SAFETY EVALUATION

AMENDMENT NO. 26 TO NPF-10

AMENDMENT NO. 15 TO NPF-15

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

Introduction

Southern California Edison Company (SCE), on behalf of itself and the other licensees, San Diego Gas and Electric Company, the City of Riverside, California, and The City of Anaheim, California has submitted several applications for license amendments for San Onofre Nuclear Generating Station, Units 2 and 3. The evaluations of four such requests are presented below.

I. By letter dated June 27, 1984, SCE requested that the NRC revise Technical Specification 3/4.9.6, Refueling Machine (reference PCN-50). Technical Specification 3/4.9.6 defines the operability and surveillance requirements for the refueling machine to ensure that (a) the refueling machine will be used for all movements of fuel assemblies and Control Element Assemblies (CEAs), (b) the refueling machine has sufficient load capacity to lift a fuel assembly, and (c) the core internals and pressure vessel are protected from excessive lifting forces in the event of inadvertent mechanical interference during fuel handling operations. At present, Technical Specification 3/4.9.6 requires that the refueling machine be used for all movement of the CEAs. However, during refueling, the Control Element Drive Motor (CEDM) drive shaft extensions must be manually uncoupled and recoupled. Coupling and uncoupling is verified by weighing of the CEAs. These operations involve small movements of the CEAs. At present, these small movements of CEAs are prohibited by Technical Specification 3/4.9.6 because they are not done by the refueling machine. However, the refueling machine cannot be used for either coupling, uncoupling or weighing of CEAs. Note that the existing technical specifications allow the four-fingered CEAs to be removed without using the refueling machine. The proposed change would add a note which allows coupling, uncoupling and weighing of CEAs during refueling.

II. By letters dated June 27, 1984 and July 18, 1984, SCE requested that the NRC revise Technical Specification 3/4.9.10 (Refueling) Water Level - Reactor Vessel (reference PCN-179). Specification 3/4.9.10 requires that a minimum water level of 23 feet be maintained above the reactor vessel flange during movements of CEAs or fuel assemblies in the reactor vessel. During refueling, the CEDM drive shaft extensions are uncoupled and recoupled to the CEAs, and the four-finger CEAs are removed using the upper guide structure. Coupling or uncoupling of the CEDM drive shaft extensions involves small movements of the CEAs as does the verification of coupling/uncoupling, and the removal of the four-finger CEAs. Therefore, under the current Technical Specifications, a water level of 23 feet must be maintained above the vessel flange during these operations. However, the design of the tools used to couple and uncouple the CEAs from the CEDM drive shaft extensions requires that the work platform be positioned less than 23 feet above the reactor vessel flange. Verification of CEA coupling/uncoupling is most efficiently accomplished at the time the CEAs are coupled/uncoupled. Thus, the existing specification prohibits coupling, uncoupling, and verification of the CEAs using the tools required for these activities. Also, the design of the upper guide structure lift rig and work platform requires the water level to be less than 23 feet above the reactor vessel flange when latching the four-finger CEAs to the work platform and lifting them into the upper guide structure.

The proposed change adds a note to the applicability for Specification 3/4.9.10 which allows the water level to be lowered to 23 feet above the fuel assemblies rather than the vessel flange (a difference of about 11 feet) during CEA coupling and uncoupling, verification of coupling/uncoupling, and removal of the four-finger CEAs.

III. By letter dated June 27, 1984, SCE requested that the NRC revise Technical Specification 3/4.9.12, Fuel Handling Building Post Accident Cleanup Filter System (reference PCN-180). Specification 3/4.9.12 requires that the Fuel Handling Building Post Accident Cleanup Filter System (FHBPACFS) be operable when irradiated fuel is in the storage pool and defines a number of functional tests which periodically must be conducted to assure such operability. The FHBPACFS includes electrical heaters to maintain the relative humidity at the inlet to the charcoal filters at or below 70% to preserve charcoal adsorber efficiency. Specification 4.9.12.d.3 requires verification that the heater thermal dissipation is within ±5% of the specified rating. The heater ratings contained in the specification are based on the nominal operating voltage. However, when the plant is on line in normal operation, the bus voltages are higher than nominal. Additionally, Specification 4.8.1.1 (Diesel Generator Surveillance Requirements) permits a ±10% bus voltage variation during diesel generator operation. Because the power dissipated by a heater varies with the square of the voltage, small deviations from the nominal voltage (e.g., ±2.5%) will result in heater dissipations outside of the allowable range, thereby rendering the system inoperable.

The proposed change revises Specification 4.9.12.d.3 to allow correction of measured heater dissipation to the nominal voltage for the purpose of determining operability. In addition, the proposed change corrects a typographical error in the specified dissipation for heater E-464. Heater E-464 is actually rated at 28.7 kw versus the 28.4 kw listed currently.

IV. By letter dated June 29, 1984, SCE requested that the NRC revise Technical Specification 3/4.4.4, Steam Generator (reference PCN-141). Technical Specification 3/4.4.4 requires that the steam generator be operable and specifies surveillance requirements to verify steam generator integrity. The current acceptable level of steam generator tube wall thinning shown in Figure 4.4.1 of the Technical Specification is 44% for tube rows 0 through 92 and decreases linearly to 26% in tube row 147. The proposed change will delete Figure 4.4.1 and specify a tube thinning limit of 44% for all steam generator tubes.

Evaluation

I. Revise Technical Specification 3/4.9.6 to allow manual coupling, uncoupling, and weighing of CEAs (PCN-50). The NRC criteria in this area is given in Section 9.1.4 of NUREG-0800 (the Standard Review Plan, or SRP), which discusses acceptance criteria for the fuel handling system. The objectives of the SRP are to preclude criticality accidents and releases of radioactivity. Criticality accidents are, in part, prevented by verification of uncoupling of the CEA extension shafts prior to removal of the upper guide structure, thereby preventing CEA withdrawal when the upper guide structure is removed. The proposed change would permit small movements of the CEAs during refueling due to manual coupling/uncoupling and verification of uncoupling, thereby reducing the probability of accidental criticality.

The proposed specification maintains the requirement to use the refueling machine for significant movements of fuel, requires the refueling machine to have sufficient capacity to lift a fuel assembly and requires an overload cutoff to assure that excessive forces are not applied. The NRC staff has reviewed the proposed change and finds that it meets the SRP acceptance criteria and is therefore acceptable.

II. Revise Technical Specification 3/4.9.10 to allow the water level during refueling to be reduced to 23 feet above the fuel (PCN-179). The NRC criteria for minimum water level in the reactor vessel during refueling is defined in the Bases Section of NUREG-0212, Revision 2, Standard Technical Specifications (STS) for Combustion Engineering Pressurized Water Reactors. Specifically, Bases Section B 3/4.9.10 requires that sufficient water depth (23 feet) be available to remove 99% of the assumed 10% iodine gap activity which would be released by an irradiated fuel assembly striking the reactor vessel flange and rupturing. However, with the fuel assemblies seated in the reactor vessel, as will be the case during CEA coupling, uncoupling, and weighing, and during removal of the four-finger CEAs, no fuel damage could occur above the top of the fuel. The proposed specification requires that 23 feet of water be maintained above the top of the fuel rather than the vessel flange as previously required. This will continue to ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from a fuel assembly damaged by any conceivable accident. Therefore, the proposed change meets the acceptance criteria delineated in the Bases of the STS and is acceptable.

III. Modify Technical Specification 3/4.9.12 to allow correction of the measured heat dissipation by the FHBPACFS heater to the nominal voltage (PCN-180). The NRC criteria for this change is given in SRP Section 9.4.2, Spent Fuel Pool Area Ventilation System (SFPAVS). This section references Regulatory Guide 1.52 which recommends that heaters be installed in the SFPAVS upsteam of the charcoal adsorbers. The heaters must have sufficient capacity to maintain the relative humidity below 70%, thereby preserving the efficiency of the charcoal adsorber. The proposed change effects the manner in which the results of the heater dissipation surveillance tests are evaluated to accommodate the allowed variations for the nominal bus voltage which may exist at the time the surveillance is conducted. The proposed change does not reduce the heater dissipation requirements and maintains the 70% relative humidity acceptance criteria. Therefore, the proposed change satisfies the SRP acceptance criteria and is acceptable.

Additionally, the proposed change increases the required dissipation for heater E-464 from 28.4 kw to 28.7 kw. This corrects a typographical error. Therefore, this part of the proposed change is acceptable.

IV. Revised Technical Specification 3/4.4.4, Steam Generator, to change the tube thinning criteria to 44% for all tubes (PCN-141). The original analysis upon which this Technical Specification is based established the structural adequacy of the San Onofre 2 and 3 steam generator tubes and tube supports, when subjected to various hypothetical accident conditions. It was determined that the limiting event was a combination of a Loss of Coolant Accident (LOCA) and Safe Shutdown Earthquake (SSE). The calculated stresses occurring in the steam generator tube walls as a result of the limiting event were compared to the maximum allowable stresses as defined by the NRC staff's criteria (Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes). The analysis indicated that degradation of up to 64 percent is acceptable for both the straight portion of all tubes and the "U" bend region for the majority of tube rows. The outer tubes rows experienced significant stress in the "U" bent region, due to the combination of hydraulic loads associated with blowdown of the primary system as a result of the LOCA and earthquake-induced accelerations resulting from the SSE. Thus, in the outer tube bundle bend region, the allowable degradation decreased linearly from 64 percent to a minimum of 46 percent at the outermost row. The values in the technical specification vary from 44% to 26%. The 20% differences between the allowable degradation and the Technical Specification limit represents margin for instrument error and degradation between inspections.

However, a revised analysis has shown that tube degradation of 64% is acceptable for all rows of tubes. Thus, the proposed change will remove unnecessary conservatism from the assumptions used in the original analysis and thereby establish a more accurate steam generator tube thinning limit. In addition, the proposed change prevents unnecessary (a) plugging of tubes, (b) associated high personnel radiation exposure and (c) decreases in the steam generator heat transfer surface area.

The revised analysis of the limiting LOCA/SSE scenario includes the frictional or binding restraint on the tubes provided by the vertical tube supports in the horizontal tube run on top of the "U" tube span; this was previously neglected in the original calculations. In addition, the LOCA and SSE peak loads were combined by the square-root-of-the-sum-of-the-squares (SRSS) method; these loads were combined by addition in the original calculations. The use of SRSS, under appropriate circumstances, is an acceptable method under the SRP for combining loads. The combination of a LOCA and SSE is still the limiting event. The revised analysis shows that tube degradation of up to 64 percent is acceptable for all steam generator tubes in meeting the criteria of Regulatory Guide 1.121, which results in the proposed Technical Specification limit of 44% for all tubes, based on the 20% margin previously used.

Section 5.4.2.2 of the SRP reference Regulatory Guide 1.83 which specifies inservice inspection criteria for determining steam generator operability. The proposed change prescribes steam generator tube thinning criteria which was developed in accordance with Regulatory Guide 1.83 and the SRP.

The NRC staff has reviewed the revised analysis of tube stresses and has concluded that the proposed change meets the above-described staff criteria for steam generator tube thinning, and therefore is acceptable.

Contact With State Official

The NRC staff has advised the Chief of the Radiological Health Branch, State Department of Health Services, State of California, of the proposed determinations of no significant hazards consideration. No comments were received.

Environmental Consideration

These amendments involve changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendments involve no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings. Accordingly, the amendments meet 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 these amendments.

Conclusion

Based upon our evaluation of the proposed changes to the San Onofre Units 2 and 3 Technical Specifications, we have concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and 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. We, therefore, conclude that the proposed changes are acceptable.

Dated: October 26, 1984

DL:LB#3	DL:LB#3
HRood/yt	GWKnighton
10/2/84	10/25/84

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

ISSUANCE OF AMENDMENT NO. 26 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

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