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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 34 TO FACILITY OPERATING LICENSE NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

I. INTRODUCTION

By letter dated May 23, 1984, South Carolina Electric and Gas Company requested a change to Technical Specifications to allow installation of a P-9 interlock which would prevent a direct reactor trip following a turbine trip (anticipatory reactor trip) at or below 50% reactor power. The purpose of this change is to prevent needless challenges to the reactor protection system and unnecessary transients during reactor startup. Additional information relating to this request was provided by letter dated November 27, 1984.

II. EVALUATION

The V. C. Summer steam dump system, including condenser dump valves, steam generator power relief valves (safety grade), and atmospheric dump valves (non-safety grade) has sufficient capacity to pass 85% of the main steam flow at full load temperature and pressure. A turbine trip can occur due to a generator trip, a malfunction of the turbine or its ancillary systems (e.g., low hydraulic or oil pressure, turbine overspeed, excessive vibration, low condenser vacuum), high-high steam generator level, and other secondary or primary plant malfunctions. Also, a loss of electrical load would probably result in a turbine overspeed trip. Normally, the steam dump system would accommodate the excess steam generation, preventing any significant increase in reactor coolant system (RCS) temperature and pressure. However, in the event of a failure of the steam dump valves to open following a loss of load or turbine trip, secondary pressure would rise, which may result in lifting the steam generator safety valves (SGSVs). The RCS pressure rise would be limited by pressurizer spray if the reactor coolant pumps (RCPs) are operating, or by the power operated relief valves (PORVs). Above 50% power, the turbine trip would also trip the reactor.

If the anticipatory reactor trip is bypassed at or below 50% power, the reactor may trip on high RCS pressure or overtemperature ΔT. A loss of offsite power (LOOP) may also occur during the bus transfer following turbine trip, resulting in a loss of RCS flow transient. The licensee considers this transient as bounding if it occurs at full power since it results in the minimum acceptable DNBR. The pressurizer safety valves and SGSVs are sized to protect the RCS and steam generators against overpressure, respectively, for all load losses without assuming operation of the steam dump system, pressurizer spray, PORVs, automatic rod control and anticipatory reactor trip.

In order to justify bypassing the anticipatory reactor trip at and below 50% power, the licensee submitted analyses for three cases involving turbine trip. The LOFTRAN, FACTRAN and THINC codes were used. Cases 1 and 2 assume turbine trip at 60% power, minimum reactivity feedback, manual reactor control, and no credit for steam dump, main and auxiliary feedwater. Case 1 assumes actuation of the PORVs and SGSVs to limit primary and secondary pressure increase. The reactor trips at 30 seconds due to bus undervoltage and LOOP occurs at this time. Case 2 is the more severe case. It assumes that the PORVs are unavailable and the LOOP occurs at 6.5 seconds to produce the most limiting transient with respect to departure from nucleate boiling (DNB). The reactor trips at 8.0 seconds due to high pressure. The pressurizer safety valves are actuated. Minimum departure from nucleate boiling ratio (DNBR) (2.5) and peak primary pressure (2523 psia) occur in 9.5 seconds. For both cases the RCS pressure is maintained at less than 110% of design pressure and the DNBR is maintained above the acceptable 95 percent probability at a 95 percent confidence level (95/95) DNBR limit of 1.3. The acceptance criteria for anticipated operational occurrences are thus met.

In order to demonstrate that deletion of the anticipatory reactor trip at or below 50% power does not significantly increase the probability of a small break loss of coolant accident (SBLOCA) resulting from a stuck open PORV, the licensee has performed a "better estimate transient" analysis. (Case 3) This analysis assumes turbine trip at 50% reactor power and operability of the network control. Beginning of core life reactivity feedback was assumed. The results of this case showed smooth control rod insertion to hot shutdown conditions without reactor trip, and a peak pressure of about 2300 psia, well below the PORV setpoint. The licensee concludes that installation of the P-9 interlock to prevent an anticipatory reactor trip at or below 50% power would not substantially affect the probability of a SBLOCA resulting from a stuck open PORV. We concur with this conclusion based on the licensee's analyses, which indicate that, except for unusual conditions, the PORVs would not be actuated for turbine trip at and below 50% power, (due to the high capacity of the V. C. Summer steam dump system), and challenges to the reactor protection system would be reduced.

The configuration of the bistable/relay driver board and transformer, which receives the input to the P-9 interlock circuitry from the power range detectors, is the same as that for the P-8 interlock circuitry currently installed at the V. C. Summer Nuclear Station. Therefore, isolation of the P-9 modification is in accordance with existing plant design. Therefore, from the above evaluation, the staff concludes that the proposed change is acceptable.

III. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

IV. CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (49 FR 33370) on August 22, 1984, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

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

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Dated: November 30, 1984