DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS
PROPOSED CHANGE NO. 204 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE DPR-13

This is a request to revise Appendix A Technical Specifications 2.1, "REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature," 3.5.2, "Control Group Insertion Limits," 3.11, "Continuous Power Distribution Monitoring," 3.3.3, "Refueling Water Storage Tank," 4.1.1, "Operational Safety Items," and 4.2.1, "Safety Injection and Containment Spray System Periodic Testing."

Reason for Proposed Change

In conjunction with the Cycle 10 Reload Safety Evaluation for plant operation on the reduced Tavg program with 20% steam generator tube plugging, changes to the Technical Specifications for SONGS 1 are necessary. These changes are related to the reanalysis of the Large Break Loss of Coolant Accident, Reactor Coolant Pump Shaft Break Event and Main Steam Line Break Event performed concurrently with the Cycle 10 reload.

The Large Break Loss of Coolant Accident (LBLOCA) was reanalyzed since the existing analysis only covered two cases:

LBLOCA on the nominal Tavg program with 20% tube plugging

LBLOCA on the reduced Tavg program with 15% tube plugging

Since the steam generator tube plugging levels have exceeded 15%, reanalysis was necessary to allow SCE to continue operation on the reduced Tavg program. The reanalysis of the LBLOCA for operation on the reduced Tavg program with up to 20% steam generator tube plugging will require a change to the allowable linear heat rate of the fuel rods of about 2 kW/foot which affects the peaking factor and Axial Offset Limits. This requires changes to the basis of technical specification 3.5.2 "Control Rod Insertion Limits" and technical specification 3.11 "Continuous Power Distribution Monitoring".

The Reactor Coolant Pump (RCP) Shaft Break Event was reanalyzed with an assumed single failure of the reactor coolant system (RCS) low flow reactor trip. Since the low flow trip is designed with 1 channel per loop, a single failure could prevent a reactor trip. The RCP shaft break event was reanalyzed at the nominal Tavg program in 1987 with reactor trip generated by the variable low pressure (VLP) trip. In 1988, Westinghouse informed SCE that on the reduced Tavg program the existing VLP calculator would not generate a reactor trip before the acceptance criteria for this event was exceeded. Subsequently, Westinghouse reanalyzed the RCP shaft break with a revised VLP setpoint and has demonstrated acceptable results. This requires a change to Table 2.1 "Maximum Safety System Settings" of technical specification 2.1, "REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature."

The Main Steam Line Break event (MSLB) was reanalyzed to model the injection of borated water through a single injection path and to permit a reduced boron concentration for the water in the safety injection lines. A single injection path results from the resolution of single failure concerns related to the operability of the safety injection valves MOV 850 A, B and C, for MSLB outside containment. This assumption in the analysis does not affect any technical specifications. The existing technical specifications require the safety injection lines to be maintained at the technical specification boron concentration for the Refueling Water Storage Tank. Leakage at the interface of the safety injection system and the main feedwater system dilutes the water contained in the lines and required frequent recirculation of the safety injection system. To account for the dilution of this water and determine an acceptable limit, SCE had Westinghouse reanalyzed this event. The reanalysis determined that a boron concentration of 1500 ppm for the water in the safety injection lines was acceptable. This requires a change to technical specifications 3.3.3, "Refueling Water Storage Tank," 4.2.1, "Safety Injection and Containment Spray System Periodic Testing" and Table 4.1.2 "Minimum Equipment Check and Sampling Frequency" of technical specification 4.1.1, "Operational Safety Items" to include the specific requirements for the safety injection lines.

Existing Specifications

To be provided later.

Proposed Specifications

To be provided later.

Safety Analysis

The results of the transient reanalysis discussed above will be documented in the Reload Safety Evaluation Report for Cycle 10 operation of San Onofre Nuclear Generating Station Unit 1. For those transients which are being reanalyzed as part of the Cycle 10 reload, Westinghouse will use standard conservative methods, assumptions and acceptance criteria consistent with previous safety analysis. The reanalysis will demonstrate that the applicable acceptance criteria and design limits are not exceeded. Other events either were not reanalyzed because they remain bounded by the previous safety analysis of record or were reanalyzed and the reanalysis did not result in changes to the technical specifications.

Therefore, it can be concluded that: (1) the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Supplemental information on the impact to safe operation of San Onofre Nuclear Generating Station Unit 1 will be describe in detail for each part of this proposed change as soon as it is available.