UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 24, 1997

50-261

Mr. J. S. Keenan, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
  Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: CHANGES TO TECHNICAL SPECIFICATION BASES SECTIONS 3.10, 3.1.1.5, AND 4.5 FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (TAC NOS. M97760 AND M99077)

Dear Mr. Keenan:

By letter dated October 1, 1996, Carolina Power & Light Company (CP&L) proposed a change to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR), Technical Specifications (TS) Bases section 3.10. The change replaced a specific misalignment assumption for the rod cluster control assembly (RCCA) analysis with a statement that the static misalignment of a single RCCA has been analyzed in Section 15 of the Updated Final Safety Analysis Report, and that it has been demonstrated that the maximum credible misalignment of a single control rod does not result in exceeding core safety limits. The change to the bases wording was made to limit the need to change the bases page whenever a cycle-specific change is made with respect to control rod insertion limits.

By letter dated October 7, 1996, CP&L proposed changes to HBR TS Bases sections 3.1.1.5 and 4.5. The change to Bases section 3.1.1.5 was made to correct an error associated with licensee submittals made on June 18, 1992, December 8, 1992, and February 3, 1995, regarding a TS change associated with Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." Bases section 3.1.1.5 had incorrectly credited power-operated relief valves for mitigating a steam generator tube rupture coincident with a loss of offsite power.

Bases section 4.5, "Emergency Core Cooling, Containment Cooling and Iodine Removal System Tests" indicated that an ohmmeter was used for continuity checks to verify a complete circuit from the logic matrices to the master relay. Because an ohmmeter is not the only way to check for continuity, a change in the bases was made to simply indicate that verification is accomplished by a continuity check.

DF01 1/1

Mr. J Keenan

The NRC staff has reviewed the proposed bases changes and has no objection to the changes as proposed.

Enclosed are copies of the revised Bases pages.

Sincerely,

(Original Signed By)

Brenda Mozafari, Project Manager
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Division of Reactor Projects - I/II
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Docket No. 50-261

Enclosures: Bases pages 3.1-3g, 3.1-3h, 3.10-11, and 4.5-3

cc w/enclosure: See next page

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Basis

At the conditions of the RCS temperature (T_{avg}) greater than 350°F or the reactor critical, the power-operated relief valves (PORVs) provide an RCS pressure boundary and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary is the safety-related function of the PORVs at the conditions noted above. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The automatic RCS pressure control function of the PORVs is not a safety-related function at the conditions noted above. The automatic pressure control function limits the number of challenges to the safety valves, while the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Operation with the block valves open is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed to isolate PORV leakage is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The ability to operate with the block valve(s) closed with power maintained to the block valve(s) is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage so that maintenance can be performed on the PORVs to eliminate the leakage condition. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor coolant system pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has leakage resulting in excessive RCS leakage. (Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.) The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to exceeding cold shutdown following the associated refueling outage.

The OPERABILITY of the PORVs and block valves at the conditions noted above is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary.

2. Manual closing of a block valve to isolate a stuck open PORV and.

3. Manual closing of a block valve to isolate a PORV with excessive seat leakage.

A PORV is defined as leaking with up to and including one (1) gpm of seat leakage, but is not inoperable and is not experiencing "excessive" seat leakage as identified within Specification 3.1.1.5.a. With leakage up to and including ten (10) gpm, the PORV would be considered to have "excessive" seat leakage and would be subject to the compensatory actions described within Specification 3.1.1.5.a. This condition would continue to require block valve testing on a 92 day interval as required by Surveillance Requirement 4.2.4.2. Finally, with PORV leakage exceeding ten (10) gpm, the PORV is considered inoperable in accordance with Specifications 3.1.1.5.b. and c., and block valve testing is not required.

shutdown margin. The specified control rod insertion limits meet the design basis criteria on (1) potential ejected control rod worth and peaking factor,[4] (2) radial power peaking factors, $F_{\Delta H}$, and (3) required shutdown margin.

When the control Rod Banks are outside the acceptable insertion limits, they must be restored to within those limits. The restoration may be accomplished either by repositioning control rods to the insertion limits consistent with core power or by boration to reduce power to be consistent with the power limit associated with the existing rod position. Restoration of rod position to the power dependent insertion limit ensures adequate Shutdown Margin in accordance with Figure 3.10-2.

Operation beyond the insertion limits is allowed (for a short time period in order to take conservative action) because the simultaneous occurrence of either a LOCA, loss of reactor coolant flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. The allowed completion time of one hour for restoring the banks to within the insertion limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

The various control rod banks (shutdown banks, control banks) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks, and a linear position indicator (LVDT) which indicates the actual rod position.[2] At rod positions ≥ 200 steps, full power reactivity worths of the control rods are sufficiently small such that a 15-inch indicated misalignment from the rod bank has no significant effect on the incore power distribution and is therefore allowable. For rod positions < 200 steps, maintaining indicated rod position within 7.5 inches of the average of the indicated bank position provides an enforceable limit which assures design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.1.5 are acceptable because the static misalignment of a single RCCA has been analyzed in Section 15 of the UFSAR and it has been demonstrated that the maximum credible misalignment of a single control rod above or below its bank does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

4.5.2.2 At monthly intervals during power operations each valve (manual, power operated, or automatic) in the safety injection (low and high pressure) and containment spray system flow paths that is not locked, sealed or otherwise secured in position shall be verified as correctly positioned.

Basis

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1)(2)(4)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked each shift and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplished in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupt the logic matrix output to the master relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the logic matrices to the master relay is complete is accomplished by a continuity check. In