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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO JUSTIFICATION FOR INTERIM PLANT OPERATION REGARDING ESF SINGLE FAILURE VULNERABILITY SOUTHERN CALIFORNIA EDISON COMPANY SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1 DOCKET NO. 50-206

1.0 INTRODUCTION

On October 16, 1987, the licensee submitted an ESF single failure analysis for the staff to review. The submittal included a description of the scenarios for which a single failure of an ESF function would result in consequences not bounded by the analyses of record. Each scenario includes a specific justification for continued operation which referenced a better estimated analysis case and additional operator actions. In these analyses, credit was taken for realistic plant behavior and existing conditions of Moderator Temperature Coefficient (MTC). The MTC used in the analysis was applicable for core burn-up until December 14, 1987. The submittal also provide a description of the operator actions which have been identified to correct equipment misoperations resulting from the postulated single failure.

Based on our review of the above stated licensee's submittal, the staff issued an SER which concluded that the licensee's justification for continual plant operation, until modifications to the affected systems are implemented during the upcoming refueling outage, is acceptable. However, since the better estimated analyses were performed with an MTC which is be valid only to December 14, 1987, additional analyses were required for staff evaluation and approval in order to justify continuous plant operation after December 14, 1987.

On December 1 and 8, 1987, the licensee submitted the updated main steam line break analysis using End of Life (EOL) MTC curve to support its justification of interim plant operation until the forthcoming refueling outage.

2.0 EVALUATION

On December 1, 1987, the licensee submitted the results of an updated better estimated main steam line break analysis. MTC curve applicable to EOL was used in this analysis. In response to the staff request, the licensee in its letter dated December 8, 1987, provided additional information to support its better estimated analysis which demonstrated that the results of a postulated main steam line break accident at EOL meet the acceptance criteria of the event.

The licensee identified the better estimated assumptions which differ from the design basis San Onofre Unit 1 main/steam line break as follows:

1. Credit was taken for charging pump flow to deliver borated water to the Reactor Coolant System (RCS). The safety injection sequences realigns a charging pump to provide flow to the RCS from the refueling water storage tank (RWST).

2. The end of life shutdown margin value assuming all control rods in was used.

3. The Reactor Coolant Pumps (RCPs) were assumed to trip, consistent with actual plant behavior. The RCPs would trip on turbine/generator trip. Also, the RCPs get a trip signal from the safety injection sequences.

4. The addition of thick metal heat to RCS was assumed during system cooldown.

The licensee stated that the resulting RCS cooldown for this transient was outside the applicability range of the W-3 DNB correlation. Therefore, the Machbeth DNB correlation was used in this main steam line break analysis. The Machbeth correlation generated a limit on DNBR of 1.37 at low RCS pressure conditions for San Onofre Unit 1. The Machbeth DNB correlation has been used in main steam line break analysis for other PWR plants. The staff considers the use of this DNB correlation in a better estimated analysis for San Onofre Unit 1 acceptable.

The limiting main steam line break analyzed is a double ended rupture of a steam line outside containment. This would cause all three steam generators to blowdown and rapidly cool the RCS. This rapid RCS cooldown would result in a safety injection signal (SIS). A main feedwater isolation valve is assumed to fail to close upon receipt of the SIS and the result is the diversion of the SI flow to the steam generators instead of feeding RCS. The redundant main feedwater isolation valves are assumed to fail to open due to the adverse conditions from the steam line break in the vicinity of the valves. The SI flow to the steam generators enhances the RCS cooldown. Upon receipt of the SIS, the charging pump realigns to the RWST and delivers borated water to the RCS. The rapid cooldown of the RCS causes reactivity insertion due to the MTC and doppler temperature parameters. The reactivity insertion eventually causes a return to power. Ten minutes after transient initiation, the operator actions terminate the SI flow to the steam generators. The reactivity is eventually turned around due to doppler power feedback and the boron injected into the RCS from the charging pump. Peak power of approximately 6% of rated power is reached at about 680 seconds. DNBR remains above the limit of 1.37 during the entire transient.

The licensee also evaluated the previously identified single failure scenarios resulting in less severe cooldown transients which may have higher terminal pressures. The evaluation confirmed that these cases are bounded by the previous San Onofre Unit 1 analysis regarding pressurized thermal shock (PTS).

3.0 CONCLUSION

The acceptance criteria for a postulated main steam line break accident permit some fuel failure with radiological consequences not exceeding 10 CFR 100 limits. The results of the licensee's better estimated analysis show that there will be no DNB following a main steam line break accident. Therefore, sufficient safety margin exists in the plant design. The realistic assumptions used in the better estimated analysis are reasonable. Also, the procedures provided for operator to mitigate the postulated single failure scenarios have been reviewed and accepted. Therefore, we conclude that the licensee's justification for continued plant operation, until modifications to the affected systems are implemented during the upcoming refuel outage, is acceptable.