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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.122TO PROVISIONAL OPERATING LICENSE NO. DPR-13

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-206

1.0 INTRODUCTION

By letters dated January 11, 1989 (Ref. 1), January 27, 1989 (Ref. 2), March 4, 1989 (Ref. 3), March 11, 1989 (Ref. 4), and April 1, 1989 (Ref. 5), Southern California Edison Company (SCE) submitted Amendment Application No. 164 to Provisional Operating License DPR-13 for San Onofre Nuclear Generating Station Unit 1 (SONGS-1). Amendment Application No. 164 consists of Proposed Change No. 204 which is a request to revise the Technical Specifications for SONGS-1. Specifically, Proposed Change No. 204 would revise Technical Specifications 2.1, "Reactor Core," 3.3.3, "Refueling Water Storage Tank," 3.5.1, "Reactor Trip System Instrumentation," 3.5.2, "Control Group Insertion Limits," 3.10, "In-Core Instrumentation," 3.11, "Continuous Power Distribution Monitoring," and 4.1.1, "Operational Safety Items."

2.0 EVALUATION

Due to tube degradation, some of the tubes in the SONGS-1 steam generators have been removed from service through plugging while others have been sleeved and returned to service. Since sleeving a sufficient number of tubes has the same effect as plugging a tube, a hydraulic equivalence between sleeving and plugging can be made for analysis purposes. Therefore, the safety analyses can be performed with an assumed equivalent level of tube plugging such that the actual equivalent levels of plugging are conservatively modeled.

Additional steam generator tubes were plugged during the Cycle 10 refueling outage such that the actual levels of equivalent plugging now exceed the previously analyzed value. The actual plugging and sleeving levels are given below where the equivalent plugging for the sleeved tubes is based on a ratio of 18.6 sleeves per plug.

	Steam Generator		
	A	B	C
Tubes Plugged	11.47%	12.07%	13.05%
Tubes Sleeved (Equiv.)	3.06%	2.90%	2.95%
Total Equiv. Plugging	14.53%	14.97%	16.00%

Since the 15% tube plugging level has been exceeded by the tube plugging performed during the refueling outage, a reevaluation of the Updated Final Safety Analysis Report (UFSAR) Chapter 15 transients and accidents was made by the licensee and the fuel vendor, Westinghouse, to determine if the current analyses continue to be applicable for operation with up to 20% plugging in any steam generator and reduced T_{avg}. Based on this reevaluation as presented in Amendment Application No. 164, the staff concurs that, except for the loss of coolant accident (LOCA), no reanalyses are required to support the increase in tube plugging level for reduced T_{avg} operation. Although a reevaluation of the steam generator tube rupture event resulted in slightly higher offsite doses, the effect of up to 20% tube plugging would not increase the doses reported in the UFSAR above the 10 CFR 100 regulatory limits. The LOCA analyses results indicate higher peak clad temperatures for reductions in the vessel average temperature. Therefore the 20% maximum tube plugging LOCA analysis may not bound plant operation with increased tube plugging at the reduced vessel average temperature condition. Reanalysis is also required to address the increase in safety injection delay time for the steamline break core response analysis. Also, reanalysis of the locked rotor/shaft break event is required to address the unavailability of the RCS low flow reactor trip function assuming a single active failure of the low flow channel located in the loop with the affected reactor coolant pump.

The current safety analysis for the steamline break core response is applicable for SONGS-1 operation on the reduced T_{avg} program with 20% tube plugging. However, a reanalysis is required to address the increase in safety injection delay time of 4 seconds. The steamline break core response transient used in the departure from nucleate boiling ratio (DNBR) evaluation is sensitive to the time that safety injection water reaches the RCS. Therefore, reanalyses were performed assuming an increase in the safety injection delay time to 26 seconds from 22 seconds. The results show that the DBNR remains above the limit value, ensuring that DNB will not occur. Therefore, no releases of fission products from the fuel will result from a hypothetical break with a total safety injection delay time of 26 seconds from the time the low pressurizer pressure safety injection setpoint is reached until the safety injection pumps reach full speed. This reanalysis also demonstrated that a boron concentration of 1500 ppm for the water in the safety injection lines is acceptable. This reduced boron concentration accounts for the leakage at the interface of the safety injection system and the main feedwater system. The mass/energy releases following a steamline break are not very sensitive to any small delay in time that safety injection would reach the RCS. Therefore, the steamline break mass/energy release analyses remain applicable for an increase in the safety injection delay time of 4 seconds.

A reanalysis of the locked rotor/shaft break event was performed in order to address the unavailability of the RCS low flow reactor trip function assuming a single active failure of the low flow channel located in the loop with the affected reactor coolant pump. This reanalysis assumed that a reactor trip is actuated on RCP breaker opening due to either overcurrent (locked rotor) or undercurrent (shaft break) to the RCP motor. The existing reactor protection system for SONGS-1 provides a reactor trip on RCP breaker opening due to overcurrent to the RCP motor. However, for a shaft break, an overcurrent condition is not expected and, therefore, SCE has committed to modifying this function so that a breaker opening trip will also occur for an undercurrent condition to the RCP motor. The results indicate that as long as control rod motion begins within 6.1 seconds following the locked rotor/shaft break, all of the applicable safety criteria are met. SCE has, therefore, included the undercurrent and overcurrent RCP breaker trips in Technical Specification Table 2.1, "Maximum Safety System Settings," to prevent core damage from the locked rotor and shaft break events. These trip settings have been selected to meet the analysis assumptions that control rods begin to drop within 6.1 seconds after the initiating event.

The LOCA was reanalyzed to bound operation with up to 20% equivalent steam generator tube plugging at reduced thermal conditions and an increase in safety injection initiation time. The linear heat rate was 13.2 Kw/ft and the heat flux hot channel factor (F_Q) was 2.78. The interim Acceptance Criteria (IAC) assumptions and the 1971 IAC analytical models were used. Since SONGS-1 fuel is clad with stainless steel rather than zirconium, adherence to the IAC is acceptable. The limiting break was found to be the double ended cold leg guillotine (DECLG) with a discharge coefficient of 0.8. The limiting peak clad temperature was 2260°F which satisfies the IAC limit of 2300°F.

3.0 TECHNICAL SPECIFICATION CHANGES

(1) TS 2.1 Table 2.1, "Maximum Safety System Settings," has been modified to include maximum safety system settings for overcurrent and undercurrent trips of the RCP breakers. These trips protect the core following a locked rotor/sheared shaft event and the trip settings have been selected to meet the analysis assumptions that rods begin to drop within 6.1 seconds after the initiating event. In addition, an undervoltage trip has been added. In loss of forced coolant flow events, this trip provides redundancy and diversity to the low flow reactor trip. The changes are, therefore, acceptable.

(2) TS 3.3.3 has been modified to include a new minimum safety injection line boron concentration limit of 1500 ppm. This new limit has been analyzed in the steam line break event and the results have demonstrated continued compliance with the NRC acceptance criteria. The change is, therefore, acceptable.

(3) TS 3.5.1 Table 3.5.1-1, "Reactor Trip System Instrumentation," has been modified to include RCP breaker position (above 50% of full power). This change is acceptable for the reasons given in item (1) above.

(4) TS 3.5.2 Bases has been modified to include a more restrictive limit of 13.2 Kw/ft on the allowable LHR which affects the peaking factor and axial offset limits. This change reflects the LOCA reanalysis to allow continued operation on the reduced T_{avg} program with up to 20% steam generator tube plugging and is, therefore, acceptable.

(5) TS 3.10, "Incore Instrumentation," has had the mode applicability and action statements revised. This change assures that extended operation without core monitoring does not occur and that the reactor remains in a previously analyzed condition. The change is, therefore, acceptable.

(6) TS 3.11, "Continuous Power Distribution Monitoring," has revised the parameters in the axial offset equation as well as the applicability and action statements. The changes to the axial offset equation merely reflect current core parameters and are acceptable. The changes to the applicability and action statements represent an updating to prevent extended operation below 90% power with no axial offset monitoring. Since without monitoring, extended periods of time in this condition could lead to core limits being exceeded, these changes are also acceptable.

(7) TS 4.1.1, "Operational Safety Items," has been modified to include specific surveillance requirements on RCP breaker position and the refueling water storage tank (RWST) water samples and contained water volume. Since credit for the RCP breaker open reactor trip is taken to provide redundancy to the low flow trip, surveillance is required. The changes to the RWST surveillances ensure that safety system parameters are properly monitored. Therefore, these changes are acceptable.

The staff has reviewed Proposed Change No. 204 to revise the San Onofre Unit 1 Technical Specifications. Based on the preceding safety evaluation, these changes have been found to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on April 13, 1989 (54 FR 14893). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

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,

(2) such activities will be conducted in compliance with the Commission's regulations and (3) 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: April 14, 1989

5.0 REFERENCES

1. Letter from Robert Dietch (SCE) to NRC, submitting Amendment Application No. 164, January 11, 1989.

2. Letter from Kenneth P. Baskin (SCE) to NRC, submitting Supplement 1 to Amendment Application No. 164, January 27, 1989.

3. Letter from Kenneth P. Baskin (SCE) to NRC, submitting Supplement 2 to Amendment Application No. 164, March 4, 1989.

4. Letter from Kenneth P. Baskin (SCE) to NRC, submitting Supplement 3 to Amendment Application No. 164, March 11, 1989.

5. Letter from F.R. Nandy (SCE) to NRC, submitting additional information regarding Amendment Application No. 164.