July 6, 2001

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 RE: INCREASE IN REACTOR POWER TO 3438 MWt (TAC NOS. MB1623 AND MB1624)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 180 to Facility Operating License No. NPF-10 and Amendment No. 171 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The amendments consist of changes to the Facility Operating Licenses in response to your application dated April 3, 2001, as supplemented by letters dated April 23, May 11, May 25, May 31, and June 25, 2001, to increase the licensed thermal power level and to change the Technical Specifications definition of thermal power for operation of SONGS Units 2 and 3 to 3438 MWt. This corresponds to an increased electrical output of approximately 16 MWe for each unit.

A copy of the related Notice of Issuance for publication in the *Federal Register* and our Safety Evaluation are also enclosed.

Sincerely,

/RA/

Joseph E. Donoghue, Senior Project Manager
Project Directorate IV, Section 2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 180 to NPF-10
2. Amendment No. 171 to NPF-15
3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

July 6, 2001

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DATE	6/22/01	6/15/01	6/14/01	6/20/01	7/5/01	7/5/01	7/5/01	07/05/01

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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.180
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 April 3, 2001, as supplemented by letters dated April 23, May 11, May 25, May 31, and June 25, 2001, 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 Operating License and 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 180, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License
and Technical Specifications

Date of Issuance: July 6, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Replace the following pages of the Facility Operating License No. NPF-10 and Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Operating License	Operating License
3	3
Technical Specifications	Technical Specifications
1.1-5	1.1-5

(3) SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2. Transshipment of Unit 1 fuel between Units 1 and 2 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18, and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No.   , 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.

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. 171
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 April 3, 2001, as supplemented by letters dated April 23, May 11, May 25, May 31, and June 25, 2001, 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 Operating License and 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. 171, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License
and Technical Specifications

Date of Issuance: July 6, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Replace the following pages of the Facility Operating License No. NPF-15 and Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Operating License	Operating License
3	3
Technical Specifications	Technical Specifications
1.1-5	1.1-5

-3- SCE, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

-4- SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

-5- SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

-6- SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3. Transshipment of Unit 1 fuel between Units 1 and 3 shall be in accordance with SCE letters to U.S. Nuclear Regulatory Commission dated March 11, March 18 and March 23, 1988, and in accordance with the Quality Assurance requirements of 10 CFR Part 71.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

a. Maximum Power Level

Southern California Edison Company (SCE) is authorized to operate the facility at reactor core power levels not in excess of full power (3438 megawatts thermal).

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 171 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 letter dated April 3, 2001 (Reference 1), and supplemented by letters dated April 23, May 11, May 25, May 31, and June 25, 2001 (References 2-6), Southern California Edison (SCE or the licensee) proposed changes to Facility Operating Licenses No. NPF-10 and NPF-15 for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The supplemental letters dated April 23, May 11, May 25, May 31, and June 25, 2001, contained clarifying information and did not expand the scope of the proposed amendment as described in the Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action that was published in the *Federal Register* on April 18, 2001 (66 FR 19996).

Presently, SONGS Units 2 and 3 are authorized to operate at the maximum power level of 3390 megawatts thermal (MWt). The licensee proposes to take advantage of a recent amendment to Nuclear Regulatory Commission's (NRC's) regulations that allows nuclear power plants to reduce the instrumentation uncertainty factor applied to reactor power level used in evaluations of emergency core cooling system (ECCS) performance. This change in the NRC's regulations allows SONGS Units 2 and 3 to apply a reduced uncertainty for ECCS evaluation from the 2 percent previously required. This action enables SONGS Units 2 and 3 to pursue an increase in the licensed reactor power level for each unit of approximately 1.42 percent, from the current 3390 MWt to 3438 MWt, while maintaining sufficient margins of safety for the facility.

In addition, the licensee requests continued use of topical reports identified in SONGS Technical Specification 5.7.1.5 Core Operating Limits Report (COLR). These topical reports describe the methodologies approved by the NRC that support the SONGS safety analysis, including the small break and large break loss of coolant accidents (LOCA) analyses. In many

of these topical reports, reference is made to the use of a 2-percent uncertainty applied to the reactor power, consistent with 10 CFR Part 50 Appendix K. The licensee proposes that these topical reports be approved for use consistent with this license amendment request. The licensee also seeks NRC's determination that the proposed change in the power uncertainty does not constitute a significant change as defined in 10 CFR 50.46 and 10 CFR Part 50 Appendix K.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specific core thermal power level. The power level is indicated in the control room by neutron flux instrumentation that is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system (NSSS). This calculation is called a "secondary calorimetric" for a pressurized water reactor. The accuracy of the calculation depends primarily upon the accuracy of feedwater flow and main steam and feedwater temperature and pressure measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. An accurate measurement of these parameters is necessary for an accurate determination of core thermal power and accurate calibration of the nuclear instrumentation.

The instrumentation used for measuring feedwater flow is typically an orifice plate, a venturi meter, or a flow nozzle. These devices generate a differential pressure proportional to the feedwater velocity in the pipe. Of the three differential pressure devices, a venturi meter is most widely used for feedwater measurement in nuclear power plants. The major advantage of a venturi meter is a relatively low head loss as the fluid passes through the device. The major disadvantage of the device is that the calibration of the flow element shifts when the flow element is fouled, causing the meter to indicate a higher differential pressure and hence a measured flow rate higher than the actual value. This leads the plant operator to calibrate nuclear instrumentation high, a conservative condition with respect to reactor safety. However, this causes the electrical output to be proportionally lower than when the plant is operated at its thermal power rating. To eliminate the fouling effects, the flow-measuring device has to be removed, cleaned, and re-calibrated.

In 1999, Westinghouse, formerly ABB Combustion Engineering Nuclear Power (ABB-CE), submitted the Crossflow Topical Report (Reference 7) for NRC review. The Topical Report documents the theory, design, and operating features of the Crossflow ultrasonic flow meter (UFM) and its ability to achieve increased accuracy of flow measurement. ABB-CE stated that this meter provides improvement in feedwater flow measurement accuracy as compared to the currently-installed venturi-type flow meters in most nuclear power plants. The reduction in flow measurement uncertainty allows a nuclear power plant licensee to more accurately determine reactor thermal power.

The NRC staff issued a safety evaluation (Reference 8), approving the Topical Report.

By its application (Reference 1), the licensee proposes to implement the approved Topical Report at SONGS Units 2 and 3. The licensee indicated that the Crossflow system at SONGS Units 2 and 3 would comply with the terms and conditions listed in the approved Topical Report and associated NRC Safety Evaluation Report.

3.0 EVALUATION

The NRC staff's review of the licensee's proposal to increase licensed reactor power focused on the following areas:

3.1 Nuclear Steam Supply and Balance of Plant Systems
3.2 Instrumentation and Control Systems
3.3 Electrical Systems
3.4 Accident Evaluations
3.5 Radiological Consequences
3.6 Operations
3.7 Technical Specification Changes

3.1 Nuclear Steam Supply and Balance of Plant Systems

The licensee evaluated the effect of plant operation at the proposed higher reactor power level on:

- NSSS components including the primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer);

- NSSS/Balance of Plant (BOP) interface systems;

- Reactor vessel integrity;

- Mechanical and structural loading conditions including seismic loads and the leak-before-break analysis conclusions;

- Primary and secondary over pressure relief capacities;

- Containment integrity;

- Auxiliary systems and components (including spent fuel pool cooling, radiological waste, and fire protection systems).

In Reference 9, the NRC issued amendments for SONGS Units 2 and 3 to reduce the minimum cold leg temperature from 544°F to 535°F (i.e., a reduction of 9°F). This change has bearing on operation under the proposed uprated power condition and is referred to as "the T_{cold} reduction amendment."

The NRC staff reviewed the licensee's evaluation and determined that operation at the proposed uprated reactor power level would not adversely affect safety systems, structures and components, as summarized in the following:

a. The licensee conducted an evaluation of the structural integrity of primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) under the proposed uprated conditions. In their evaluation, the licensee used the parameters

associated with the proposed uprated condition provided in Reference 4, such as system flow rates, pressures, and temperatures. The licensee concluded that, because the system parameters for the uprated condition are bounded by the system parameters for the design condition, the primary system components would continue to comply with their applicable structural limits under the proposed uprated power conditions and would continue to perform their intended design functions. The NRC staff accepts the licensee's conclusion that operation under the proposed uprated power conditions would not adversely affect the structural integrity of primary system components.

b. The licensee predicted that steam generator (SG) temperatures may increase under the proposed uprated power condition. The licensee stated that any changes in SG operating characteristics are evaluated consistent with the practices and procedures for managing SG tube integrity specified in the SONGS Steam Generator Program. Although the SONGS Technical Specifications allow operation at significantly higher reactor coolant system (RCS) temperatures than normally encountered during operation, the current plant procedures and practices restrict operating temperatures to help prevent tube degradation. Further, the temperatures predicted by the licensee for the uprated condition would be within current operating temperatures and the operating temperatures associated with the T_{cold} reduction amendment (Reference 9).

   The effects of the proposed uprate on SG tube integrity will be controlled by procedures and practices consistent with the SONGS Steam Generator Program that is required by SONGS Technical Specifications to monitor SG tube integrity. Operational assessments of SG tube integrity are required each cycle, and consider the effects of operating experience. These assessments are expected to indicate impacts on SG tube integrity that might result from operation under uprated conditions, and corrective or compensatory measures would then be taken to assure adequate SG tube integrity. The NRC staff accepts the licensee's analysis and conclusion that the existing SONGS Steam Generator Program can continue to be used to manage SG tube integrity under the proposed uprated power conditions.

c. The licensee addressed reactor pressure vessel integrity in its submittal. The existing fast neutron fluence values for SONGS Units 2 and 3 were based on fluence evaluations performed in conjunction with surveillance capsule measurement and testing. The values have been reviewed and approved for 20 Effective Full Power Years (EFPY). The current Technical Specification 3.4.3, RCS Pressure and Temperature Limits, are based on the projected fluence at 20 EFPY. Both units are presently at about 13.6 EFPY.

   RCS fluid density affects the neutron flux reaching the reactor vessel. Previously, the licensee gained approval to reduce the minimum cold leg temperature as described in the T_{cold} reduction amendment (Reference 9). Lowering the water temperature in the downcomer increases the water density and thus, the neutron thermalization rate. Although the licensee did not propose a specific value of the water density effect, the NRC staff calculations on a similar plant indicate an effect of about a 1/3 percent lower fluence per degree F decrease in downcomer temperature. Therefore, for a decrease of 9°F in downcomer fluid temperature, the NRC staff estimated the fluence reduction to be about 3 percent. Although the licensee did not refer to the bypass region, it is apparent that lowering the downcomer temperature will lower the temperature of the bypass region, reinforcing the neutron thermalization rate, and further lowering the fluence.

Assuming that the neutron leakage for future core loadings would increase in proportion to the proposed power uprate, the NRC staff estimates that the fluence decrease from the lower cold leg temperature would more than compensate for the neutron leakage increase due to the proposed power uprate. Therefore, the NRC staff finds that there is reasonable assurance of safety at uprated conditions and concurs with the licensee's statement that the existing fast neutron fluence data supporting operation to 20 EFPY remains bounding for the proposed uprated power conditions.

d. The licensee determined that leak-before-break analysis conclusions remain valid because the key RCS system design parameters (i.e., pressures and temperatures) fall within the range between the current plant operating condition (following the T_{cold} amendment) and the original design condition. Thus, the limiting break sizes determined in the analysis remain bounding for the proposed uprated power condition. Also, the licensee concluded that existing piping stress analyses and pipe support designs are not impacted by the implementation of the proposed power uprate. The NRC staff agrees with the licensee's conclusions because design pressures and temperatures used in the existing leak-before-break analysis bound the operating parameters associated with the uprated condition.

e. SONGS has an established flow accelerated corrosion (FAC) Monitoring Program by which it monitors and identifies FAC-susceptible BOP systems and components so that degradation can be arrested before a failure occurs. The licensee adjusts piping component inspections and replacements in accordance with its FAC Monitoring Program, based on wear data as it is collected in the FAC program. Flow rates under the proposed uprated power condition would fall within the range between the current plant operating condition (following the T_{cold} amendment) and the original design condition. The licensee determined that the flow rates for FAC-susceptible systems will not reduce the effectiveness of the FAC program. Based on its review of the licensee's assessment and the ability of the FAC monitoring program to identify degradation before failures occur, the NRC staff concurs with the licensee's conclusion.

f. The licensee performed an evaluation of the structural integrity for the RCS and components (i.e., reactor vessel and internals, RCS coolant piping and attached nozzles, the pressurizer, surge line, pressurizer spray nozzles, the SGs, the reactor coolant pumps, and reactor control components). The licensee provided key design parameters such as RCS pressure, hot leg temperature, cold leg temperature, SG steam pressure, SG outlet temperature, main feedwater temperature and main feedwater flow rate for the design basis analysis, under current operating conditions, and proposed uprated power conditions. The licensee found that the design parameters for the proposed uprated power condition (see Table 1 of Reference 4) are bounded by the design conditions. The licensee concluded that operation under the proposed uprated power conditions is bounded by the design basis analyses for the structural integrity of the RCS and components. The NRC staff agrees with the licensee's assessment.

g. The licensee also evaluated the effect of the proposed power uprate on the hydraulic forces and the flow induced vibration loading applied to RCS components. The proposed power uprate slightly increases T_{cold} and thus, decreases RCS fluid density. This will decrease the loading on the RCS components during normal operation and during a postulated LOCA. Relative to flow and pump induced vibration effects, the licensee indicated that the proposed power uprate condition is bounded by the design condition of

120 percent minimum flow at 500°F. The licensee concluded that the proposed power uprate will not increase the potential for flow induced vibration and that the original plant design-basis hydraulic forces applied to the RCS piping and components would not be changed as a result of the proposed power uprate. The NRC staff found that the RCS flow under the proposed uprated conditions would be up to about 112 percent of the minimum flow, including a 5-percent flow rate measurement uncertainty. Because the uprated flow condition is bounded by the design condition, the NRC staff agrees with the licensee's conclusion.

h. The licensee evaluated the structural integrity of SG components, including effects on hydraulic loading resulting from operation under the proposed uprated condition. The licensee concluded that the tube fluid-elastic stability ratio, stresses and fatigue usage factors calculated at critical locations would be within the allowable limits for the proposed power uprate condition. The NRC staff finds that the revised design parameters provided in Reference 4 are bounded by the existing design-basis condition and that there is sufficient margin regarding the tube fluid-elastic stability ratio to accommodate the slight changes due to the proposed power uprate. The NRC staff agrees with the licensee's conclusion.

i. The licensee evaluated the adequacy of BOP systems, comparing the existing design-basis parameters with the proposed power uprate conditions. The licensee concluded that the existing design-basis analyses for the pipe break locations previously identified and evaluated for potential impact on essential safety-related systems and components remain unchanged for the uprated condition. The licensee reviewed its motor-operated-valves (MOVs) program and indicated that the MOV evaluation at SONGS was based on differential pressure and flow conditions that bound the proposed uprated power condition. The licensee evaluated its commitments relating to; GL 95-07 (Reference 10) associated with pressure locking and thermal binding of safety-related power-operated gate valves; GL 96-06 (Reference 11) regarding the over-pressurization of isolated piping segments; and set point calculations for MOVs under the GL 89-10 program (Reference 12). The licensee found that the existing analysis conditions remain bounding for the proposed uprated power condition. Because the existing design-basis analyses bound the proposed uprated conditions, the NRC staff concurs with the licensee's conclusions that the proposed power uprate would have no adverse effects on safety-related valves and that conclusions regarding the SONGS MOV program remain valid.

j. The licensee determined that main steam safety valves (MSSVs) provide adequate relief capacity to maintain the SG pressures within design limits at uprated power conditions. The basis for the licensee's conclusion is that the operating condition under the proposed power uprate will be bounded by the design capacity for the MSSVs determined at 102 percent of currently-licensed reactor power. The licensee also found that the atmospheric dump valves (ADVs) and steam bypass control valves (SBCVs) meet design sizing requirements at the proposed uprated power level. The licensee stated that the design capacities for the ADVs and SBCVs are based on an assumed reactor power level of 102 percent of the currently-licensed reactor power, therefore, bounding the proposed uprated power condition. The NRC staff accepts the licensee's conclusions because the proposed power uprate condition would not exceed the design-basis operating condition for the valves.

k. Containment Integrity: LOCA and main steam line break (MSLB) are the design-basis events used to determine the limiting containment pressure and temperature response. In its submittal, the licensee indicated that the design basis LOCA and MSLB events in containment were re-analyzed for the T_{cold} reduction amendment. The mass and energy releases used in the re-analyses were based on a reactor power of 102 percent of the currently-licensed reactor power level. The NRC staff concludes that because the analyses assume a power level greater than the proposed uprated power condition, containment integrity is reasonably assured under the proposed uprated condition.

l. Safety-Related Cooling Systems: The licensee reviewed the capability of the shutdown cooling heat exchanger, the component cooling water system, and the salt water cooling system. The licensee determined that each would accommodate the increased heat loads associated with uprated conditions within its design basis. The licensee further determined that required condensate inventory is available and is unaffected by the proposed uprate because the current analysis of record assumes 102 percent of currently-licensed reactor power. The NRC staff accepts the licensee’s conclusions because the proposed uprated power condition would not exceed the assumed design-basis operating condition for the system.

m. Auxiliary Feedwater System (AFW): The licensee determined that the condensate storage tank capacity is sufficient to supply AFW requirements for transient or accident conditions. The calculations for determining AFW condensate storage requirements are based on an assumed initial reactor power level of 102 percent of currently-licensed reactor power. The licensee also determined that the minimum flow requirements for AFW, as dictated by the accident analyses are unaffected by the proposed increased reactor power. The NRC staff accepts the licensee’s conclusions because the proposed power uprate condition would not exceed the design-basis operating condition for the system.

n. Other BOP Systems: The licensee evaluated heating, ventilation, and air conditioning systems inside and outside of containment and determined that the systems would operate within their design capacity under uprated conditions. The licensee also determined that the existing fire protection program can accommodate uprated conditions because the Appendix R analysis for Units 2 and 3 is based on operation at 102 percent of currently-licensed reactor power. The licensee determined that existing analyses for the radioactive waste systems bound the uprated condition because the estimated releases assume a reactor power of 3600 MWt. The licensee evaluated the capacity of the spent fuel pool cooling system at uprated conditions and determined that the maximum temperature limits under normal and abnormal heat load conditions were not affected. The NRC staff accepts the licensee’s conclusions for these systems because the proposed power uprate condition would not exceed the design-basis operating condition for the systems.

Staff Conclusion for NSSS and BOP Systems

The NRC staff reviewed the licensee’s evaluations of the effects on NSSS and BOP systems of operation under the proposed uprated power condition. Based upon the licensee’s evaluation of component or system capability, comparison of relevant parameters used in existing design analyses with those for the proposed uprated power condition, and use of component and system monitoring programs at SONGS (e.g., the Steam Generator Program), the NRC staff

found that operation of these systems and components would be acceptable at the proposed higher power level of 3438 MWt.

3.2 Instrumentation and Control Systems

3.2.1 Crossflow UFM System

To reduce the maintenance costs associated with conventional flow measuring systems, nuclear power plant licensees have begun using ultrasonic flow measurement systems, the Crossflow UFM being one alternative. Using a Crossflow UFM essentially eliminates the measurement uncertainties due to venturi fouling and instrumentation drift and calibration shifts.

The Crossflow Topical Report (Reference 7) describes the Crossflow UFM system for the measurement of feedwater flow and provides a basis for the proposed increase in licensed reactor power. The topical report includes a Crossflow measurement uncertainty calculation based on a typical feedwater loop (straight pipe, fully developed flow) that established the Crossflow UFM system's ability to achieve an uncertainty of 0.5 percent or better with 95 percent confidence. The topical report provides specific guidelines for determining uncertainty values of the Crossflow input parameters [Velocity Profile Correction Factor (VPCF), inside diameter, transducer spacing, feedwater density, and Crossflow time delay]. The plant-specific uncertainties are determined when the meter is installed, using the guidelines provided in the topical report. The topical report stated that a trained vendor representative will install the hardware and software of the Crossflow UFM. Since the Crossflow measurement uncertainty is affected by temperature change, the topical report recommended improving the accuracy of feedwater temperature instrumentation.

The Crossflow UFM does not replace the currently-installed plant venturi, but provides an in-plant capability for periodically re-calibrating the feedwater venturi to adjust for the effect of fouling. A unique advantage of the Crossflow UFM system is that it is installed external to the pipe in which flow is to be measured, thereby not compromising pressure boundary integrity.

The operation of the Crossflow UFM is based on the fact that an ultrasonic beam traveling across fluid flowing in a pipe is affected (modulated) by the turbulence (eddies) present in the flowing liquid. When this modulated signal is processed, a random signal representing a signature of the flowing eddies can be obtained. In the Crossflow UFM, this operation is carried out by four ultrasonic transducers mounted on a metal support frame which is clamped on the feedwater piping. There is one upstream and one downstream transducer station, each station consisting of one transmitting and one receiving transducer. The Crossflow UFM calculates the interval required for a unique pattern of eddies to pass between the two ultrasonic transducer stations, and divides this known distance by the calculated time to obtain the flow velocity. This measured velocity is not an average velocity, but corresponds to the highest velocity at the center of the pipe, and must be multiplied by the VPCF to obtain the average flow velocity.

The Crossflow UFM system consists of a mounting/transducer support frame with ultrasonic transducers, a signal conditioning unit (SCU), and a data processing computer (DPC). The DPC receives a feedwater flow signal from the SCU and smoothed values of flow and temperature of main feedwater, main steam, and blowdown for each loop from the plant computer. Using the plant computer inputs and a built-in signal processing algorithm, the Crossflow DPC compares the ultrasonically-determined average flows and temperatures with

the existing process measurements and produces a correction factor. The DPC software validates each correction factor and, if the expected accuracy is achieved, generates a "good" flag; if the signals deviate from the expected accuracy, a "bad" flag is generated. The Crossflow-calculated mass flow is periodically compared to the venturi-measured mass flow to determine an adjustment of the venturi flow coefficient for obtaining the corrected mass flow signal. This corrected mass flow is used in calculating core thermal power and thereby calibrating nuclear instrumentation in accordance with the plant Technical Specification requirements.

3.2.2 Crossflow at SONGS

Presently at SONGS, feedwater flow is measured by an earlier version of the Crossflow system that lacks the features required for achieving the accuracy needed for the proposed power uprate. This instrumentation has been used only to periodically verify the accuracy of the feedwater flow venturi and to calibrate the main steam flow venturi, and does not meet the requirements of the Crossflow Topical Report. Additionally, the existing feedwater temperature and steam generator blowdown flow and temperature instrumentation uncertainties were such that installing Crossflow in the feedwater line would not by itself sufficiently improve the secondary calorimetric power uncertainty to support the proposed power uprate. Therefore, the licensee plans to install a new improved Crossflow UFM system and an Advanced Measurement Analysis Group (AMAG) high-accuracy ultrasonic temperature measurement (UTM) system on the main feedwater and the steam generator blowdown pipes. This new Crossflow UFM will replace the existing version of the Crossflow system and will use the same transducer mounting locations used for the existing UFM. These locations meet the Crossflow Topical Report requirements.

The licensee's submittal indicates that the Crossflow UFM and the AMAG UTM measurements of feedwater flow and temperature and steam generator blowdown flow and temperature provide a 4 to 5-fold decrease in the instrument uncertainty. This would reduce the total secondary calorimetric power uncertainty from 2 percent with flow venturis alone to less than 0.58 percent with the Crossflow UFM and UTM. Like the Crossflow UFM, the UTM will have a dedicated set of transducers and mounting frame and will use the ultrasonic signal transit time to determine the temperature of the process fluid. Flow and temperature data will be transmitted to the plant process computers used for calculating reactor power. These are the plant monitoring system (PMS), the core operating limit supervisory system (COLSS), and backup computer system (CBCS). The CBCS is run as a backup channel and is typically used only when PMS is not available. The PMS and CBCS receive correction factors and quality flags from the Crossflow DPC. If the quality flags are "good" and the plant is above a minimum power level, the COLSS programs in each computer multiply the flow and temperature signals by their associated correction factors. The COLSS uses these corrected values of flow and temperature to calculate reactor thermal power. If the quality flags become "bad," both computers alarm and the COLSS continues using the last good quality correction factors up to a predetermined time. If the quality flag cannot be restored to "good" during this interval, the correction factors will be changed to conservative default values.

The NRC staff's safety evaluation of the Crossflow Topical Report included four criteria to be addressed by a licensee requesting power uprate based on use of the UFM system. The licensee addressed each of the four criteria as follows:

1. The first criterion is related to maintenance and calibration procedures that will be implemented with the Crossflow UFM installation, including the process and contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.

   In its submittal, the licensee stated that calibration and maintenance of the Crossflow UFM and AMAG UTM will be performed using SONGS maintenance and calibration procedures that will be developed from vendor information and SONGS-specific experience, or will be performed by a combination of vendor procedures and SONGS procedures. The current software was verified and validated under CENP's Verification and Validation Program and a periodic online monitoring of the Crossflow system will verify that the SCU, DPC, and software remain within the stated accuracy. The licensee stated that the vendor or another qualified calibration facility will calibrate the timer and amplifiers in the UFMs and UTMs based on the vendor's recommendations.

   In response to the NRC staff's request for additional information (Reference 5), the licensee stated that the instrument calibration, software control, and hardware configuration will meet the same standards as the existing instrumentation and are subject to the requirements of 10 CFR 50.59. SCE described its programs for the calibration of UFMs, UTMs, and all other instrumentation whose measurement uncertainties affect the plant power calorimetric uncertainty. Reference 5 lists the plant instrument calibration procedures that are applicable to these instruments. The licensee stated that each instrumentation and control loop is calibrated following its applicable calibration procedure. If the loop output is not acceptable, individual components are calibrated according to their applicable calibration procedures. The licensee also listed and described its procedures for performing corrective actions, reporting deficiencies to manufacturers, and receiving and addressing manufacturer deficiency reports on these instruments. The NRC staff believes that the licensee's plant procedures can sufficiently assure instrumentation capability to provide acceptable power calorimetric uncertainty for the proposed power uprate.

   As described by the licensee, the UFM system is inoperable when "sufficiently valid for use" correction factors are not produced and the correction factor quality flag changes from "good" to "bad", alarming in the plant computer system. The Crossflow correction factors are anticipated to be updated every 4 minutes under the automatic update conditions. With an inoperable Crossflow system, plant operation will continue at the uprated power level for up to 31 days using the most recently generated "good " correction factors. If the Crossflow system remains inoperable in excess of the allowed outage time, reactor power will be reduced to the currently licensed rated thermal power of 3390 MWt. Continued operation of the COLSS programs with a "bad" quality flag does not affect safety since the COLSS programs will continue using the "good" correction factor. As discussed in Section 2.0 of this evaluation, continued operation will result in increased venturi fouling and a conservative indication of feedwater flow relative to the actual condition. The allowed outage time of 31 days is currently used and based on the worst case drift of the existing instruments affecting power calorimetric uncertainty.

2. The second criterion states that for plants that currently have the Crossflow UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the

installed UFM and confirm that the instrumentation is representative of the Crossflow UFM and is bounded by the requirements set forth in the Crossflow Topical Report.

The licensee stated in its submittal (Reference 1) that since 1997, an earlier version of the Crossflow UFM has been successfully used at SONGS to measure feedwater flow rate. This UFM was used to verify the feedwater flow signal and calibrate the steam flow signal used by the COLSS program for power calorimetric calculation. Considerable experience has been gained in setting up and tuning the equipment, as well as conducting measurements using the existing SONGS procedure. This experience should be directly applicable to the installation, calibration, tuning, and use of the upgraded Crossflow UFM.

3. The third criterion is that the licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater flow instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both the venturi and the Crossflow UFM for comparison.

   In Reference 1, SCE stated that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater flow instrumentation is based on accepted plant setpoint methodology set forth in Regulatory Guide 1.105 “Instrument Setpoints for Safety-Related Systems” and ISA S67.04, “Safety-Related Instrumentation Used in Nuclear Power Plants.” The licensee confirmed that an alternate methodology was not used and the Westinghouse calculation for the SONGS Units 2 and 3 site-specific installation found the Crossflow UFM uncertainty to be equal to or better than ±0.5 percent of rated feedwater flow and ±10 percent of rated blowdown flow. These uncertainties were statistically combined with other instrumentation uncertainties that affect the plant secondary calorimetric power uncertainty. The total secondary calorimetric power uncertainty with a margin for the AMAG UTM being out of service and future plant changes was found to be ±0.58 percent of rated thermal power, thereby justifying the proposed 1.42-percent power uprate. The NRC staff review found the licensee’s calculation to be based on an accepted plant setpoint methodology and, therefore, acceptable.

4. This criterion states that the licensee of a plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profile and meter factors not representative of the plant-specific installation) should submit additional justification. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for a previously installed and calibrated Crossflow UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the Crossflow UFM Topical Report.

   The licensee stated (Reference 1) that for SONGS there will be no site-specific configuration calibration because the installation is equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds number. The meter installations are located on long straight sections of piping and will be far enough from disturbances to conform to the installation requirements of the Crossflow Topical Report.

The NRC staff finds that SCE's response to these criteria sufficiently addresses the plant-specific concerns about Crossflow UFM maintenance and calibration, hydraulic configuration, and procedures and contingency plans for an inoperable Crossflow. The licensee used an approved methodology to calculate the plant-specific Crossflow measurement uncertainty and the plant power calorimetric measurement uncertainty.

Based on its review of the information provided by the licensee regarding the Crossflow UFM system measurement uncertainty and plant power calorimetric measurement uncertainty, the NRC staff finds that the SONGS Units 2 and 3 thermal power measurement uncertainty using the Crossflow UFM is limited to ±0.58 percent of reactor thermal power and can support the proposed increase in licensed reactor power.

3.3 Electrical Systems

The licensee reviewed the SONGS Units 2 and 3 electrical systems to identify the major items that may be affected by uprated power conditions and to evaluate the potential impact of an uprate on that equipment. The licensee has reviewed equipment ratings, sizing criteria, existing loading, and design margins for the electrical equipment powered by the onsite distribution system and has determined that they would remain within their respective ratings.

a. Generator and Onsite Distribution Systems: The generator nominal ratings for Units 2 and 3 are 1127 MW and 1180 MW at a 0.9 power factor. The generator manufacturer, Alsthom, confirmed that the generator is capable of operating at an active power output up to 1220 MW without any modification to the auxiliary equipment. As the anticipated net increase of approximately 16 MW lies within the maximum output rating of the generator, there will be no equipment limitations to prevent operation at a core power of 3438 MWt. The licensee's review of applicable calculations did not identify any changes to equipment protective relay settings for the generator. Therefore, the NRC finds that the proposed uprated power condition would not affect the electrical systems associated with the turbine auxiliary systems.

   The isophase bus main section is rated at 36,300 amps. The bus conductor is rated for a temperature rise of 55°C, and the bus enclosure is rated for a 30°C rise. These temperature ratings would permit a total load of 1383 MVA under uprated conditions. The isophase bus temperature ratings are well in excess of the anticipated generator output of 1311 MVA. The isophase bus would support the power increase with no modifications. The main transformer is rated for 1378 MVA and would support the power increase with no modifications. In addition, standard design practice at SONGS requires that switch yard equipment meet or exceed the rated capacity of the main generator. Thus, the licensee found that the onsite distribution system would not be affected by the power uprate. The NRC staff finds the licensee's conclusion acceptable.

b. Grid Stability: On an annual basis, the licensee performs a grid system analysis that is reviewed by the California Independent System Operator. In support of the proposed amendment, the licensee reviewed its grid system analysis using a bounding uprate of 50 MW and gross generator output of 1200 MW and determined that the proposed power uprate would have no impact on grid stability and reliability, and SONGS would continue to be in conformance with General Design Criterion 17 for the uprated power electrical conditions. The NRC staff finds the licensee's conclusion acceptable.

c. Station Blackout (SBO) Analysis: SBO is defined in 10 CFR 50.63 as the complete loss of preferred offsite and Class 1E onsite emergency power systems. The licensee has performed an evaluation of the SBO analysis to determine the impact of operation under uprated power conditions. A slight increase in decay heat generation would result (slightly higher cooling loads during cooldown). However, the proposed operating conditions are bounded by the existing plant analysis. The atmospheric dump valves are designed to provide a means of decay heat removal and plant cooldown during loss of condenser vacuum. Other elements of the SBO analysis would not significantly change and none of the associated instruments require control setpoint changes, or exceed design basis because of the proposed power uprate. Therefore, the NRC staff finds that the power uprate would not affect the SBO analysis.

d. Environmental Qualification: The licensee's evaluation of temperature effects and radiological consequences on electrical equipment qualification determined that the proposed power uprate temperature and radiological conditions would not affect the environmental qualification of electrical and mechanical equipment. The source term and core activity inventories of record bound the EQ loss-of-coolant-accident (LOCA) dose analyses and core activity inventories associated with a reactor power of 3458 MWt (102 percent of currently licensed power). Also, the range of temperatures permitted by the T_{cold} reduction amendment envelope the operating conditions associated with the proposed power uprate. Thus the proposed power uprate conditions would have no significant effect on the environmental conditions currently used for the safety-related equipment qualification program.

3.4 Accident Evaluation

The licensee determined that the SONGS Units 2 and 3 Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses remain valid because either the accident conditions are bounded by those considered in the initial licensing basis analyses or the transient conditions are unaffected by the proposed power level increase.

As part of the T_{cold} reduction amendment (Reference 9), the licensee re-analyzed the design-basis LOCA and main steam line break (MSLB) analyses. The mass and energy releases used in these re-analyses for the limiting events were based on a reactor power level of 3458 MWt (102 percent of currently licensed power) and therefore, remain valid for the proposed power level of 3438 MWt.

The pipe break evaluation (for mass and energy releases) was performed for a MSLB in the Main Steam Isolation Valves/Main Feed Water Isolation Valves enclosure area. Pipe break locations previously identified and evaluated for potential impact on essential safety-related systems and components remain unchanged.

The mass and energy releases determined for the limiting pipe break outside containment in existing analyses bound the mass and energy releases that would be predicted for operation under uprated conditions. Further, the pressure and temperature conditions in other connected systems outside containment would not be significantly affected by the power uprate, and mass and energy releases for previously evaluated pipe breaks in these other systems outside containment would not be impacted by the power uprate. Therefore, potential impacts from

pipe breaks in the Piping Penetration Area, the Safety Equipment Building, the Auxiliary/Radwaste Building, Auxiliary Feedwater Pumphouse, and various piping tunnels would remain unaffected by the power uprate.

The NRC staff found that UFSAR Chapter 15 accident analyses would remain valid because either the accident conditions are bounded by existing analyses or appropriate re-analyses have been performed.

The licensee's analysis relative to boric acid precipitation during long-term cooling following a LOCA is consistent with the NRC-approved CE topical report CENPD-254-NP-A (Reference 13). Recently, the NRC staff found that analyses of long-term cooling following LOCA at some reactors were conducted assuming a decay heat generation rate that may be inconsistent with a reading of 10 CFR Part 50 Appendix K. That is, the assumed decay heat generation rate is sometimes below the level specified in 10 CFR Part 50 Appendix K for ECCS performance evaluation. The NRC staff does not consider this difference to be safety-significant and it is assessing the regulatory requirements on a generic basis. The existing analyses of record are acceptable for the requested thermal power level because; (1) the licensee uses a previously approved methodology; (2) the total of the requested thermal power level plus instrument uncertainty remains unchanged; and (3) there are no other changes in the analyses of record.

3.5 Radiological Consequences

The licensee evaluated the effect of operation at the uprated power condition on the predicted radiological consequences of safety analyses of record. Moderate frequency and infrequent events were included in the licensee's evaluation. The licensee concluded that in all cases, moderate frequency events yield radiological consequences that are enveloped by other more severe events. The licensee found that the radiological consequences of some limiting faults were not enveloped by another more severe event.

The licensee evaluation included consideration of the following events: inadvertent opening of a steam generator atmospheric dump valve, increased main steam flow, steam line breaks, reactor coolant pump sheared shaft, control element assembly ejection, LOCA, fuel handling accidents, and spent fuel pool boiling. The licensee concluded that for most of these events, the non-LOCA source term modeled in the dose analysis equals or is more severe than any non-LOCA source term for core power less than 3458 MWt (e.g., 3438 MWt). In those cases, the radiological consequences are unchanged from the analyses of record for operation at the uprated power level and are acceptable to the NRC staff.

The analysis of record for some events did not bound operation at the proposed uprated power conditions. These events were: control element assembly ejection, LOCA, fuel handling accidents, and spent fuel pool boiling. These cases are specifically addressed below.

3.5.1 Control Element Assembly (CEA) Ejection

The licensee stated that the noble gas dose contributions from the source term for the CEA ejection analysis of record equals or is more severe than the non-LOCA source term for reactor power below 3458 MWt. However, the licensee stated that its evaluation of the thyroid dose contributions of iodine core isotopes showed that the source term used in the analysis of record

is approximately one percent less severe than the 3458 MWt non-LOCA source term. The licensee explained that the CEA ejection analysis of record that evaluates thyroid dose uses conservative dose conversion factors from Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" rather than the ICRP 30 dose conversion factors, which would reduce the thyroid dose by an estimated 30 percent. This significant level of conservatism offsets that small change in the source term predicted for operation at the uprated power condition.

The NRC staff concurs that the analysis using the Regulatory Guide 1.109 dose conversion factors is conservative. The NRC staff considers the licensee conclusion reasonable that the expected change of the dose consequences for the CEA ejection accident at the uprated power condition is within the level of conservatism of the analysis relative to use of the ICRP 30 dose conversion factors. The NRC staff accepts the analysis of record for operation at the uprated power condition based on: (1) the assumption that the change to the dose consequences because of operation at the proposed higher power level would be well within the conservatism of the analysis, (2) that the dose consequences of the existing analyses are well within regulatory limits and, (3) that the expected results of an analysis using the ICRP 30 dose conversion factors would be well within regulatory limits.

3.5.2 Loss-of-Coolant Accident (LOCA)

The licensee determined that the LOCA source term modeled in the analysis of record would not bound the 3458 MWt source term. The licensee conducted new LOCA dose analyses using the 3458 MWt source term that showed small increases in thyroid inhalation and whole body doses. The results remained well within the offsite dose criteria of 10 CFR Part 100 and 10 CFR Part 50 Appendix A General Design Criterion 19. The revised results are acceptable to the NRC staff because the licensee used approved analysis methods and previously-accepted analysis parameters (with the exception of power level), the changes in dose consequences are within the range expected for a small increase in licensed reactor power, and because the resulting dose consequences are well within regulatory limits.

3.5.3 Fuel Handling Accident

The dose analysis of record for the fuel handling accident does not bound the 3458 MWt source term. The licensee modeled a source term for the event that was increased by 2 percent to yield fuel rod activity inventories corresponding to 3458 MWt. The licensee concluded that the choices for peaking factors and iodine fuel rod gap release fractions contribute conservatism to the results that offsets the effect that would result from including the factor for power measurement uncertainty. The staff's experience with similar analyses using more realistic choices for peaking factors and iodine fuel rod gap release fractions is consistent with the licensee's conclusion. The NRC staff agrees with the licensee's assessment regarding the conservatism of the analysis results and accepts the analysis of record for operation at the uprated power condition based on: (1) the assumption that the change to the dose consequences because of operation at the proposed higher power level would be well within the conservatism of the analysis, (2) that the dose consequences of the existing analyses are well within regulatory limits and, (3) the staff's experience with similar analyses using more realistic choices for peaking factors and iodine fuel rod gap release fractions that yield results well within regulatory limits.

3.5.4 Spent Fuel Pool Boiling

As is the case with the fuel handling accident, the source term for the spent fuel pool boiling event does not bound the 3458 MWt source term. The licensee conducted a revised analysis for the spent fuel pool boiling event modeling the 3458 MWt source term and the related increased heat load on the spent fuel pool. The results indicated no increase in offsite thyroid, whole body, and skin dose effects. The revised results are acceptable to the NRC staff because the licensee used approved analysis methods and previously-accepted analysis parameters (with the exception of power level) and because the resulting dose consequences are well within regulatory limits.

3.6 Operations

The licensee described the effects of the proposed power uprate on: (1) control room alarms, displays, and controls, (2) emergency and abnormal operating procedures, (3) the safety parameter display system, and (4) the operator training program and the control room simulator. The licensee provided information stating that there are no changes in plant emergency and abnormal operating procedures and no changes to the time available for operator response because of the power uprate. Therefore, there are no changes to risk-important operator actions sensitive to power uprate.

Control room alarms would be affected by installation of the Crossflow system. The licensee stated that operating procedures would be changed to accommodate the Crossflow alarms and actions necessary to address problems with the Crossflow system. Operator displays, instrumentation, and control features in the control room are unchanged as a result of installation of the Crossflow system. Displays for reactor power would show 100 percent power for the proposed uprated power level of 3438 MWt. Some plant operating parameters will have minor changes as a result of operation at the uprated condition. Those parameters found to be outside of the existing zone markings on control board indicators would be addressed under the licensee’s process for determining appropriate indicator operating band zones.

The licensee stated that operation under uprated power conditions will have a negligible effect on the parameter displays of the safety parameter display system. The licensee also stated that the power uprate would not change the type and scope of plant emergency and abnormal operating procedures and the existing procedures are not changed. The licensee would provide classroom and simulator training on all changes resulting from the power uprate that affect operator performance. The licensee would modify operator training and the plant simulator as required to address discrepancies identified during the startup test program.

The NRC staff finds that the licensee has adequately identified changes resulting from the power uprate on control room alarms, displays, and controls, emergency and abnormal operating procedures, the safety parameter display system, and the operator training program and the control room simulator. In those instances where changes were identified, the licensee has indicated that appropriate actions would be taken to ensure that operator performance would not be adversely affected by operation at the uprated power level. Specifically, the licensee has proposed measures acceptable to the NRC staff to incorporate changes associated with the proposed power uprate to control room displays, alarms, and operator training and control room simulation.

3.7 Technical Specification Changes

The licensee requests changes to the Facility Operating Licenses and the Technical Specifications for SONGS Units 2 and 3 to reflect the proposed new power level. The changes only affect the maximum power level specified in the Facility Operating Licenses and the definition of rated thermal power in Section 1.1 of the Technical Specifications for the units. These changes are consistent with the proposed changes to the plant and the results of the associated safety analysis. Therefore, the NRC staff finds the proposed changes acceptable. The NRC staff noted that suggested changes to the Technical Specifications Bases were consistent with the licensee's proposed increase in licensed power level.

The licensee requests continued use of topical reports identified in SONGS Technical Specification 5.7.1.5 Core Operating Limits Report (COLR). These topical reports describe the NRC-approved methodologies that support the SONGS safety analysis, including the small break and large break LOCA analyses. In many of these topical reports, reference is made to the use of a 2-percent uncertainty applied to measured reactor power, consistent with 10 CFR Part 50 Appendix K. The licensee proposes that these topical reports be approved for use consistent with this license amendment.

Before its revision in 2000, 10 CFR Part 50 Appendix K required that a 2-percent instrumentation uncertainty factor be applied to the licensed power level to evaluate ECCS performance. Appendix K now states that ECCS evaluations use an assumed power level demonstrated to account for uncertainties due to power level instrumentation error. The licensee has demonstrated that instrumentation uncertainty is less than 0.58 percent. The original intent of incorporating a 2-percent uncertainty to reactor power in the topical reports will be met by using the 0.58-percent value. Consequently, the NRC staff approves continued use of topical reports identified in SONGS Technical Specification 5.7.1.5 COLR consistent with this license amendment when reference is made to assuming a 2-percent uncertainty for reactor power.

SONGS also requested that the NRC acknowledge that the change in the power uncertainty does not constitute a significant change as defined in 10 CFR 50.46 and 10 CFR 50 Appendix K. For purposes of marginal changes in reactor power level, the NRC staff considers the reactor power assumed in the ECCS evaluation model to be the licensed thermal power plus the demonstrated instrumentation uncertainty. Therefore, in cases where licensed thermal power is increased on the basis of reduced instrumentation uncertainty, there is no evaluation model change (and no peak fuel cladding temperature change) and the total power used in the evaluation model remains unchanged.

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

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on June 19, 2001 (66 FR 32964 and corrected in 66 FR 66 FR 33982) for these amendments. Accordingly, based upon the

environmental assessment, the Commission has determined that issuance of these amendments will not have a significant effect on the quality of the human environment.

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.

7.0 REFERENCES

1. Letter from D. E. Nunn (SCE) to NRC, "Proposed Change Number NPF-10/15-514 - Increase in Reactor Power to 3438 MWt - San Onofre Nuclear Generating Station Units 2 and 3," April 3, 2001.

2. Letter from D. E. Nunn (SCE) to NRC, Response to NRC Request for Additional Information, April 23, 2001.

3. Letter from D. E. Nunn (SCE) to NRC, Response to NRC Requests for Additional Information, May 11, 2001.

4. Letter from D. E. Nunn (SCE) to NRC, Response to NRC Requests for Additional Information, May 25, 2001.

5. Letter from D. E. Nunn (SCE) to NRC, Response to NRC Request for Additional Information, May 31, 2001.

6. Letter from D. E. Nunn (SCE) to NRC, Response to NRC Request for Additional Information, June 25, 2001.

7. Topical Report, CENPD-397-P-A, Revision 1, "Improved Flow Measurement Accuracy using Crossflow Ultrasonic Flow Measurement Technology," May 2000.

8. Letter from Stuart A. Richards (NRC) to Ian C. Rickard (PSEG), "Acceptance for Referencing of CENPD-397-P, Revision-01-P, Improved Flow Measurement Accuracy using Crossflow Ultrasonic Flow Measurement Technology," March 20, 2000.

9. Letter from J.W. Clifford, NRC to H.B. Ray, Southern California Edison, "Issuance of Amendment for San Onofre Nuclear Generating Station," February 12, 1999.

10. NRC Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," August 17, 1995.

11. NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," September 30, 1996.

12. NRC Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” June 28, 1989.

13. CE Topical Report, CENPD-254-P-A, “Post LOCA Long Term Cooling Evaluation Model,” June 1980.

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Date: July 6, 2001

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN CALIFORNIA EDISON COMPANY

DOCKET NOS. 50-361 AND 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos.   and   to Facility Operating Licenses Nos. NPF-10 and NPF-15, Southern California Edison Company (SCE or the licensee), which revised the Operating License and Technical Specifications for operation of the San Onofre Nuclear Generating Station (SONGS), Units Nos. 2 and 3, located in San Diego County, California. The amendments are effective as of the date of issuance.

The amendments modified the Technical Specifications and Operating License for SONGS Units 2 and 3, to allow SCE to increase the maximum reactor core power level for each unit from 3390 megawatts thermal (MWt) to 3438 MWt, which is an increase of 1.42 percent of rated core thermal power for SONGS Units 2 and 3.

The proposed action is in accordance with the licensee's application for amendment dated April 3, 2001, and supplemented by letters dated April 23, May 11, May 25, May 31, and June 25, 2001.

The application for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on April 18, 2001 (66 FR 19996). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendments will not have a significant effect on the quality of the human environment (66 FR 32964, and corrected in 66 FR 33982).

For further details with respect to the action see (1) the application for amendment dated April 3, 2001, (and supplemented by letters dated April 23, May 11, May 25, May 31, and June 25, 2001), (2) Amendments No. 180 to License No. NPF-10, and No. 171 to License No. NPF-15, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/NRC/ADAMS/index.html. If you do not have access to ADAMS or if there

are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 6th day of July 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Joseph E. Donoghue, Senior Project Manager
Project Directorate IV, Section 2
Division of Licensing Project Management
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