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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 97 TO PROVISIONAL OPERATING LICENSE NO. DPR-13

SOUTHERN CALIFORNIA EDISON COMPANY AND

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-206

1.0 INTRODUCTION

In a letter dated November 12, 1986 the licensees provided revised safety analyses for postulated loss of main feedwater and feedwater line break events. Previous analyses assumed operability of the steam/feedwater flow mismatch reactor trip to mitigate these events. Failure of pressure transmitter PT-459 on July 30, 1986 resulted in the identification of a single failure which could make this trip inoperable. The licensees, therefore, reanalyzed loss of main feedwater and feedwater line break events without assuming operability of the steam/feedwater flow mismatch reactor trip. In order that the results from the loss of main feedwater analysis fall within the original design basis, it is necessary that the pressurizer high level trip set point be reduced from 27.3 feet to 20.8 feet. The licensees changed this set point on August 1, 1986 and instituted administrative controls to maintain the lowered set point. At the request of the NRC, the licensees then proposed formal changes to the Technical Specifications by application dated November 12, 1986. The pressurizer high level trip will be maintained at the lower level until the steam/feedwater flow mismatch trip circuitry has been modified for protection from single failure.

2.0 EVALUATION

Reanalyses of the loss of main feed and feedwater line break events were done using the Westinghouse LOFTRAN computer code which has been approved by the NRC staff. Following loss of main feedwater, the decrease in primary to secondary heat flow caused the reactor system pressure, temperature, and pressurizer level to increase. At about 30 seconds, reactor system pressure reached a plateau of 2190 psig which is the set point of the pressurizer relief valves.

Reactor trip occurred at 62.5 seconds when the revised high pressurizer level trip was reached. Had the pressurizer relief valves been assumed to fail closed, the reactor system pressure would have increased to the safety valve set point causing an earlier reactor trip. Pressurizer level increased until 1729 seconds into the event when heat removal by the auxiliary feedwater system exceeded reactor decay heat. The pressurizer level did not reach the top of the pressurizer; therefore, no liquid would flow through the pressurizer relief or safety valves.

The Standard Review Plan (NUREG-0800) requires that reactor system pressure remain below 110% of design or 2750 psig for loss of main feedwater events. This criterion is met since the reactor system pressure did not increase above the pressurizer relief valve set point of 2190 psig. Had the relief valves not functioned, the pressure would be limited to 2500 psig which is the set point of the pressurizer safety valves. The pressurizer safety valves' capacity exceeds that of the relief valves.

For a postulated break of a main feedwater line, an almost complete water loss from the affected steam generator was calculated to occur within the first 20 seconds. This loss of secondary system heat sink produced an increase in reactor system temperature increasing the compensated reactor system low pressure trip set point which produced reactor trip in 20.3 seconds. Following reactor trip the two intact steam generators would blow down through connecting steam lines to the affected steam generator and out the break. The steam lines at San Onofre Unit 1 are not equipped with main steam isolation valves. The blowdown of the intact steam generators and the reactor trip was calculated to produce cooling and depressurization of the reactor systems until all secondary coolant was lost at about 100 seconds. The steam driven auxiliary feedwater pump was assumed not to function as a result of the loss of steam pressure. The motor driven auxiliary feedwater pump would autostart and would provide adequate cooling water to the secondary system to remove decay heat. If a simultaneous loss of offsite power were also assumed, manual operator action in the control room would be required to load the motor driven auxiliary feedwater pump onto the emergency diesel. The licensees conservatively assumed that 20 minutes would be required to begin auxiliary feedwater delivery to the intact steam generators if manual loading of the motor driven pump to the emergency diesel were required. Additional backup cooling is provided by a second motor driven auxiliary feedwater pump which can be started remotely and by safety grade charging pumps which can inject cooling water into the reactor system even at high pressures.

During the 20 minute delay before auxiliary feedwater was assumed to reach the intact steam generators, the pressurizer level would gradually increase so that at 1000 seconds liquid flow would occur through the pressurizer relief or safety valves. The NRC staff is currently evaluating the ability of the safety valves to reseat after passing liquid. If the valves did not reseat however, adequate cooling could be supplied to the reactor core by the emergency core cooling system (ECCS) to replenish that lost through the leaking valve.

The Standard Review Plan (NUREG-0800) concludes that calculated reactor system pressures less than 110% of design or 2750 psig are acceptable following a feedwater line break accident. This criterion is met since the reactor system pressure did not increase above the pressurizer relief valve set point of 2190 psig. Had the relief valves not functioned, the pressure would be limited to 2500 psig which is the set point of the pressurizer safety valves.

The staff concludes that with the pressurizer high level reactor trip set point reduced as proposed by the licensees, the core and reactor system will be adequately protected even if the steam/feedwater flow mismatch reactor trip fails to function.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use 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 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.

4.0 CONCLUSION

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 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.

5.0 ACKNOWLEDGEMENT

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