October 24, 2003

Mr. Harold B. Ray
Executive Vice President
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS ON EMERGENCY CORE COOLING SYSTEM (ECCS) SURVEILLANCE REQUIREMENT (TAC NOS. MB8123 AND MB8124)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. NPF-10 and Amendment No. 181 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 25, 2003.

The amendments revise TS 3.5.2, "ECCS - Operating," Surveillance Requirement 3.5.2.5. Specifically, the changes replace the requirement to verify specific surveillance test values for the ECCS pumps with the requirement to verify the developed head for each ECCS pump in accordance with the Inservice Testing Program. These changes are requested to implement recommendations of the Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432, Revision 2.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Bo M. Pham, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 190 to NPF-10
2. Amendment No. 181 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

October 24, 2003

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. NPF-10

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee), dated March 25, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 190, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: October 24, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.5-6	3.5-6

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 181
License No. NPF-15

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated March 25, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-15 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 181, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 24, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 181

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3.5-6	3.5-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 181 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By application dated March 25, 2003, Southern California Edison Company (the licensee), requested changes to the Technical Specifications (TSs) for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The proposed changes would revise TS 3.5.2, "Emergency Core Cooling Systems (ECCS) - Operating," Surveillance Requirement (SR) 3.5.2.5. Specifically, the changes replace the requirement to verify specific surveillance test values for the ECCS pumps with the requirement to verify the developed head for each ECCS pump in accordance with the Inservice Testing (IST) Program. These changes are requested to implement recommendations of the Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432, Revision 2.

The purpose of the ECCS is to inject borated water into the Reactor Coolant System (RCS) to cool the core following a loss-of-coolant accident (LOCA) and to maintain the reactor subcritical following a LOCA or a Main Steam Line Break. The Safety Injection System (SIS) is designed to provide emergency core cooling and reactivity control following any loss of reactor coolant. The SIS accomplishes these functions by providing borated water from the Refueling Water Storage Tank to the RCS by means of the ECCS pumps. Borated water is also provided to the RCS from the Safety Injection Tanks (SITs) in the event that RCS pressure falls below the pressure of the SITs. The ECCS or SIS is actuated automatically by a Safety Injection Actuation Signal. SIS actuation mitigates fuel and clad damage that could interfere with continued effective core cooling, limits fuel clad metal-water reaction to negligible amounts, and ensures that reactivity control with appropriate shutdown margin for stuck rods is maintained under postulated accident conditions.

Periodic surveillance testing of the ECCS pumps verifies that measured pump performance is within an acceptable tolerance of the pump baseline performance and that the test flow and developed head are greater than or equal to the performance assumed in the safety analysis.

Discharge head at design flow is a normal test of charging pump performance required by Section XI of the ASME Code. Such ISTs detect component degradation and incipient failure.

This safety evaluation reviews the licensee's TS change proposal for the ECCS based on guidance from the Standard Review Plan (SRP) Sections 6.3, "Emergency Core Cooling Systems," and 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."

2.0 REGULATORY EVALUATION

The regulatory requirements for which the staff based its acceptance are:

Section 10 CFR 50.46 of Title 10 of the *Code of Federal Regulations*, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, requires the following criteria be met: peak fuel element cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling.

Appendix K to 10 CFR Part 50 defines the required and acceptable features of ECCS evaluation models.

Section 10 CFR 50.55a(f)(4) requires that licensees perform inservice testing of certain pumps and valves designated as Code Class 1, 2, or 3 under the Boiler and Pressure Vessel Code (BPV Code) or the Code for Operation and Maintenance of Nuclear Power Plants (OM Code) promulgated by the American Society of Mechanical Engineers (ASME). Further, 10 CFR 50.55a requires that licensees perform this testing in accordance with the ASME BPV Code Section XI or OM Code, and applicable addenda.

Section 10 CFR 50.36(c)(3) requires that the licensee's technical specifications include surveillance requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.0 TECHNICAL EVALUATION

In its application to revise TS 3.5.2, the licensee is asking to replace the requirement to verify specific test values for ECCS pumps with the requirement to verify the developed head for each ECCS pump in accordance with the IST program. With this change proposal, the NRC staff evaluated whether the licensee's proposal will satisfy regulatory requirements of 10 CFR 50.46, and whether the safety analysis model used by the licensee is acceptable per Appendix K to 10 CFR Part 50.

Also, the staff evaluated whether the changes proposed are acceptable with respect to 10 CFR 50.55a(f)(4), which requires IST to be performed per requirements of the ASME/ANSI [American National Standards Institute] Code and Addenda.

The staff has reviewed the licensee's technical analysis in support of its proposed license amendment, described in Section 4.0 of the licensee's submittal, to determine if it meets the regulatory requirements described above. The findings are as follows:

3.1 Section 10 CFR 50.46 and Appendix K to 10 CFR Part 50

The licensee's Updated Final Safety Analysis Report Sections 15.6.3.3 and 15.10.6.3.3 present the design basis accident analyses associated with postulated LOCAs. LOCAs are accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary at a rate in excess of the capability of the normal reactor coolant makeup system. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the RCS pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished.

To satisfy the ECCS performance requirements of 10 CFR 50.46, the licensee performed safety analyses of ECCS performance under accident conditions. These analyses provide input to the design of the RCS piping and support structures, high pressure safety injection (HPSI) system, low pressure safety injection (LPSI) system, SITs, chemical and volume control system, and to the design of the steam generators and the containment structures.

The review of the applicant's analysis of the spectrum of postulated LOCAs is closely associated with the review of the ECCS, as described in SRP Section 6.3. As a portion of the review effort described in SRP Sections 6.3 and 15.6.5, the NRC staff evaluated whether the entire break spectrum (break size and location) had been addressed; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance have been provided. For postulated break sizes and locations, the staff's review included the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

The calculational framework used for the evaluation of the ECCS system in terms of core behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. Appendix K to 10 CFR Part 50 requires the licensee's evaluation model to comply with the acceptance criteria for ECCS given in 10 CFR 50.46. The licensee's evaluation model was previously documented and reviewed by the staff in NUREG-0712 Safety Evaluation Report for San Onofre, Units 2 and 3 Operation, Section 15.3.5, "Small Break LOCA," and more recently updated in an NRC letter dated February 22, 2000, “San Onofre Nuclear Generating Station, Units 2 and 3 – Issuance of Amendments on Small Break Loss-of-Coolant Accident, Charging Flow and Main Steam Safety Valve Setpoints.” Per the latter, the staff approved the licensee's use of ABB Combustion Engineering's SBLOCA evaluation model, CENPD-137(P), Supplement 2, “Calculative Methods for the C-E Small Break LOCA Evaluation Model” (ABB-CE S2M SBLOCA analysis methodology).

In its application to revise TS 3.5.2, "Emergency Core Cooling Systems (ECCS) - Operating," SR 3.5.2.5, the licensee did not propose changes that would impact the safety analysis approved by the NRC on February 22, 2000. Therefore, the NRC concludes that the requested changes satisfy the regulatory requirements of 10 CFR 50.46, and that the safety analysis model used by the licensee is acceptable per Appendix K to 10 CFR Part 50.

3.2 Section 10 CFR 50.55a(f)(4)

Discharge head at design flow is a normal test of charging pump performance required by Section XI of the ASME Code. The Code requirement for the frequency of such tests is every quarter, as such inservice testing detects component degradation and incipient failures.

The licensee implemented a risk-informed inservice testing (RI-IST) program at SONGS, Units 2 and 3, in March 2000. Per the RI-IST, pump testing would be performed in accordance with the requirements stated in the ASME/ANSI OM, except that the testing frequencies are determined per the methodology outlined by the licensee's alternative testing strategy, incorporating a probabilistic approach to testing periodicity. The frequency of testing for ECCS pumps, however, remains in conformance with the ASME Code requirements, as these pumps are classified as high safety significant components in the SONGS RI-IST Program, and are conducted at the Code-specified frequency using approved Code methods.

The licensee proposed to change the requirement in TS 3.5.2, "ECCS - Operating," SR 3.5.2.5 to replace specific surveillance test values for the ECCS pumps with the requirement to verify the developed head for each ECCS pump in accordance with the IST Program. The staff concludes that the proposed change meets the requirements of 10 CFR 50.55a(f)(4) to perform IST per the ASME Code Section XI or OM Code, and applicable addenda, and is therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change a surveillance requirement. The NRC staff has determined that the amendments involve 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 18285 dated April 15, 2003). Accordingly, the amendments meet 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 the amendments.

6.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: B. Pham

Date: October 24, 2003