SOUTHERN CALIFORNIA
EDISON
An EDISON INTERNATIONAL℠ Company

Dwight E. Nunn
Vice President

May 11, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: **Docket Nos. 50-361 and 50-362**
Proposed Change Number NPF-10/15-514
Increase in Reactor Power to 3438 MWt
San Onofre Nuclear Generating Station
Units 2 and 3

References:
1. SCE to NRC letter dated April 3, 2001, Subject: Proposed Change Number NPF-10/15-514 Increase in Reactor Power to 3438 MWt, San Onofre Nuclear Generating Station Units 2 and 3
2. NRC to SCE letter dated April 18, 2001, Subject: Request for additional Information Re: License Amendment Request to Increase Reactor Power from 3390 MWt to 3438 MWt San Onofre Nuclear Generating Station Units 2 and 3

Gentlemen:

This letter provides responses to NRC requests for additional information (RAIs) concerning the Southern California Edison (SCE) request to increase the reactor power to 3438 MWt at San Onofre Units 2 and 3, Amendment Applications 207 and 192, Proposed Change Number 514 (Reference 1).

Enclosure 1 provides the information requested in an April 11, 2001 telephone call. Enclosure 2 provides the information requested in an April 16, 2001 telephone call and April 18, 2001 letter (Reference 2). Enclosure 3 provides the information requested in an April 24, 2001 telephone call. Enclosure 4 provides revised pages as applicable for Proposed Change Number 514 description with text additions shaded and deletions struck out.

A001

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If you have any questions regarding these amendment applications, please contact me or Mr. Jack L. Rainsberry (949) 368-7420.

Sincerely,

Enclosures

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
J. G. Kramer, NRC Acting Senior Resident Inspector, San Onofre Units 2 & 3
L. Raghavan, NRC Project Manager, San Onofre Units 2 and 3
S. Y. Hsu, Department of Health Services, Radiologic Health Branch

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 2 of the San Onofre Nuclear Generating Station	)	Docket No. 50-361 Amendment Application No. 207

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit information in support of Amendment Application No. 207. This information consists of responses to NRC requests for additional information on Proposed Change No. NPF-10-514 to Facility Operating License NPF-10. Proposed Change No. NPF-10-514 is a request to revise the Facility Operating License by increasing the licensed power for operation.

Subscribed on this 11th day of May, 2001.

Respectfully submitted,
SOUTHERN CALIFORNIA EDISON COMPANY

By: [signature]
Dwight E. Nunn
Vice President

State of California
County of San Diego
On May 11, 2001 before me, Frances M Thurber personally appeared Dwight E. Nunn, personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.
WITNESS my hand and official seal.

Signature Frances M. Thurber

FRANCES M. THURBER
Commission # 1295266
Notary Public - California
San Diego County
My Comm. Expires Mar 23, 2005

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 3 of the San Onofre Nuclear Generating Station	)	Docket No. 50-362 Amendment Application No. 192

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit information in support of Amendment Application No. 192. This information consists of responses to NRC requests for additional information on Proposed Change No. NPF-15-514 to Facility Operating License NPF-15. Proposed Change No. NPF-15-514 is a request to revise the Facility Operating License by increasing the licensed power for operation.

Subscribed on this 11th day of May, 2001.

Respectfully submitted,
SOUTHERN CALIFORNIA EDISON COMPANY

By: ______________________
Dwight E. Nunn
Vice President

State of California
County of San Diego
On May 11, 2001 before me, Frances M. Thurber, personally appeared Dwight E. Nunn personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.
WITNESS my hand and official seal.

Signature ______________________

FRANCES M. THURBER
Commission # 1295266
Notary Public - California
San Diego County
My Comm. Expires Mar 23, 2005

Enclosure 1

Response to the April 11 NRC/Southern California Edison (SCE) Telephone Call RAI

Item 1. Document that safety related cooling systems, Heating Ventilation and Air Conditioning (HVAC) systems, fire protection systems, radioactive waste systems, and Spent Fuel Pool (SFP) systems were evaluated.

Response:

Safety Related Cooling Systems

The adequacy of the safety related cooling systems were reviewed with the following results:

1. Shutdown Cooling Heat Exchanger (cooled by the Component Cooling Water System)

   The post uprate shutdown heat load is within the design basis heat load of the heat exchanger for normal shutdown conditions. The post-accident heat loads and temperatures remain unchanged because the design heat load is 102% of core power, consistent with the Safety Analysis, and remains unchanged.

2. Component Cooling Water (CCW) System

   The CCW system was reviewed for the post uprate conditions. The small increase in heat load due to a slightly higher reactor coolant system (RCS) temperature is bounded by the temperature ranges assumed in the original design and the Tcold reduction, Amendments 179 and 165 (Reference).

Reference: Letter from James W. Clifford (NRC) to Harold B. Ray (SCE), Issue of Amendment for San Onofre Nuclear Generating Station, Unit No. 2 (TAC No. MA2238) and Unit No. 3 (TAC No. MA2239), February 12, 1999

3. Salt Water Cooling (SWC) System

   The SWC system removes heat from the Component Cooling Water System and transfers the heat to the Pacific Ocean (the ultimate heat sink). The small increase in heat load due to the small increase in CCW heat load is within the design heat rejection capability of the heat exchanger and the Pacific Ocean.

HVAC

The HVAC systems inside and outside containment were evaluated. The heat load incontainment remains well below the heat load prior to Tcold reduction (when RCS temperatures were higher), therefore the design capacity of the containment HVAC is unaffected by the uprate. Since there will be no increase in emergency core cooling system piping temperatures outsidecontainment (because the shutdown cooling entry conditions remain unchanged and there are no changes to motor brake horsepower ratings), the design heat load on the HVAC systems outside containment are unaffected.

Fire Protection Program

The proposed uprate has no effect on the Fire Protection Program since the San Onofre Nuclear Generating Station (SONGS) 2 and 3 Appendix R analysis is based on 102% reactor power.

Radioactive Waste Systems

In the SONGS 2/3 NRC Safety Evaluation Report (SER) (NUREG-0712, February 1981) Section 11.0, Radioactive Waste System, the estimated releases of radioactive materials in the liquid and gaseous effluents were calculated based on a reactor power of 3600 MWt. The calculated release results were used as the basis for determining: 1) the capability of the waste systems for keeping levels of radioactivity in effluents "as low as reasonably achievable," 2) the capability of the systems to maintain releases below limits of 10CFR20, 3) the design features incorporated to control the releases of radioactive materials in accordance with Criterion 60 of the General Design Criteria, and 4) the capability of the systems to meet station processing demands. The proposed uprate to 3438 MWt will still remain well within the bounds of the 3600 MWt value used to calculate radioactive release results in the original design of the SONGS 2 and 3 radioactive waste systems.

Spent Fuel Pool

The capacity of the Spent Fuel Pool Cooling system has been evaluated for the increased decay heat load as a result of the power uprate amendment requests in Proposed Change Number 514. Although the evaluation resulted in a slightly higher spent fuel pool temperature, the Updated Final Safety Analysis Report maximum normal heat load pool temperature limit of 140°F and the maximum abnormal heat load pool temperature limit of 160°F are unaffected. The dose consequences were reevaluated as described in Section 4.2.2.7 of Proposed Change Number 514.

Item 2. Provide discussion of the environmental evaluation.

Response:

The effect that the proposed change would have on the environment and the general public was evaluated. With regard to non-radiological discharges, the current National Pollutant Discharge Elimination System (NPDES) permits were issued August 11, 1999 and will not expire until August 11, 2004. SCE expects subsequent NPDES permits will be issued every five (5) years upon expiration with the next renewal in 2004. There will be no significant non-radiological impact on the environment with regard to liquid discharges from San Onofre Units 2 and 3 as a result of changing power rating since SCE will abide by the NPDES permits. Continued operation of San Onofre Units 2 and 3 will avert non-radiological environmental effects of airborne effluents from non-nuclear plants that would be required to operate in order to replace the power supplied by San Onofre Units 2 and 3.

Enclosure 2

Response to the April 18, 2001 NRC to SCE letter
and April 16 NRC/SCE telephone call RAI

The NRC staff requests a description of the programs and procedures that will control calibration of the Crossflow system and the pressure and temperature instrumentation whose measurement uncertainties affect the power calorimetric uncertainties determined in the Westinghouse calculation, as referenced in Section 2.2.3 of your April 3, 2001 submittal. In this description, please include the procedures for:

1. **Maintaining calibration,**
2. **Controlling software and hardware configuration,**
3. **Performing corrective actions,**
4. **Reporting deficiencies to the manufacturer, and**
5. **Receiving and addressing manufacturer deficiency reports.**

Response:

Reactor Power is determined by a calorimetric heat balance on the secondary plant. A calorimetric heat balance primarily uses steam flow, feed flow, and steam pressure to determine the power transferred through the steam generators to the secondary plant.

The CROSSFLOW System is being purchased and installed to meet the requirements of the CROSSFLOW Topical Report (Reference). These requirements and standards meet or exceed the existing instrumentation used to perform the calorimetrics heat balance. The instrument calibration, software control, and hardware configuration will be performed to the same standards as the existing instrumentation and subject to the requirements of 10CFR50.59. In addition, specific technical and contractual requirements have been imposed on the supplier, such as deviation from the specification, material substitution, and design document submittal.

The SONGS Action Request System will be used to control corrective actions activities (see item 3 below). This system provides a method to request engineering assistance and corrective action maintenance orders. Deficiencies will be reported to the manufacturer if an engineering evaluation, through an Action Request, determines that vendor notification is appropriate. SCE, as part of its business practices, evaluates manufacturer deficiency reports. Appropriate corrective actions are taken and, if necessary, an Action Request is generated for equipment installed in the plant.

Reference: Westinghouse/ABB-CE Topical Report CENPD-397-P-A, Revision 1, Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology, dated May 2000.

1. Maintaining calibration

SO23-V-2.10, Feedwater Ultrasonic Flow Measurement and Main Steam Flow Calibration: The temporary CROSSFLOW ultrasonic flow measurement system, presently used at SONGS, is installed and operated by this procedure. New operating instructions are being developed as part of the design change that will direct the performance of the internal calibration check at intervals specified by the vendor. The calibration procedures for the permanently installed CROSSFLOW ultrasonic flow and temperature systems are under development as part of the design change and will be based on the vendor's manuals and experience at SONGS. The vendor has calibrated the timers and amplifiers for the existing temporary units although the units could also be calibrated by another qualified calibration facility. The vendor or another qualified calibration facility will calibrate the timers and amplifiers in the permanent installation based on the vendor's recommendations.

The temporary and the new permanent CROSSFLOW systems have an internal calibration check to assure proper operation.

2. Controlling software and hardware configuration

SO123-XXIV-10.1, Preparation, Review, Approval, Issuance, Implementation, and Closure of Engineering Change Packages (ECPs) and Engineering Changes Notices (ECNs): This procedure will be used for all design changes and provides a detailed description of the process and controls for design activities. It identifies the sequential stages of the design process from the initial assignment of a design task through final drawing revision.

SO123-XXIV-5.1, Engineering & Technical Services Software Quality Assurance: This procedure establishes the program for acquiring, developing, qualifying, maintaining, and controlling Engineering & Technical Services (E&TS) computer software used for Quality Affecting activities. The procedure was written in accordance with licensing commitments which require that newly acquired or revised Quality Affecting Software be properly designed, configured, verified, and documented prior to use. This procedure shall be followed for software on new plant computer systems in support of design change activities.

SO123-V-4.71 Software Development and Maintenance: This procedure establishes a program for acquiring, developing, maintaining, and controlling computer software and associated responsibilities used to support Quality Affecting activities. (Note: the CROSSFLOW software is being purchased as "quality affecting" software.) This procedure shall be followed for modifications to software on existing plant computer systems.

3. Performing corrective actions

SO123-CA-1, Corrective Action Program: This Order describes the Corrective Action Program used by the Nuclear Organization to identify, evaluate and resolve conditions adverse to quality. The overall program structure, responsibilities, and requirements are specified to implement regulatory and nuclear organization management requirements.

SO123-XV-50, Corrective Action Process: This procedure defines the Corrective Action Process for identifying, evaluating and resolving conditions adverse to quality, including problems contrary to nuclear safety, public safety, and regulatory compliance. This procedure also defines requirements and guidance for root cause evaluations, corrective action assignments, apparent cause evaluations, common cause evaluations, corrective action follow-up activities, and trending activities.

SO123-XX-1 ISS 2, Action Request/Maintenance Order Initiation and Processing: This procedure provides a single system for reporting of conditions adverse to quality, events, proposed improvements (equipment and non-equipment related) and for resultant actions. It delineates responsibilities for the management and oversight of the Action Request process and defines the process for ensuring timely corrective actions are taken commensurate with the safety significance of the reported condition.

SO123-XX-50.39, Cause Evaluation Standards and Methods: This procedure defines the standards and methods for conducting root cause, apparent cause, and common cause evaluations.

4. Reporting deficiencies to the manufacturer

SO123-XX-1 ISS 2, Action Request/Maintenance Order Initiation and Processing: This procedure provides a single system for reporting of conditions adverse to quality, events, proposed improvements (equipment and non-equipment related), and for resultant actions. It delineates responsibilities for the management and oversight of the Action Request process and defines the process for ensuring timely corrective actions are taken commensurate with the safety significance of the reported condition. Action Requests provide a mechanism to direct reporting deficiencies to the manufacturer, including any potential 10 CFR 21 issues.

5. Receiving and addressing manufacturer deficiency reports

SO123-XX-1 ISS 2, Action Request/Maintenance Order Initiation and Processing: This procedure provides a single system for reporting of conditions adverse to quality, events, proposed improvements (equipment and non-equipment related), and for resultant actions. It delineates responsibilities for the management and oversight of the Action Request process and defines the process for ensuring timely corrective actions are taken commensurate with the safety significance of the reported condition. Action Requests provide a mechanism to address actions resulting from receipt of manufacturer deficiency reports.

SO123-V-4.71 Software Development and Maintenance: This procedure establishes a program for acquiring, developing, maintaining, and controlling computer software and associated responsibilities used to support Quality Affecting activities. (Note: the CROSSFLOW software is being purchased as "quality affecting" software.) This procedure shall be followed for modifications to software on existing plant computer systems. This procedure is applicable to software modifications which result from a manufacturer's deficiency report.

Enclosure 3

Response to the April 24 NRC/SCE telephone call RAI

The following requests are based on an NRC telephone call on April 24, 2001.

Item 1. Provide a description of the process for controlling power level when the Crossflow is out of service and how corresponding plant instrument drift values are determined. Describe how "good"and "bad" flags are assigned to correction factors.

Response:

The response to this request will be provided by May 21, 2001.

Item 2. Provide a list of the contributors to the uncertainty factors in the calorimetric calculations for pre- and post-change conditions. Explain how the old and new uncertainties fit together to understand what is calculated and how it is calculated.

Response:

The secondary calorimetric power measurement uncertainty is comprised of two separate calculation steps. The first step is to determine the input instrument uncertainty terms and the second combines the input instrument uncertainty terms into the secondary calorimetric power measurement uncertainty. The NRC approved methodology and the input uncertainty terms used in the calculation of the secondary calorimetric power measurement uncertainty are described in Reference 1.

For the existing plant with flow venturis, each input instrument uncertainty term was calculated with a +/- 2 standard deviation accuracy and confidence level. These calculations account for the uncertainty associated with loop components from the primary sensing element to the relevant output device. Drift studies were performed for the relevant primary sensing elements. Uncertainties were combined using the square root of sum of squares (SRSS) method for uncertainties that are independent, random, and normally distributed. Uncertainties that do not meet this criteria were combined arithmetically. This method of combining uncertainties is consistent with the methodology contained in NUREG/CR-3659 (Reference 2). The uncertainty components are listed in the table below with the values used for the existing plant and the plant with a power uprate to 3438 MWt.

The aforementioned input uncertainty terms are then combined using stochastic simulation and the secondary calorimetric power equations with the result being the total secondary calorimetric power measurement uncertainty (Reference 1). In stochastic simulations larger uncertainty terms dominate the smaller uncertainty terms, which result in the small uncertainty terms having an insignificant effect on the total secondary calorimetric power measurement uncertainty, consistent with Appendix C in Reference 2. Therefore, the secondary calorimetric power measurement uncertainty is calculated using the dominant uncertainty terms.

After Power Uprate with Crossflow ultrasonic flow measurement the affected input instrument uncertainty terms will be calculated as described in Reference 3. Input instrument uncertainty terms, unaffected by the CROSSFLOW System, will continue to be calculated as described above.

NRC approved methodology (Reference 1) continues to be applicable with a power uprate to 3438 MWt because all dominant terms continue to be modeled in a conservative manner and the non-dominant uncertainty terms do not significantly affect the total secondary calorimetric power measurement uncertainty. To demonstrate the conservative model, the impact of the largest non-dominant uncertainty term which is neglected in the current methodology, Reactor Coolant Pump (RCP) heat, is discussed below. The RCP heat was measured in Cycle 1 startup testing as reported in References 4 and 5. Assuming an artificially conservative 10 % uncertainty (2σ) of the net RCP heat input (rounded up to 20 MWt), the total secondary calorimetric power measurement uncertainty changes by less than 0.01 % for the plant with power uprate as follows:

The uncertainty is combined using the SRSS method since the RCP heat uncertainty is independent, random, and normally distributed.

$$U_{power} = [\ (U_{powerSG})^2 + (U_{RCPHeat})^2\ ]^{1/2}$$

where

U_{power}	$= 2\sigma$ uncertainty in secondary calorimetric power, % power
$U_{powerSG}$	$= 2\sigma$ uncertainty in energy removed by steam generators, % power
$U_{RCPHeat}$	$= 2\sigma$ uncertainty in RCP heat input to RCS, % power
2σ	= Twice the standard deviation representing the 95% tolerance interval

Substituting the value above for uncertainty in RCP heat and the value of the Total Secondary Calorimetric Power base uncertainty from the table below of 0.44% using Crossflow instrumentation for a power uprate to 3438 MWt and using the NRC approved methodology of Reference 1 yields

$$U_{power} = [\ (0.0044)^2 + \ ((0.1 * 20)\ /3438)^2\ ]^{1/2} = 0.00444 = 0.444\ \%$$

Therefore, the total secondary calorimetric power measurement uncertainty does not significantly change from the base uncertainty (0.44%). In addition, the RCP heat uncertainty is less than one-fifth the value of the larger uncertainty which is the criteria for eliminating small uncertainty components established in Reference 2 Appendix C, and therefore can be neglected in the calculation of the total secondary calorimetric power measurement uncertainty.

Hence, the approved methodology continues to be valid and applicable to the plant with a power uprate to 3438 MWt.

References:

1) Topical Report CEN-356(V)-P-A, Revision 01-P-A, "Modified Statistical Combination of Uncertainties," May 1988.

2) NUREG/CR-3659, "A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors, Appendix C, February, 1985.

3) Topical Report CEN-397(V)-P-A, Revision 01-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurements," January 2000.

4) Startup Report to the USNRC, License Number NPF -10, Docket Number 50-361, August 8, 1983.

5) Startup Report to the USNRC, License Number NPF -15, Docket Number 50-362, May 29, 1984.

San Onofre Units 2 and 3 Secondary Calorimetric Power Uncertainty Components For Existing Plant and Power Uprate

	Existing Plant with Flow Venturis		Power Uprate with Crossflow UFM	
Uncertainty Term	Value	Distribution	Value	Distribution
Feedwater Flow (6)	±13.0 in H_2O (3.4 % span)	2σ normal	±0.5 % indicated flow (klbm/hr)	2σ normal
Feedwater Temperature (6)	±8.5 °F	2σ normal	±1.8 °F	2σ normal
Steam Flow (6)	±10.0 in H_2O (2.6 % span)	2σ normal	±0.53 % indicated flow (klbm/hr)	2σ normal
Blowdown Flow	±100 gpm (4) (50 % flow)	Uniform	±10.0 % (6) indicated flow (klbm/hr)	2σ normal
Calibration / Repeatability	Included in flow uncertainty	--	±0.2 %	2σ normal
Steam Generator Pressure (6)	±16.0 psi	2σ normal	±16.0 psi (1)	2σ normal
Steam Header Pressure (6)	±23.0 psi	2σ normal	±23.0 psi (1)	2σ normal
Steam Quality (5)	0.002	Uniform	0.002 (1)	Uniform
Total Secondary Calorimetric Power Uncertainty	±1.56 % power	95%/95%(3)	±0.44% power	95%/95%(3)
Uncertainty Margin Allowance for AMAG out-of-service (2) and Future Plant Changes	±0.44 % power	–	±0.14% power	--
Total Secondary Calorimetric Power Uncertainty Limit	±2.00 % power	95%/95%(3)	±0.58% power	95%/95%(3)

Notes:

(1) Unaffected by Crossflow instrumentation or power uprate
(2) For AMAG out-of-service see this RAI response, Enclosure 3, Item 3
(3) 95% probability at 95% confidence level
(4) 50% of flow indication assumed as conservative uncertainty
(5) Entire operating range assumed as conservative uncertainty
(6) Input instrument uncertainty term

Item 3. Provide a summary statement for the overall acceptability of the change. This could state that the change largely is bounded by the existing condition and that previously accepted for the pre-Tcold reduction.

Response:

The impact on systems, structures, and components of uprating to 3438 MWt (an approximate 1.42% increase) based on increased instrument accuracy in determining thermal power level was evaluated. The 1.42% proposed uprate may result in a slightly higher Reactor Coolant System temperature. Since the Technical Specifications were revised to reduce the minimum reactor coolant system (RCS) cold leg temperature (Tcold) at or above 70% power (Reference), any increase in RCS temperature as a result of the uprate is bounded by the original plant design and the analysis supporting the reduction in cold leg temperature.

Bounding evaluations were performed on primary plant and secondary plant equipment. Since the proposed post-uprate RCS parameters fall between the current operating conditions (post-Tcold reduction) and the original design, the steam generators continue to meet the licensing basis and will continue to be operated in accordance with the Technical Specifications and managed by the SONGS Steam Generator Program.

The reduction in the uncertainty allowance provided for the power calorimetric measurement allows current safety analyses to be used, without change, to support operation at a core power of 3438 megawatts thermal (MWt). As such, all Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 1.42% uprated condition for all events except the spent fuel pool boiling and the large break LOCA. These analyses have been reperformed and found to meet all acceptance criteria.

Reference: Letter from James W. Clifford (NRC) to Harold B. Ray (SCE), Issue of Amendment for San Onofre Nuclear Generating Station, Unit No. 2 (TAC No. MA2238) and Unit No. 3 (TAC No. MA2239), February 12, 1999.

Item 4. Clarify the wording in Section 3.3.1.4.

Response:

Revise the wording in Proposed Change Number 514 description, Section 3.3.1.4, from "operating conditions" to "design operating conditions."

Item 5. Determine whether there is a contradiction between the statements that core uplift force will be reduced while core differential pressure will increase.

Response:

The following line of text in section 3.3.8.2 RCP Motor Analysis should be deleted for the reasons discussed below: "The raised Tcold will cause a small amount of RCP loading decrease, while the increased power in the core will cause a slight increase in loading due to increased differential pressure."

This text may cause confusion without any added value even though the statement is correct within the given context. The text was intended to provide the thought process used in the evaluation. Increasing Tcold, by itself, does reduce RCP loading since the density of the water being pumped by the RCP goes down, hence less work is being done. Increasing core power, by itself, will result in greater fluid heating and attendant greater decrease in fluid density as the primary coolant passes through the core. This additional decrease in density results in an increased fluid velocity and therefore an increased pressure drop through the core. Each of these effects is negligible for a 1.42% increase in reactor power. The description in the core uplift Section 3.3.7 is correct.

Item 6. Reconcile the statement that "there is no direct effect in SG tube integrity" and discussion that RCS temperature may increase.

Response:

The Description and No Significant Hazards Analysis for Proposed Change NPF-10/15-514 is rewritten as follows to address this question:

3.4.1 Tube Performance

The effect of temperature on Inconel 600 steam generator tubing has been well documented by the industry. Therefore, the proposed power uprate will be managed to control and limit the change effect on steam generator tube integrity. An increase in RCS temperature requires evaluation in conjunction with SONGS procedures and practices for managing the steam generators tubing integrity. The tube integrity will continue to be monitored and maintained through the SONGS Technical Specifications 5.5.2.11 and the SONGS Steam Generator Program. SONGS had previously reduced the reactor coolant system operating temperature approximately 13 degrees. The effect of proposed uprate on the tube integrity will be controlled by procedures and practices consistent with NEI 97-06 (Reference) and take into consideration relevant operating experience and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity is maintained. These procedures and practices provide active measures to ensure that the effects of tube degradation are being safely managed. Operational assessments, which consider operating experience, are required each cycle. If these assessments indicate an impact to operating intervals corrective or compensatory measures to ensure tube integrity are implemented.

Although the San Onofre Technical Specifications allow operation at significantly higher RCS temperatures than that which SONGS currently operates, procedures will restrict RCS temperatures to limit steam generator tube degradation. Temperature changes will be reviewed and evaluated on a cycle by cycle basis to ensure that steam generator tube integrity is maintained.

Reference: NEI 97-06, "Steam Generator Program Guidelines Revision 1," dated January 2001

Item 7. Provide the RCS temperature used in the procedure to enter Shutdown Cooling. Verify sufficient condensate is available.

The required condensate inventory is available and is unaffected by the uprate because the current analysis of record assumes 102% RTP.

To initiate shutdown cooling (SDC) the Emergency Operating Instructions require an RCS cool down to 375°F, or if possible, to 350°F. However, the SDC system is designed such that at 400°F it can be used to remove decay heat. Technical Specification 3.7.6 "Condensate Storage Tank (CST T-121 and T-120)" requires minimum levels to be maintained to ensure sufficient condensate is available, 144,000 gallons in T-121 and 360,000 gallons in T-120. The details of supporting these minimum level requirements are provided in the following four letters:

1) Letter from Robert Dietch (SCE) to H.R. Denton (NRC), Dated August 16, 1982; Subject: Amendment Application No. 9, San Onofre Nuclear Generating Station, Unit 2.

2) Letter from George Knighton (NRC) to Robert Dietch (SCE), Dated October 26, 1982; Subject: Issuance of Amendment No. 10 to Facility Operating License NPF-10, San Onofre Nuclear Generating Station, Unit 2.

3) Letter from D. E. Nunn (SCE) to the Document Control Desk (NRC) Dated January 11, 1999; Subject: Docket Nos. 50-361 and 50-362, Condensate Storage Tank Volume, Amendment Applications Numbers 185 and 171, Change to Technical Specification 3.7.6 "Condensate Storage Tank (CST T-121 and T-120)", San Onofre Nuclear Generating Station, Units 2 and 3

4) Letter from L. Raghavan (NRC) to H. B. Ray (SCE) Dated February 22, 2000; Subject: San Onofre Nuclear Generating Station, Units 2 and 3 - Issuance of Amendments Re: Condensate Storage Tank Capacity (TAC Nos. MA4569 and MA4570)

Item 8. Determine the effect of Xenon on the timing of operator action following a CVCS Malfunction. Provide a clarification in section 15.4.1.4 of Table 4-2, if necessary.

Response:

Revise the "Impact" column text to the following:

> This is not a Mode 1 event. In lower modes this event is initiated from the reactivity associated with the minimum Technical Specification 3.1.1 shutdown margin, with no credit for xenon. Therefore, the reactivity effects of xenon do not adversely impact the event progression. Thus, this event is not impacted by the power uprate.

Item 9. Clarify the wording of section 15.4.1.5 of Table 4-2.

Response:

In the "Impact" column, change "operational" to "operating."

Enclosure 4a

Replacement Pages for

Proposed Change Number 514

San Onofre Units 2 and 3

(Text additions are shaded and deletions are in strike-out)

3.3.1.3 Safety Injection System

The Safety Injection System (SIS) is an Engineered Safety Features System designed to provide emergency core cooling and combined reactivity control following any loss of reactor coolant accident. The basic functions of this system include providing short- and long-term core cooling and maintaining core shutdown reactivity margin following an accident. The SIS is also referred to as the Emergency Core Cooling System (ECCS). The SIS accomplishes this function by providing borated water from the Refueling Water Storage Tank (RWST) to the RCS by means of the High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) pumps. Borated water is also provided to the RCS from the Safety Injection Tanks (SITs) for Large Break LOCAs, certain Small Break LOCAs, and for certain MSLB Accidents.

The revised operating conditions have no direct effect on the overall performance capability of the SIS. The accident analysis for these systems was performed at reactor operating conditions based on 102% of the original licensed power and would thus remain unchanged by this modification.

3.3.1.4 Low Temperature Over-Pressurization (LTOP)

The LTOP relief valve provides overpressure protection to the RCS at low temperature conditions during shutdown cooling when the shutdown cooling system suction valves are open and the shutdown cooling system is not isolated from the RCS. This change will not impact LTOP as design operating conditions during shutdown cooling are not affected.

3.3.1.5 Pressurizer Safety Valves (PSV)

The Pressurizer Safety Valves are not impacted by uprate because the safety analysis continues to meet the acceptance criteria for primary pressure with the initial conditions of 3458 MWt.

3.3.2 Reactor Vessel Fluence

The existing fast neutron fluence data used in the reactor vessel design remains bounding for the uprated power conditions. This conclusion is based on a fluence evaluation performed in conjunction with the withdrawal of surveillance capsules at San Onofre. Technical Specification LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits, was developed based on the projected fluence at 20 Effective Full Power Years (EFPY). Currently, both units have accumulated above 13.6 EFPY. The power uprate from 3390 MWt to 3438 MWt may result in a slight increase (1.4%) in the flux level and a negligible (< 1%) increase in the 20 EFPY fluence. Furthermore, a reduction in the original fluence estimate was realized when reactor inlet temperature was reduced from 553 °F to 540 °F per reference 8.3. The reductions in fluence are measured and incorporated in completing technical specification surveillance requirement 3.4.3.2 (10CFR50 Appendix H) controlling reactor vessel material irradiation surveillance specimen removal and examination. In the most recent Unit 2 refueling outage (13.6 EFPY), a surveillance capsule was removed and efforts are underway to evaluate and project the vessel fluence. The uprated power of 3438 MWt

3.3.8.2 RCP Motor Analysis

The RCP motors were evaluated for the limiting case loads based on the revised operating conditions for continuous operation, for starting, and for loads on thrust bearings. It was determined that for operation at the revised operating conditions, the RCPs continue to comply with their applicable hot and cold loop operating ratings. The proposed post uprate operating condition is between the current post Tcold operating condition and the original plant design. The RCPs are able to accelerate at the resultant loads for the limiting case design conditions, and the thrust bearings do not exceed their load ratings. ~~The raised Tcold will cause a small amount of RCP loading decrease, while the increased power in the core will cause a slight increase in loading due to increased differential pressure.~~ A review of the pump curves show that there will be a negligible change in efficiency or motor/pump loading due to this power uprate and all parameters stay within design criteria.

3.4 Steam Generators (SG)

Operation of the SONGS Units 2 and 3 steam generators was reviewed for the proposed post uprate operating parameters. The proposed post-uprate RCS parameters fall between the current operating conditions (post-Tcold reduction) and the original design.

3.4.1 Tube Performance

The effect of temperature on Inconel 600 steam generator tubing has been well documented by the industry. Therefore the proposed power uprate will be managed to control and limit the change effect on steam generator tube integrity. An increase in RCS temperature requires evaluation in conjunction with SONGS procedures and practices for managing the steam generators tubing integrity. The tube integrity will continue to be monitored and maintained through the SONGS Technical Specifications 5.5.2.11 and the SONGS Steam Generator Program. SONGS had previously reduced the reactor coolant system operating temperature approximately 13 degrees. The effect of proposed uprate on the tube integrity will be controlled by procedures and practices consistent with NEI 97-06 (Reference) and take into consideration relevant operating experience and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity is maintained. These procedures and practices provide active measures to ensure that the effects of tube degradation are being safely managed. Operational assessments, which consider operating experience, are required each cycle. If these assessments indicate an impact to operating intervals corrective or compensatory measures to ensure tube integrity are implemented.

Although the San Onofre Technical Specifications allow operation at significantly higher RCS temperatures than that which SONGS currently operates, procedures will restrict RCS temperatures to limit steam generator tube degradation. Temperature changes will be reviewed and evaluated on a cycle by cycle basis to ensure that steam generator tube integrity is maintained.

~~SONGS Technical Specifications and the SONGS Steam Generator Program require monitoring of tube integrity. SONGS procedures and practices are consistent with NEI 97-06 (reference 8.7) and take into consideration relevant operating experience and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity is maintained. These procedures and practices provide active measures to ensure that the effects of tube corrosion are being safely managed. Steam generator tube integrity assessments, which consider operating experience, are required each cycle. If these assessments dictate, corrective or compensatory measures to ensure tube integrity are implemented.~~

~~The proposed power uprate has no direct effect on steam generator tube integrity. However, due to the current plant configuration, an increase in RCS temperature may be required to make full use of the proposed uprate. Any increase in RCS temperature is evaluated in conjunction with the SONGS procedures and practices for managing the steam generators discussed above. As such, although the San Onofre Technical Specifications allow operation at significantly higher RCS temperatures than that which SONGS currently operates, current procedures and practices restrict RCS temperatures to limit steam generator tube degradation.~~

~~In the past, the SONGS practice for managing steam generators at San Onofre have led to reduced RCS temperature, with a corresponding impact on main generator output. These practices will continue in the future. As such, the requested uprate will be evaluated along with operating experience and potential additional physical or procedure modifications to ensure that steam~~

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.4.1.2	Uncontrolled CEA Withdrawal at Power	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	A combination of Preserved DNBR margin and the CPCS filters are set to minimize fuel failures. The filter verification is impacted by the rate of change of power and not the initial power and is thus not adversely impacted by power uprate. The trip credited for this event is the VOPT. As discussed in Section 4.1.1.5.2, this trip is used to establish an adequate transient duration for which the filter verification is performed and is thus not impacted by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
15.4.1.3	Control Element Assembly Misoperation	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	The event involves preserving DNBR margin (Section 4.1.1.5.1) such that the consequences of the event do not violate the acceptance criteria. The required thermal margin for the event is the ratio of the available thermal margin at the start of the event to the available thermal margin at the termination of the event. Since the choice of initial power equally affects the initial and final conditions for these events, the choice of initial power becomes insignificant. Therefore, the power uprate has no impact on any of the acceptance criteria.
15.4.1.4	CVCS Malfunction	Time after Boron Dilution Alarm for operator Action ≤ 15 minutes	This is not a Mode 1 event. In lower modes this event is initiated from the reactivity associated with the minimum Technical Specification 3.1.1 shutdown margin, with no credit for xenon. Therefore, the reactivity effects of xenon do not adversely impact the event progression. Thus, this event ~~it~~ is not impacted by the power uprate.
15.4.1.5	Startup of an Inactive Reactor Coolant System Pump	Shutdown % > 0.0	Per Technical Specifications the reactor must be subcritical if all four pumps are not ~~operational~~operating. Therefore, this event is not impacted by the power uprate.
15.4.3.1	Inadvertent Loading of a Fuel Assembly into an Improper Position	N/A	This event is detectable during the strartup testing via flux map at ≤ 30% power. Therefore, the event is not impacted by power uprate.

Enclosure 4b

Replacement Pages for

Proposed Change Number 514

San Onofre Units 2 and 3

3.3.1.3 Safety Injection System

The Safety Injection System (SIS) is an Engineered Safety Features System designed to provide emergency core cooling and combined reactivity control following any loss of reactor coolant accident. The basic functions of this system include providing short- and long-term core cooling and maintaining core shutdown reactivity margin following an accident. The SIS is also referred to as the Emergency Core Cooling System (ECCS). The SIS accomplishes this function by providing borated water from the Refueling Water Storage Tank (RWST) to the RCS by means of the High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) pumps. Borated water is also provided to the RCS from the Safety Injection Tanks (SITs) for Large Break LOCAs, certain Small Break LOCAs, and for certain MSLB Accidents.

The revised operating conditions have no direct effect on the overall performance capability of the SIS. The accident analysis for these systems was performed at reactor operating conditions based on 102% of the original licensed power and would thus remain unchanged by this modification.

3.3.1.4 Low Temperature Over-Pressurization (LTOP)

The LTOP relief valve provides overpressure protection to the RCS at low temperature conditions during shutdown cooling when the shutdown cooling system suction valves are open and the shutdown cooling system is not isolated from the RCS. This change will not impact LTOP as design operating conditions during shutdown cooling are not affected.

3.3.1.5 Pressurizer Safety Valves (PSV)

The Pressurizer Safety Valves are not impacted by uprate because the safety analysis continues to meet the acceptance criteria for primary pressure with the initial conditions of 3458 MWt.

3.3.2 Reactor Vessel Fluence

The existing fast neutron fluence data used in the reactor vessel design remains bounding for the uprated power conditions. This conclusion is based on a fluence evaluation performed in conjunction with the withdrawal of surveillance capsules at San Onofre. Technical Specification LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits, was developed based on the projected fluence at 20 Effective Full Power Years (EFPY). Currently, both units have accumulated above 13.6 EFPY. The power uprate from 3390 MWt to 3438 MWt may result in a slight increase (1.4%) in the flux level and a negligible (< 1%) increase in the 20 EFPY fluence. Furthermore, a reduction in the original fluence estimate was realized when reactor inlet temperature was reduced from 553°F to 540°F per reference 8.3. The reductions in fluence are measured and incorporated in completing technical specification surveillance requirement 3.4.3.2 (10CFR50 Appendix H) controlling reactor vessel material irradiation surveillance specimen removal and examination. In the most recent Unit 2 refueling outage (13.6 EFPY), a surveillance capsule was removed and efforts are underway to evaluate and project the vessel fluence. The uprated power of 3438 MWt

3.3.8.2 RCP Motor Analysis

The RCP motors were evaluated for the limiting case loads based on the revised operating conditions for continuous operation, for starting, and for loads on thrust bearings. It was determined that for operation at the revised operating conditions, the RCPs continue to comply with their applicable hot and cold loop operating ratings. The proposed post uprate operating condition is between the current post Tcold operating condition and the original plant design. The RCPs are able to accelerate at the resultant loads for the limiting case design conditions, and the thrust bearings do not exceed their load ratings. A review of the pump curves show that there will be a negligible change in efficiency or motor/pump loading due to this power uprate and all parameters stay within design criteria.

3.4 Steam Generators (SG)

Operation of the SONGS Units 2 and 3 steam generators was reviewed for the proposed post uprate operating parameters. The proposed post-uprate RCS parameters fall between the current operating conditions (post-Tcold reduction) and the original design.

3.4.1 Tube Performance

The effect of temperature on Inconel 600 steam generator tubing has been well documented by the industry. Therefore the proposed power uprate will be managed to control and limit the change effect on steam generator tube integrity. An increase in RCS temperature requires evaluation in conjunction with SONGS procedures and practices for managing the steam generators tubing integrity. The tube integrity will continue to be monitored and maintained through the SONGS Technical Specifications 5.5.2.11 and the SONGS Steam Generator Program. SONGS had previously reduced the reactor coolant system operating temperature approximately 13 degrees. The effect of proposed uprate on the tube integrity will be controlled by procedures and practices consistent with NEI 97-06 (Reference) and take into consideration relevant operating experience and appropriate diagnostic, corrective, or compensatory measures to ensure tube integrity is maintained. These procedures and practices provide active measures to ensure that the effects of tube degradation are being safely managed. Operational assessments, which consider operating experience, are required each cycle. If these assessments indicate an impact to operating intervals corrective or compensatory measures to ensure tube integrity are implemented.

Although the San Onofre Technical Specifications allow operation at significantly higher RCS temperatures than that which SONGS currently operates, procedures will restrict RCS temperatures to limit steam generator tube degradation. Temperature changes will be reviewed and evaluated on a cycle by cycle basis to ensure that steam generator tube integrity is maintained.

3.4.2 Structural Integrity

The bases for the existing structural and fatigue analyses of the steam generators are contained in reference 8.8.

The existing structural and fatigue analysis of the steam generators in SONGS Units 2 and 3 was reviewed by comparing the uprate and the analysis of record conditions to determine if the analysis of record conditions remain bounding. The review considered the most critical components with regard to stress and fatigue usage and found that the structural and fatigue conditions for the proposed increase in RTP remain bounded by existing analyses.

3.4.2.1 Upper Bundle Wear

Wear at tube support structures is a known degradation mechanism at SONGS. At SONGS, rapid wear was observed on tubes surrounding the stay cylinder in the center of the steam generator during the first cycle of operation. Many tubes in the most susceptible region around the stay cylinder have been preventively plugged. The first preventive plugging was done after 0.7 EFPY of operation. The preventively plugged region was expanded during the Cycle 3 outage. Typical active wear in CE designed steam generators has occurred at the support structures in the upper bundle region of the steam generator. These supports consist of diagonal straps (frequently called bat wings) and vertical strap supports.

This currently active wear mechanism is influenced by both flow velocities and tube to support gap wear. The variable influenced by the proposed uprate is the inner bundle flow velocities. Accordingly, wear growth rates will be managed by existing steam generator programs.

3.4.2.2 Eggcrate Wear

Visual inspections of the secondary side of the SONGS Unit 3 steam generators prior to chemical cleaning revealed significant degradation of the peripheral regions of eggcrate tube support structures. These inspection findings and subsequent root cause failure analysis have been previously documented. Removal of the deposits through steam generator chemical cleaning has arrested flow accelerated corrosion (FAC) in the eggcrate lattice structure.

Because the root cause of eggcrate wear was determined to be highly localized in the steam generator periphery due to excessive deposit build up, the proposed uprate will not affect the periphery eggcrate wear.

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15.4.3.1	Inadvertent Loading of a Fuel Assembly into an Improper Position	N/A	This event is detectable during the strartup testing via flux map at ≤ 30% power. Therefore, the event is not impacted by power uprate.