Enclosure 5 to our letter dated August 27, 1996, which will allow a single train of SI capability in the RCS temperature range from ≥ 175°F to < 350°F.

Question 5.

Justify why H. B. Robinson is using peak reactor vessel lower head pressure for the calculation of the maximum pressures. Explain how this is used (i.e., whether it is used as the peak pressure or if back calculations of pressures at the limiting locations are performed).

Response

The maximum pressure observed during the transient was obtained from the reactor vessel lower head volume. The pressure increase due to the static head between the vessel beltline and the lower head more than offsets the dynamic losses in the vessel downcomer between the beltline location and the lower head. Use of the lower head pressure for the beltline pressure limitation is therefore conservative. Use of the calculated pressure in the reactor vessel head conservatively accounted for the static head from the LTOP pressure transmitter sensing location at the top of the pressurizer to the beltline location and the dynamic head from the RCPs.

Question 6.

You stated that opening and closing setpoints of 430 psig and 415 psig, respectively, were used in the analysis. Further, you stated that these values incorporate a 30 psi error allowance on the 400 psig opening setpoint and a hysteresis in closing pressure of 30 psi ±15 psi. However, the TS-required setpoint is 420 psi. Indicate whether the value is in psig or psia. Explain the discrepancy between the 400 psig analysis value and the 420 psi TS value.

Response

The CTS Section 3.1.2.1 requires an LTOP lift setting of less than or equal to 420 psi. The units for the lift setting are psig. The new analysis margin for overpressure events, which allows for operation in MODE 4 with a single OPERABLE SI pump, requires a reduction in the currently required LTOP setpoint of 420 psig as specified in the CTS to the new setpoint of 400 psig as specified in proposed ITS Section 3.4.12. An inconsistency exists in the application of LTOP setpoint uncertainty in the proposed ITS that will be discussed below.

The LTOP System instrumentation consists of the items identified below:

- Field transmitters and process sensors which provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

- Signal processing which provides signal conditioning, bistable setpoint comparison, process actuation, compatible electrical signal output to protection system devices, and control room indications;

- Relay logic which initiates LTOP actuation in accordance with the defined logic, which is based on the bistable outputs from signal processing; and

- Pressurizer PORVs which receive a signal to open when the system is armed and the pressure setpoint is exceeded.

To meet the design demands for redundancy and reliability, two field transmitters and sensors are used to measure the pressure parameter. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the trip setpoint. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

The trip setpoint is the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., rack calibration + comparator setting accuracy).

The trip setpoints used in the bistables are based on the calculated peak pressure of 562.4 psig stated in Table 2.1 of Enclosure 5 to our letter dated August 27, 1997. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the CP&L setpoint methodology procedure which is based upon current Instrument Society of America (ISA) standards[2]. Using the proposed required LTOP setpoint of 400 psig as an “as left” LTOP setpoint, the 30 psi instrument error referred to in Section 3.2 of Enclosure 5 to our letter of August 27, 1996, allows for instrumentation uncertainty and a small additional margin. The resulting value of 430 psig is then input into the LTOP analysis to determine that pressure overshoot will remain below the required pressure limit.

While evaluating Question 6, CP&L has determined that an inconsistency exists between the specification of the LTOP setpoint in proposed ITS Section 3.4.12 and the specifications of instrument setpoints in proposed ITS Section 3.3, “Instrumentation.” ITS Section 3.3 specifies both the instrument setpoint and an allowable value for the setpoint that represents the maximum allowable “as found” value for the instrument for the instrument to be considered OPERABLE during calibration. The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the allowable value to account for changes in random measurement errors detectable by a Channel Operational Test (COT). One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the allowable value, the channel is considered OPERABLE. Because the 1993 evaluation of instrument uncertainty was based upon a total channel calibration and the proposed ITS surveillance requirement for performance of a COT does not include the sensor, a new allowable value should be calculated.

Therefore, in order to maintain the specification for LTOP setpoint consistent with the methodology in ITS Section 3.3, the proposed Limiting Condition for Operation (LCO) 3.4.12 will be revised in a supplement to our submittal to require two power operated relief valves (PORVs) with the lift settings of ≤ 400 psig and an allowable value of less than [later] psig. The allowable value will be included in the supplement based upon new calculations supporting performance of a COT.

2 ISA Standard S67.04, Part I, 1994, “Setpoints for Nuclear Safety Related Systems,” and
ISA Recommended Practice RP67.04, Part II, 1994, “Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation,”

Question 7.

For Section 3.1, "Modeling Changes," provide a discussion of the changes, their bases, and whether these were assumed in the previous mass addition LTOP analysis.

Response

The current mass addition LTOP analysis, which will remain the analysis of record for plant conditions when RCS temperature is below 175°F, is described in our letter dated July 28, 1977. The mass input analyses was performed utilizing the one loop version of the LOFTRAN[3]. The input modeling, input additions and initialization changes were described in the report, "Pressure Mitigating Systems Transient Analysis Results," prepared by Westinghouse Electric Corporation, and attached to our letter dated July 28, 1977 (i.e., Westinghouse Analyses). The Westinghouse Analyses assumed an inadvertent start of an SI pump as the limiting event in the mass input case. The Westinghouse Analyses also assumed a two (2) second opening time for the PORV. An evaluation was performed by CP&L in 1986 that took credit for the current limitation of racking out the power supply breakers to the SI pumps to determine that a 2.5 second opening time for the PORV would be acceptable. The evaluation assumed flow from three charging pumps and no flow from the SI pumps as the limiting case, and the evaluation showed that the results were bounded by the Westinghouse Analyses.

The proposed ITS Section 3.4.12, Mode 4 with RCS temperatures ≥175°F, is based upon the analyses presented in Enclosure 5 to our letter of August 27, 1996 (i.e, the SPC Analyses). Section 3.1, "Modeling Changes," describes the modeling changes and input assumptions associated with utilizing the ANF-RELAP model for performing the LTOP analysis. The ANF-RELAP model was approved for use by the NRC for use in Chapter 15 overpressurization analyses by NRC letter dated March 16, 1992.

A comparison of the input assumptions between the Westinghouse and SPC Analyses has been performed in Table 2 below.

3 WCAP-7907(P)/(NP)

	Westinghouse Analyses	SPC Analyses
Applicable RCS Conditions	≤ 350°F	175°F ≤ RCS Temperature ≤ 350°F
SI Pumps	1	1
Charging Pumps	40 gpm	77 gpm
RCPs in operation	Not considered because analysis was performed at RCS temperature of 100°F	2

Table 2, "Comparison of Westinghouse and SPC Analyses."

Westinghouse Electric Corporation advised CP&L by letter dated March 15, 1993, that certain non-conservatisms may exist in the Westinghouse methodology utilized in the performance of the Westinghouse Analyses. Specifically, the Westinghouse Analyses did not consider the pressure differences between the LTOP transmitter and the reactor vessel at the core midplane elevation, and that the analyses did not account for the dynamic head due to coolant flow through the reactor vessel to the pressure sensing location. The CP&L evaluation of this issue that was conducted in 1993 resulted in the corrective action to restrict operation of the number of RCPs to two (2) with RCS temperature less than 260°F in accordance with the previous P-T curves. With the issuance of the current P-T curves in Amendment 149 to TS, adequate margin existed to remove the operating restriction on the RCPs at RCS temperatures lower than 260°F. Therefore, the proposed ITS will not restrict operation of the RCPs at RCS temperatures < 175°F in conformance with the Westinghouse Analyses, and will restrict the number of operating RCPs in Mode 4 to two (2) at RCS temperatures ≥ 175°F in conformance with the SPC Analyses.

Question 8.

You are shifting the TS curves up by 30 psi because instrument uncertainty is 30 psi less than is accounted for in these curves. Indicate what values (curves) were used in the previous LTOP analysis, and whether you have provided the instrument uncertainty calculations for NRC staff review.

Response

The Westinghouse Analyses were performed when the more restrictive previous P-T curves, as described in the response to Question 4, were in effect (i.e., 1977). The previous P-T curves included a 60 psig and 10°F margin for instrument error. Instrument uncertainty calculations were not submitted to the NRC during the NRC evaluations culminating in issuance of TS Amendment 42 as described in the response to Question 2. The 1993 evaluation of instrument uncertainties were not provided to the NRC for review. The results of the 1993 evaluation and key parameters and assumptions are provided in response to Question 1.

Question 9.

"Provide the proposed TS for the current TS Section 3.10.6.2 resulting from the discussion in the last paragraph on page 3 of 5 of the August 27, 1996, submittal that was related to plant operation with one inoperable control rod."

Response

By letter dated December 4, 1995, Carolina Power & Light (CP&L) Company reported, in accordance with 10 CFR 50.9, that a provision in the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 Technical Specifications (TS) may create a condition beyond the current analysis. The issue involves Current TS (CTS) Section 3.10.6.2, which permits continued reactor operation with one inoperable control rod, and no associated action statement limits operation in this condition. During a review of the licensing basis for CTS Section 3.10.6.2, CP&L identified that the Updated Final Safety Analysis Report (UFSAR) Chapter 15 Accident Analysis assumes the most reactive control rod fully withdrawn. However, if during operation as allowed by CTS Section 3.10.6.2, a failure of another control rod occurred, the plant could be operating in a condition that is beyond the current accident analysis. In the letter dated December 4, 1997, CP&L stated that a TS change request to correct this provision would be provided.

By letter dated August 27, 1996, CP&L submitted a request for a change to the HBRSEP, Unit No. 2 TS to convert the HBRSEP, Unit No. 2 TS to be consistent with NUREG-1431, "Standard Technical Specifications-Westinghouse Plants," Revision 1. The submittal included deletion of CTS Section 3.10.6.2 and justified the change as a more restrictive change to the CTS.

The proposed TS for the CTS concerning deletion of CTS Section 3.10.6.2 are described in the following paragraphs. "Discussion of the Change" provides a detailed description of why the change to CTS is justified. "No Significant Hazards Consideration (NSHC) and Basis for Categorical Exclusion from 10 CFR 51.22," provide information required by 10 CFR 50.91(a) that supports a finding of NSHC. "Environmental Considerations" provides information in support of the conclusion that the proposed change satisfies the categorical exclusion specified in 10 CFR 51.22(c) to perform an Environmental Assessment or Environmental Impact Statement. Enclosure 3 to this letter provides a marked up CTS page 3.10-9 showing only the change to CTS Section 3.10.6.2, followed by a retyped proposed CTS page delineating the change.

Discussion of the Change

CTS Section 3.10.6.2 states that "no more than one inoperable control rod shall be permitted during power operation." If more than one control rod becomes inoperable, the CTS does not provide a specific Action statement, and therefore CTS Section 3.0 would be entered, which allows eight (8) hours to achieve Hot Shutdown conditions and an additional 30 hours to achieve Cold Shutdown conditions. The CTS does not stipulate an Allowed Outage Time (AOT) for a single inoperable control rod thereby permitting continuous operation under CTS Section 3.10.6.2. Current safety analyses contained in UFSAR Chapter 15 assume that in the event of any actuation of the Reactor Protection System (RPS), the single most reactive rod will remain stuck in the fully withdrawn position after all other rods have inserted into the reactor core. If the reactor operated continuously with an inoperable control rod in accordance with CTS Section 3.10.6.2, a single failure of another control rod to fully insert in the event of actuation of RPS would not fall within the analyzed assumptions of UFSAR Chapter 15.

Continuous operation with a known inoperable control rod should not be allowed to continue with the inoperable control rod counted as the single failure for which the plant was designed. Otherwise, if continuous operation with a known inoperable control rod were permitted, another single failure consisting of an inoperable control rod that is not yet detected by surveillance testing should be assumed in the safety analyses. However, two inoperable control rods are not assumed in the HBRSEP, Unit No. 2 Chapter 15 safety analyses.

The General Design Criteria (GDC) in existence at the time that HBRSEP, Unit No. 2 was licensed (i.e., July 1, 1970) for operation were contained in proposed Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants, published in the Federal Register on July 11, 1967. Although the proposed GDC relating to redundancy of reactivity control and reactivity shutdown capability are similar to the current Appendix A to 10 CFR 50, the degree of understanding regarding application of the single failure criterion has matured since initial licensing. Consequently, CP&L no longer considers CTS Section 3.10.6.2 an appropriate action for an inoperable control rod.

Deletion of CTS Section 3.10.6.2 ensures operation within the bounds of the applicable UFSAR Chapter 15 safety analysis and consistently applies the single failure criterion to the control rods. CTS Section 3.10.6.2 was not retained in the proposed HBRSEP, Unit No. 2, Improved TS submittal, because retention of CTS Section 3.10.6.2 would have been inconsistent with NUREG-1431, Revision 1.

No Significant Hazards Consideration (NSHC) and Basis for Categorical Exclusion from 10 CFR 51.22

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified to delete Technical Specifications (TS) Section 3.10.6.2 which imposes a more restrictive requirement than currently exists. The more restrictive change is being imposed to be consistent with

NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated the proposed Technical Specification change to delete TS Section 3.10.6.2 and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides a requirement determined to be more restrictive than the current Technical Specifications requirement for operation of the facility. The more restrictive requirement is not assumed to be an initiator of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirement being proposed enhances assurance that the operability of control rods are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, this change does not involve any increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The change imposes a new requirement which is consistent with assumptions made in the safety analysis and licensing basis. The additional requirement is a more restrictive Required Action for an inoperable control rod that enhances safe operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of the more restrictive requirement increases the margin of plant safety. The change is designed to enhance plant safety and is consistent with the safety analyses and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

Environmental Considerations

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulator actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. We have reviewed this request and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance with the amendment. The basis for this determination follows.

A. Proposed Change

This request proposes to delete Technical Specifications (TS) Section 3.10.6.2 to be consistent with current safety analyses.

B. Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Evaluation, the proposed change does not involve a significant hazards consideration.

2. These proposed change is being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, and to establish consistency with the safety analyses, and does not involve physical changes to the facility, nor does the change affect actual plant effluents.

3. These proposed change is being made to establish consistency with the improved Standard Technical Specifications (ISTS) - Westinghouse Plants, NUREG 1431, Rev. 1, and to establish consistency with the safety analyses and does not involve physical changes to the facility, and does not significantly affect individual or cumulative occupational radiation exposures.

Markup of CTS and Proposed Change to CTS Pages (Retyped)

A proposed markup of CTS page 3.10-9 showing only this change is provided in Enclosure 3. A retyped CTS page 3.10-9 follows the marked up page 3.10-9.