SOUTHERN CALIFORNIA
EDISON
An EDISON INTERNATIONAL℠ Company

Dwight E. Nunn
Vice President

April 3, 2001

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

Gentlemen:

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

Enclosed are Amendment Application Number 207 to Facility Operating License NPF-10, and Amendment Application Number 192 to Facility Operating License NPF-15, for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The Amendment Applications consist of Proposed Change Number (PCN)-514 and are provided in Enclosure 1 to this letter.

PCN-514 is a request to increase the licensed thermal power level and to increase the Technical Specifications definition of thermal power for operation of San Onofre Nuclear Generating Station Units 2 and 3 to 3438 MWt. This proposed change results from increased feedwater flow measurement accuracy which will be achieved by using high accuracy ultrasonic feedwater flow measuring instrumentation.

No other changes to the Technical Specifications are required to permit the proposed increase in power. SONGS recently analyzed and received NRC approval to lower Tcold, and therefore the operating conditions associated with the proposed power uprate are enveloped by the range of temperatures permitted by the existing Technical Specifications. In addition, Southern California Edison (SCE) will ensure that implementing the requested uprate is accomplished in such a manner to preclude any unacceptable degradation of the plant.

A001

P. O. Box 128
San Clemente, CA 92674-0128
949-368-1480
Fax 949-368-1490

SCE requests these amendments be issued by July 2001 and be effective as of their date of issuance, to be implemented within 30 days from the date of issuance.

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

Sincerely,

Enclosure

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
J. A. Sloan, NRC 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 Amendment Application No. 207. This amendment application consists of 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 3rd day of April, 2001.

Respectfully submitted,
SOUTHERN CALIFORNIA EDISON COMPANY

By: Dwight E. Nunn
Dwight E. Nunn
Vice President

State of California
County of San Diego
On April 3, 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 Amendment Application No. 192. This amendment application consists of 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 3rd day of April, 2001.

Respectfully submitted,
SOUTHERN CALIFORNIA EDISON COMPANY

By: Dwight E. Nunn
Dwight E. Nunn
Vice President

State of California
County of San Diego
On April 3, 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

Enclosure 1

Proposed Change Number 514

San Onofre Units 2 and 3

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DESCRIPTION AND NO SIGNIFICANT HAZARDS ANALYSIS
FOR PROPOSED CHANGE NPF-10/15-514
San Onofre Nuclear Generating Station Units 2 and 3

Proposed Change Number 514 is a request to revise the licensed Rated Thermal Power (RTP) for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, from the originally licensed rating of 3390 MWt to 3438 MWt.

EXISTING LICENSE AND TECHNICAL SPECIFICATIONS

Unit 2: See Attachment A
Unit 3: See Attachment B

PROPOSED LICENSE AND TECHNICAL SPECIFICATIONS
(highlight for additions and strike out for deletions)

Unit 2: See Attachment C
Unit 3: See Attachment D

PROPOSED LICENSE AND TECHNICAL SPECIFICATIONS
(with changes)

Unit 2: See Attachment E
Unit 3: See Attachment F

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES
For Information Only (highlight for additions and strike out for deletions)

Unit 2: See attachment G
(Unit 3 proposed Bases changes are similar to Unit 2)

DESCRIPTION OF CHANGE

1.0 Introduction

The Operating Licenses for Unit 2 (NPF-10) and Unit 3 (NPF-15) section 2.(1) identify the maximum core thermal power level for which SONGS Units 2 and 3 are authorized to operate as 3390 megawatts thermal (MWt). SONGS evaluated 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.

These amendment requests are to increase maximum core thermal power for SONGS Units 2 and 3 from 3390 MWt to 3438 MWt. (Note: all references to RTP in this submittal refer to 3390 MWt unless otherwise stated.) The definition of RATED THERMAL POWER in the Units 2 and 3 Technical Specifications will also be changed to read:

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3438 MWt.

An evaluation was performed to assess the impact on plant operations and the safety analysis to determine if any other license changes are needed. Southern California Edison (SCE) has concluded that no additional changes to the license are needed to accommodate the change in definition of 100% RTP from 3390 MWt to 3438 MWt.

Along with the proposal to increase the reactor thermal power to 3438 MWt, SONGS 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 Nuclear Regulatory Commission (NRC) approved methodologies which support the SONGS safety analysis, including the small break and large break loss of coolant accidents analyses. In many of these topical reports, reference is made to the use of a 2% uncertainty applied to the reactor power, consistent with 10CFR50 Appendix K (reference 8.1). SONGS proposes that these topical reports be approved for use consistent with these license amendment requests, and further, the NRC acknowledge that the change in the power uncertainty does not constitute a significant change, as defined in 10CFR50.46 and 10CFR50 Appendix K.

1.1 10CFR50, Appendix K

The NRC has amended its regulations to allow holders of operating licenses for nuclear power plants to reduce the assumed reactor power level used in evaluations of emergency core cooling system (ECCS) performance (reference 8.1). This amendment provides licensees the option to apply a reduced uncertainty for ECCS evaluation. This action allows SONGS Units 2 and 3 to pursue an approximate 1.42% cost beneficial power uprate without compromising the margin of safety of the facility.

A Westinghouse (formerly ABB Combustion Engineering (CE) Nuclear Power, Inc.) Advanced Measurement Analysis Group (AMAG) CROSSFLOW system consisting of ultrasonic sensors on the main feedwater and steam generator blowdown pipes, cables, and data processing computer will be installed to decrease the instrument uncertainty associated with measuring 100% reactor power level to less than 0.58%. The CROSSFLOW system and compliance with the associated NRC Safety Evaluation Report (SER)(reference 8.2) are discussed in Section 2.

1.2 Analyses Performed in Support of Previous Amendments

1.2.1 Tcold Reduction

On February 12, 1999, the NRC issued Amendment No. 149 to Facility Operating License No. NPF-10 and Amendment No. 141 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3 (Tcold Reduction Amendment) (reference 8.3). These Amendments modified the Technical Specifications to reduce the minimum reactor coolant system (RCS) cold leg temperature (Tcold) at or above 70% power, as well as other changes.

Many of the analyses performed to justify the reduction in Tcold bound the current request to increase licensed rated thermal power by 1.42%, including RCS pressure, RCS temperature, steam generator (SG) pressure, and SG outlet temperature a analyses).

2.0 Instrumentation

2.1 CROSSFLOW System

The Westinghouse CROSSFLOW (reference 8.5) ultrasonic flow measurement system (UFM) is used in conjunction with the AMAG high-accuracy ultrasonic temperature measurement system (UTM) and two plant process computers, the Plant Monitoring System (PMS) and Core Operating Limit Supervisory System (COLSS) Backup Computer System (CBCS), to support the increase in reactor power (see Figure 2-1). Reactor power is calculated in COLSS, which resides in the PMS and CBCS plant process computers, from values (including feedwater flow, feedwater temperature, steam flow, and blowdown flow) that are based on corrections from the CROSSFLOW system. The components and information flow path are shown in Figure 2-1, "Block Diagram of SONGS Ultrasonic Systems and Computers."

The CROSSFLOW system for each unit consists of ultrasonic sensors that are permanently mounted on the main feedwater and steam generator blowdown pipes, cables, signal conditioning equipment, and a data processing computer. Each unit has two main feedwater lines and two steam generator blowdown lines. The feedwater sensors measure total main feedwater flow to each steam generator and are located downstream of points that could inject main feedwater.

Each CROSSFLOW flow sensor consists of four (4) transducers mounted on a metal support frame that attaches, externally, to the feedwater and blowdown piping. The CROSSFLOW flow sensors on the main feedwater system will replace an earlier version of the CROSSFLOW system that has been used to periodically verify the accuracy of the feedwater flow venturies and to calibrate the main steam flow venturies. The blowdown system will also be instrumented with a CROSSFLOW system.

Ultrasonic temperature sensors will be externally located on the feedwater and blowdown piping near the CROSSFLOW flow sensors. The ultrasonic temperature measurement sensor will consist

of a separate set of transducers, brackets, and cables. The system operates on the principle of ultrasonic signal transit time to determine the temperature of the process fluid.

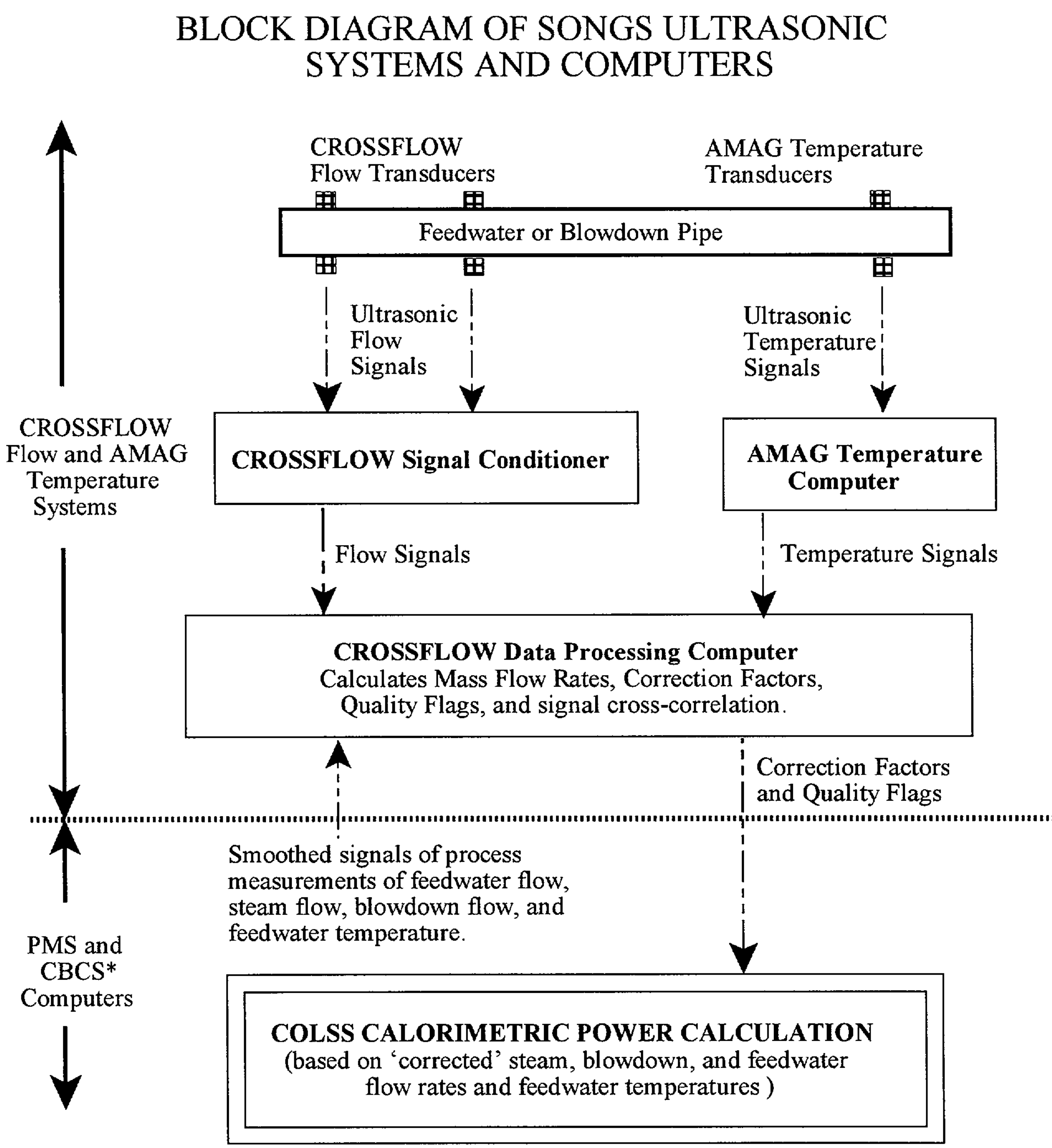
Cables from ultrasonic flow and temperature sensors will be routed to signal conditioning equipment and data processing computers located in a non-harsh area. The functions of the CROSSFLOW flow signal conditioning equipment and data processing computer are described in the Topical Report (reference 8.5).

The data processing computer receives smoothed values of process measurements of main feedwater flow, main feedwater temperature, main steam flow, and blowdown flow for each loop from the PMS and CBCS. The data processing computer calculates main steam flow for each steam generator from the difference between the ultrasonically-determined feedwater and blowdown flow rates for that generator. The data processing computer compares long-duration averages of the process measurements to long-duration averages of the ultrasonically-determined flows and temperatures, and produces correction factors for feedwater flow, feedwater temperature, blowdown flow, and main steam flow. The data processing computer also determines if each correction factor is sufficiently valid for use and, if valid, sets its quality flag to "good."

The data processing computer then sends the correction factors and quality flags to the PMS and CBCS. If the quality flags are "good" and the plant is above a minimum power level, the COLSS programs in each computer multiply plant process signals of feedwater flow, feedwater temperature, blowdown flow, and main steam flow by their associated correction factors to produce corrected process values. COLSS then uses the "corrected process values" to calculate reactor power and allows operation up to 3438 MWt. Two separate plant computers, PMS and CBCS, are available to calculate reactor power in COLSS. Under our current COLR values for Linear Heat Rate (LHR) and Departure from Nucleate Boiling Ratio (DNBR), a COLSS out of service condition may preclude full power operation based on Core Protection Calculator (CPC) operating margin.

If COLSS programs have been using correction factors and the quality flags become "bad," the COLSS programs will continue to use the last good correction factors for a predetermined time. When the quality flag changes to "bad," the PMS/CBCS computers alarm. If the quality flags for feedwater flow or steam flow rates can not be restored to "good" within a predefined interval, the correction factors will be changed to conservative default values. Continued operation of the COLSS programs with a "bad" quality flag does not affect safety since the COLSS programs will continue to use a "good" correction factor. The implemented predetermined time for using the last good correction factor will be calculated based on instrument drift characteristics of the main steam and feedwater venturi signals.

Figure 2-1, Block Diagram



*CBCS is run as a backup channel and is typically used only when PMS is not available

2.2 Compliance with the NRC SER

The installation of the CROSSFLOW flow measurement system at SONGS Units 2 and 3 complies with Topical Report CENPD-397-P-A (reference 8.5). In addition to the installation requirements, the NRC identified the following four criteria that must be addressed by licensees requesting a license amendment based on the Topical Report (reference 8.5). SONGS will comply with the four criteria described below.

2.2.1 Maintenance and Calibration Procedures

The first criterion is to develop 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.

Installation, maintenance, and calibration will be performed using SONGS maintenance and calibration procedures, which will be developed from vendor information and SONGS-specific experience, or will be performed by a combination of vendor procedures and SONGS procedures.

Verification of CROSSFLOW system operation is provided by onboard system diagnostics. CROSSFLOW operation will be monitored on a periodic basis using an internal time delay check. The onboard system diagnostics enable verification that the Signal Conditioning Unit, computer, and software remain within the stated accuracy.

An inoperable CROSSFLOW system will cause "bad" quality flags and will result in the actions discussed previously for "bad" quality.

2.2.2 Currently Installed UFM

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 Topical Report (reference 8.5).

The existing system is not consistent with the installation requirements of the Topical Report because it lacks the required upgraded components. The upgraded CROSSFLOW system, when it is installed, will satisfy the requirements of the Topical Report (reference 8.5) and will be bounded by them.

Since December 1997, an earlier version of the CROSSFLOW UFM has been successfully used numerous times at SONGS to measure feedwater flow rate. This existing CROSSFLOW UFM is used to verify the feedwater flow signal and calibrate the steam flow signals used by the COLSS program. The calibrated main steam flow signals have permitted the plant to operate closer to its licensed power limit of 3390 MWt.

Operation at higher power than 3390 MWt will require replacing the existing CROSSFLOW system. The existing CROSSFLOW UFM has an operational and maintenance history that consists of periodically calibrating the electronics, removing and reinstalling the transducers and their brackets, and tuning the system to make measurements. The existing system's brackets are installed on the main feedwater lines, and the ultrasonic transducers and associated electronics are installed only during the measurement. Considerable experience has been gained in setting up and tuning the equipment, as well as conducting measurements using an existing SONGS procedure. This experience will be directly applicable to the installation, calibration, tuning, and use of the upgraded CROSSFLOW instruments.

At SONGS, the location of the existing CROSSFLOW UFM is representative of the location requirements set forth in the Topical Report (reference 8.5). The upgraded CROSSFLOW UFM on the feedwater line will be located where the brackets for the existing units are installed, and this location, as discussed later in section 2.2.4 below, meets the requirements for installation.

2.2.3 Calculation Methodology

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.

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 in Regulatory Guide 1.105 and ISA S67.04, as described in the Topical Report (reference 8.5). An alternative methodology is not used.

Westinghouse has completed the SONGS Units 2 and 3 CROSSFLOW uncertainty calculation with a mass flow accuracy of equal to or better than 0.5% of rated feedwater flow for the SONGS Units 2 and 3 site-specific installation. SONGS also uses main steam flow to determine reactor thermal power level. The CROSSFLOW system calculates the rate of main steam flow by subtracting the rate of CROSSFLOW UFM blowdown flow from the rate of CROSSFLOW UFM main feedwater flow. The calculated uncertainty of the rate of steam mass flow is equal to or better than 0.53% of the actual rate of main steam flow at full power. The SONGS CROSSFLOW uncertainty calculations are consistent with the methodology described in the Topical Report (reference 8.5).

2.2.4 Site-Specific Calibration

Finally, the fourth criterion is 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 previously installed and calibrated CROSSFLOW UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the CROSSFLOW UFM Topical Report (reference 8.5).

For SONGS, there will be no site-specific piping 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 numbers.

The meter installations are located on long straight sections of piping and will be far enough from disturbances to conform to the proprietary installation requirements of the Topical Report (reference 8.5).

3.0 Effect of the Uprate on the Plant

3.1 Tcold Submittal

In response to a Request for Additional Information (RAI) from the NRC, SCE provided, by letter dated January 13, 1999 (reference 8.4), information in support of reducing the value of licensed Tcold (reference 8.3). This letter responds to the RAI to:

> "Provide a summary of the evaluations (including analytical methodology, assumptions, and maximum stress and fatigue usage factors) for the effects of Tcold reduction on the structural and pressure boundary integrity of the reactor vessel and internals, RCS piping, control rod drive mechanisms and housing, pressurizer, surge line (stratification), pressurizer spray nozzles, SGs, reactor coolant pumps, and pressurizer power-operated valves and safety valves. Identify changes in maximum stress and fatigue usage factors (at critical locations) from your evaluation."

Since the key design parameters (RCS pressure, RCS temperature, SG pressure, and SG outlet temperature) for this amendment request fall on or between the current operating conditions (post-Tcold reduction) and the original plant design, no new review and approval is required for the change in these key design parameters.

3.2 Containment

Pressure-Temperature (P/T) Transient Analysis determines the containment pressure and temperature response following the mass and energy releases from a high energy line break. The design basis breaks are large Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) events. Design basis LOCA and MSLB events in containment were reanalyzed during the Tcold Reduction Project. The mass and energy releases used in the reanalysis of the limiting

events are based on a reactor power level of 3458 MWt, 102% of the original licensed power level of 3390 MWt.

3.3 Nuclear Steam Supply System (NSSS)

3.3.1 NSSS Fluid Systems

3.3.1.1 Reactor Coolant System

The Reactor Coolant System (RCS) is a pressurized closed loop system with two coolant loops. The system consists of one reactor vessel and two parallel coolant loops. Each coolant loop contains one steam generator (SG), two reactor coolant pumps (RCPs), and associated piping. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to the secondary, or main steam generating, system. The SGs provide the interface between the RCS and the main steam system. Reactor coolant is separated from the secondary system fluid by the steam generator tubes and tube sheet, making the RCS a closed system and forming a barrier to the release of radioactive materials from the core. A pressurizer is connected to one hot leg to maintain system pressure through electric heaters and water sprays. Overpressure protection is provided by spring-loaded safety relief valves. The steam discharged from the safety relief valves flows through interconnecting piping to the quench tank.

The proposed RCS post-uprate parameters fall on or between the current operating conditions (post-Tcold reduction) and the original design. Assessments were performed that demonstrated that the RCS design basis functions could still be met at the revised operating conditions, which are similar to the RCS operating conditions prior to implementation of this change. Pressurizer spray flow capability is also not impacted by this change.

3.3.1.2 Chemical and Volume Control System

The Chemical Volume and Control System (CVCS) provides for boric acid addition, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant system make-up, and reprocessing of letdown water from the RCS. During plant operation, letdown flows through the tube side of the regenerative heat exchanger and then through the letdown control valve. The regenerative heat exchanger reduces the temperature of the letdown flow, and the letdown control valve modulates to maintain the desired pressurizer level. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by the final pressure reduction across the backpressure control valve. After passing through the reactor coolant filter, coolant flows through the ion exchangers, where ionic impurities are removed, and enters the Volume Control Tank (VCT).

The proposed RCS post-uprate parameters fall on or between the current operating conditions (post-Tcold reduction) and the original design. Since the RCS operating conditions will be similar following this change, operation of the CVCS will not be impacted by this change.

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 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

and fluence will be used to determine any changes if needed to LCO 3.4.3.

3.3.3 Reactor Internals

The reactor internals support and orient the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, direct flow to provide adequate cooling to various internal structures, and support the in-core instrumentation. With little or no increase in thermal design flow and changes in the RCS temperatures there will be little or no changes in the boundary conditions experienced by the reactor internals components. Increases in core thermal power will slightly increase nuclear heating rates in the reactor vessel internals, such as lower core support plate, fuel alignment plate, and core shroud, but are within the design capability of the system analysis. Evaluations have been performed verifying the increased gamma-heating will not affect the calculated component lifetimes.

3.3.4 Core Bypass Flow

Bypass flow corresponds to the amount of reactor coolant flow that bypasses the core region and is not considered effective in the core heat transfer process. The principal core bypass flows are the outlet nozzle clearances, alignment key-ways support cylinder hole, core shroud clearances, and the guide tube holes. The increase in power to 3438 MWt will impact plant parameters, such as coolant temperature and density, but will not impact any key parameters of the core bypass calculation (i.e., as-built tolerances, clearances, and guide tube dimensions). Therefore, the impact of the power uprate on core bypass flow is insignificant. For these reasons it is concluded that the proposed power uprate will not adversely impact the core bypass flow used in the safety analyses.

3.3.5 Control Element Assembly (CEA) Drop Time Analyses

Technical Specification surveillance requirement 3.1.5.5 requires that the average CEA drop time be less than or equal to 3.4 seconds. The CEA drop times are explicitly confirmed, by measurement after each refueling outage, to meet the times assumed in the accident analyses. An evaluation was performed for all Combustion Engineering (CE) designed plants to demonstrate continued compliance with the current technical specification value based on CE's robust CEA five finger design, which has not shown any failure to insert at any time in life through the end of life core burnup. Uprate to 3438 MWt will increase the power level slightly in leading rodded fuel assemblies but will not change the burnup levels of those fuel assemblies, since the excess reactivity will be depleted faster. Fluence induced changes in grid cage structures will not be affected.

3.3.6 Mechanical Evaluations

The revised operating conditions do not affect the current design bases for seismic and LOCA loads since the revised operating conditions are enveloped by evaluations performed for the Tcold

reduction project. Therefore, it was not necessary to re-evaluate the structural affects from seismic Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) loads and the LOCA hydraulic and dynamic loads. With regards to flow and pump induced vibration, the current analysis is based on a mechanical design flow which was not impacted by the revised operating conditions. Thus, the impact of the revised operating conditions is considered insignificant on the seismic and LOCA induced loads.

3.3.6.1 Structural Evaluations

Structural evaluations performed for the Tcold reduction project were reviewed and determined to bound this change. As such, the structural integrity of the reactor components is not adversely affected by the uprate to 3438 MWt.

3.3.6.2 Leak Before Break

The original plant design basis considered various size breaks at several locations in the main coolant loop piping. After development of the original design basis, additional analyses were performed to demonstrate that a rupture of the RCS main loop piping would be preceded by detectable leaks, rather than resulting in sudden catastrophic failure. This methodology is defined as Leak-Before-Break (LBB), and SCE has been authorized by the NRC to implement the LBB methodology at San Onofre Units 2 and 3 (reference 8.6). The proposed post-uprate key design parameters (RCS pressure, RCS temperature, SG pressure, and SG outlet temperature) fall on or between the current operating condition (post-Tcold reduction) and the original design. Therefore, the piping loads and seismic loads under the proposed uprate condition are bounded and LBB is not impacted by this proposed change.

3.3.7 Core Uplift Forces

A core uplift evaluation was performed as part of the fuel mechanical design to determine hydraulic uplift forces from the reactor fuel in the core and ensure that the reactor fuel assemblies remain seated and stable for all operating conditions. This evaluation was performed at a conservative RCS temperature of 500°F bounding all at power operating conditions for hydraulic drag. A minor increase in operating temperature would only reduce hydraulic drag of the core and other reactor internals. Core uplift analyzed conditions remain bounding for uprate to 3438 MWt.

3.3.8 Reactor Coolant Pumps (RCPs)

3.3.8.1 RCP Structural Analysis

The proposed RCS post-uprate parameters fall on or between the current operating conditions (post-Tcold reduction) and the original design. Structural evaluations performed on the RCPs for the Tcold reduction project were reviewed and determined to bound this change.

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

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

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.

3.4.2.3 Thermal-Hydraulic Performance

Secondary side steam generator performance characteristics such as circulation, moisture carryover, hydrodynamic stability, and heat flux are affected by increases in thermal power and steam pressure. These parameters were reviewed as part of the Tcold reduction package and are contained in reference 8.4. The magnitude and importance of changes in the secondary side thermal hydraulic performance characteristics at the uprate power, with increased tube plugging, reduced primary side temperatures, and a feedwater temperature range are assessed in other sections of this document.

3.4.2.4 Circulation Ratio/Bundle Liquid Flow

The uprate will result in small increases in steam generator pressure and temperature. The effect of these changes is a slightly reduced circulation ratio.

The circulation ratio is a measure of downcomer mass flow rates into the tube bundle and is a function of feedwater flow rate. The bundle liquid flow minimizes the accumulation of contaminants on the tube sheet and in the bundle. For the uprate there is a slight increase in the feedwater flow which lowers the circulation ratio. These changes will have a minimal affect on bundle liquid flow, and are bounded by previous evaluations associated with the Tcold reduction.

3.4.2.5 Hydrodynamic Stability - Damping Factor

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit, i.e., small perturbations of steam pressure or circulation ratio will diminish rather than grow in amplitude. The damping factors remain highly negative, at a level comparable to the current design, for all cases. Thus, the steam generators remain hydrodynamically stable for all uprate cases.

Based on a projected increase of 2.3% in the secondary side fluid velocity, normal operation flow induced vibration analysis is impacted by the velocity increase. Current analysis considered that tubes with more than one consecutive inactive eggcrate were staked and plugged, and two non-consecutive inactive eggcrates are acceptable.

The Stability Ratio (SR) is defined as:

$$SR = Veff / Vcr$$

where

Veff = effective velocity;
Vcr = critical velocity;
and Values of SR < 1 are considered acceptable

The maximum stability ratios calculated are :

SR =	0.64	(one eggcrate uncredited)
SR =	0.66	(alternate eggcrates uncredited)

Ignoring any changes in the fluid density resulting from the modification, no change in Vcr is expected. As an approximation, the modified Veff is assumed to increase by 2.3%, i.e., the same as the fluid velocity. The modified maximum SR will be 0.66 x 1.023 = 0.675 < 1 (i.e., acceptable). Therefore, the existing steam generator eggcrate evaluation will not be impacted by the uprate.

3.5 NSSS / BOP Interface Systems

Important to safety Balance-of-Plant (BOP) fluid systems were reviewed for compliance with existing system design requirements. Summaries of the evaluations are provided below.

3.5.1 Main Steam System

At 100% power operation, steam generator pressures typically vary between 800 psia and 815 psia, compared to the original nominal design operating pressure of 900 psia. The lower steam generator pressure is the result of a recent reduction in the normal range of RCS operating temperatures. The uprate will result in a slight increase in steam generator pressure from current nominal 100% RTP operating conditions.

The following summarizes the evaluation of the major steam system components relative to the power uprate conditions. The major components of the Main Steam System (MSS) are the Main Steam Safety Valves (MSSVs), the SG Atmospheric Dump Valves (ADVs), and the Main Steam Isolation Valves (MSIVs).

3.5.1.1 Main Steam Safety Valves (MSSVs)

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110% of the MSS design pressure (the maximum pressure allowed by the ASME B&PV Code). The MSSVs' rated capacity will pass the full steam flow at the currently assumed normal maximum operating condition of 102% RTP (100% + 2% for instrument error) with the valves full open. This meets the requirements of the ASME B&PV Code. The proposed revised operating conditions will not exceed the currently assumed maximum normal operating condition of 102% RTP. Based on the revised operating conditions, the capacity of the installed MSSVs meets the required sizing criterion.

3.5.1.2 Atmospheric Dump Valves (ADVs)

The primary function of the ADVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser or steam dump to the condenser is not available. Under such circumstances, the ADVs, in conjunction with the Auxiliary Feedwater System (AFWS), permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Shutdown Cooling System (SDCS) can be placed in service. During cooldown, the ADVs are either automatically or manually controlled. When in the automatic mode, each ADV controller automatically compares steam line pressure to the pressure setpoint, which is manually set by the plant operator. The ADV automatic setpoint can be lowered as desired to conduct a cooldown and/or to remain at nominal hot standby temperature and pressure.

In the event of a steam generator tube rupture (SGTR), in conjunction with loss of offsite power, the ADVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSVs. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

Accident analysis calculations for determining AFW condensate storage capacity requirements currently assume an initial reactor power level of 102% RTP. The proposed revised design conditions will not exceed the currently assumed maximum operating condition of 102% RTP. Therefore, the capacity of the ADVs under the new design conditions will be bounded by previous analyses.

3.5.1.3 Main Steam Isolation Valves (MSIVs) and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the MSIVs must be capable of closure within 8 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction.

The accident analysis following a steam line break currently assumes 102% RTP. The proposed revised operating conditions will not exceed the currently assumed maximum operating condition of 102% RTP. Therefore, closure requirements for the MSIVs will remain bounded by existing analysis.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-

load and low power conditions where the revised operating conditions have no impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, the revised operating conditions have no impact on the design requirements for the MSIV bypass valves.

3.5.1.4 Steam Bypass Control

To reduce the probability of reactor trips, the steam bypass control system is designed with the capacity to bypass up to 45% of the full load main steam flow (i.e., 3390 MWt core power). This capacity, along with the ability of the NSSS to absorb a 10% load step change, provides a means to absorb limited load reductions on the turbine-generator, which occur more rapidly than the reactor power level can be reduced. In conjunction with the power-operated ADVs and the spring-loaded safety valves, the bypass system is designed to prevent the main steam supply system from exceeding its design pressure during all phases of operation. The turbine bypass system also provides a means to remove stored heat, RCP heat, and decay heat from the RCS during cooldown and startup.

The original design basis of the SBCVs (Steam Bypass Control Valves) is to pass 6.88 x 10E6 lbm/hr allowing a 45% turbine load rejection based on original plant design conditions. Present plant configuration gives a 45% turbine load of a 102% RTP full load operation of 6.86 x 10E6 lbm/hr. The SBCVs remain within their design capacity ratings up to 102% RTP, so the loading increase due to the proposed power uprate of 1.42% is bounded, and the design basis is unchanged.

The operation of the Steam Bypass Control System (SBCS) is modeled through a simulation program. SBCS response was simulated assuming 102% RTP and design basis load rejections. The operation of the SBCS was satisfactory, and the system performed as designed, with no reactor trip or adverse effects. This analysis shows that a power uprate of 1.42% is bounded by the Tcold analysis and original design analysis, and the design basis is not affected.

3.5.1.5 Steam Hammer Analysis

A pressure transient would occur on the main steam lines upon sudden closure of the turbine stop valves. The thermal-hydraulic model was analyzed to simulate the fluid system response to the rapid closure of turbine stop valves from initial flow conditions at 100% load. The steam generators were modeled with an initial pressure of 900 psia and a flow rate of 4203 lbs/sec while the condenser pressure was kept at a constant value. The highest dynamic stress was registered as 11.3 ksi at a tapered weld joint, which is considered within acceptable design limits for the system. Steam hammer load was compared with Operating Basis Earthquake (OBE) inertia load and the larger of these two was selected and combined with other service loads for a Code compliance evaluation. The highest stress ratio for the upset condition was identified as 0.846. As a conservative assumption, a 2.3% flow rate increase will result in the same increase in steam hammer load as either OBE inertia or steam hammer in contributing to the highest upset condition stress ratio. Adequate margin is still available for the most conservative scenario to accommodate

the assumed 2.3% flow increase. Therefore, the existing steam hammer analysis is not impacted by the uprate.

3.5.2 Auxiliary Feedwater System

The Auxiliary Feed Water System (AFWS) supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides feedwater to the SGs during normal unit startup, hot standby, and cooldown operations and also functions as an Engineered Safety Features System. In the latter function, the AFWS is directly relied upon to supply feedwater to the steam generators in the event of transients or accidents which result in the loss of normal feedwater flow to the steam generators.

The minimum flow requirements of the AFWS are dictated by accident analyses (Section 4.1) which are unaffected by the 1.42% uprate. Therefore, the AFWS performance remains acceptable at the uprated operating condition.

3.5.2.1 Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the Engineered Safety Features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The AFW pumps are aligned to Seismic Category I condensate storage tank T121 which maintains a minimum of 144,000 gallons for use by the AFW pumps. Less than 144,000 gallons is required for use by the AFW pumps during loss of offsite power conditions and the most limiting single active failure (loss of one ADV) to cool the reactor to 400°F, the temperature at which the shutdown cooling system can be used to remove decay heat. This amount bounds a loss of the main feedwater pumps or a MSLB.

Seismic Category II storage tank T120 in combination with its seismic enclosure is connected to T121, which ensures an additional 200,000 gallons of water from a seismic source. The combined minimum useable volumes in T120 and T121 are sufficient to supply 24 hours of auxiliary feedwater pump operation for meeting SONGS commitments to Branch Technical Position RSB 5-1 "Design Requirements of the Residual Heat Removal System."

Calculations for determining AFW condensate storage capacity requirements currently assume an initial reactor power level of 102% RTP. The proposed revised operating conditions will not exceed the currently assumed maximum normal operating condition of 102% RTP. Therefore, the condensate storage requirements under the new operating conditions will be bounded by previous analysis.

3.5.3 Steam Generator Blowdown System

The Steam Generator Blowdown System, in conjunction with the condensate and feedwater chemical injection system, is capable of maintaining the chemical composition of the steam generator secondary water within specified limits. No main condenser inleakage or primary to secondary leakage is assumed in normal operation. However, the blowdown rate can be increased to maintain the specified limits with small amounts of main condenser inleakage or primary-to-secondary leakage. The blowdown system also controls the buildup of solids in the steam generator water.

The actual required blowdown flow rates during plant operation can vary depending on feedwater quality and will not be significantly impacted by the revised operating conditions, since neither the rate of addition of dissolved solids nor the rate of addition of particulates into the steam generators will be significantly impacted.

Ultrasonic flow and temperature instrumentation is being added to the blowdown lines to improve accuracy of these measurements as discussed in section 2.1.

3.6 Balance of Plant (BOP)

The BOP evaluation was focused on the parameter ranges required to support a 1.42% increase in the thermal power level. The systems that were evaluated include the feedwater and condensate, turbine generator, spent fuel pool cooling, auxiliary feedwater, turbine plant cooling, circulating water, and main steam.

3.6.1 Flow Accelerated Corrosion (FAC)

The SONGS commitment to the NRC is to identify all FAC-susceptible systems and establish a formal monitoring program so that FAC degradation can be arrested before failure occurs. FAC-susceptible systems and components have been identified and are in the scope of a monitoring program. The proposed increase in flow rate will have no effect on the program. There are no additional systems or components that need to be added to the monitoring program due to the 1.42% uprate. Program procedures will monitor the wear of all of the systems and components subjected to FAC and impacted by the proposed increase. Piping component inspections and replacements will be adjusted according to procedural requirements of the SONGS FAC Monitoring Program, based on wear data as it is collected in the FAC program. No reduction of the inspection interval, i.e., the cycle design length, will be required by this increase in flow.

3.6.2 Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The main feedwater system must also be able to automatically isolate the C&FS from the steam generators, when required, in order to

mitigate the consequences of an accident. The revised power uprate operating conditions will impact both feedwater volumetric flow and system pressure drop. However, in all cases, the conclusions of the evaluations were that the respective system design bases remain valid.

The major components of the C&FS are the Feedwater Isolation Valves, the Feedwater Isolation Block Valves, the Feedwater Bypass Regulating Valves, the Feedwater Regulating Valves, and the Condensate and Feedwater System Pumps.

3.6.2.1 Feedwater Isolation, Block, and Bypass Regulating Valves

The main feedwater isolation valves (MFIVs), main feedwater block valves (MFBV), and the bypass regulating valves are located outside containment and downstream or parallel to the feedwater control valves (FCVs). The condensate and main feedwater system is isolated from the steam generators by automatic closure of all these valves on a Containment Isolation Actuation Signal (CIAS) or by automatic closure of the MFIVs on a Main Steam Isolation Signal (MSIS). The condensate and main feedwater system is isolated from the steam generators by the MFIVs within 10 seconds of receipt of the MSIS.

The accident analyses which require the closure of these valves to isolate the main feedwater and condensate system are based on initial reactor power levels of 102% of 3390 MWt. Since the increase in reactor power level proposed by the power uprate is bounded by the 102% reactor load, the power uprate will not require any changes to the automatic operation of these valves.

3.6.2.2 Condensate and Feedwater System (C&FS) Pumps and Feedwater Regulating Valve

The C&FS pumps consist of the condensate, feedwater, and heater drain pumps. These pumps, in conjunction with the feedwater regulating and bypass regulating valves, serve to regulate the main feedwater flow to the steam generators to maintain steam generator level during steady-state and transient operation. During low and intermediate load operation, feedwater flow is controlled by the feedwater regulating and bypass regulating valves. During high load operation, feedwater flow is controlled by regulating main feedwater pump speed and the feedwater regulating valve. Each unit is supplied with 4 condensate pumps, 2 heater drain pumps, and 2 feedwater pumps. A feedwater regulating valve in parallel with a bypass regulating valve serve each steam generator. During normal operation, 3 condensate pumps, 2 heater drain pumps, and 2 main feedwater pumps are in operation with the fourth condensate pump in standby. The fourth condensate pump is automatically started on low condensate pump header pressure, low main feedwater pump suction pressure, an open 4 kv breaker on any of the condensate or heater drain pumps, or a sustained high conductivity in any of the condenser hotwells in the auto overboard mode. The C&FS is designed to permit continued full-load operation of the plant without reactor trip upon loss of one of the four condensate pumps or one of the two heater drain pumps. The result of the power uprate will be to increase by approximately 1.4% the amount of main feedwater supplied to the steam generators at full load.

The C&FS pump capacities were reviewed to determine whether they had sufficient capacity to accommodate the increased main feedwater flows. The C&FS pumps were determined to have sufficient capacity to supply the increased main feedwater flow to the steam generators for the power uprate full load conditions.

3.6.3 BOP Structural Analysis

The new system operating parameters (pressure, temperature, enthalpy) for the uprate are still bounded by the current design pressures and temperatures.

Per ASME Code III Subsection NC-3652.1 (reference 8.11), the effects due to design pressure, weight, and other sustained mechanical loads must meet the requirements of Equation (8). Subsection NC-3652.3 specifies the thermal expansion acceptance limits (Equations 10 and 11) for Service Level A and B Loadings. Subsection NC-3112.2 defines the design temperature as the expected maximum mean metal temperature through the thickness of the part considered for which Level A Service Limits are specified.

With the above code requirements as the basis, Code Class 2 piping systems at SONGS have been analyzed based on design pressures and temperatures. The analytical computer codes used in the stress analysis and support design are the latest revisions of the computer codes presently described in the Updated Final Safety Analysis Report (UFSAR) Section 3.9.1.2. Existing stress analyses and pipe support designs are not impacted by the implementation of the proposed uprate since design pressures and temperatures bound the new operating parameters.

3.6.4 Pipe Break Evaluation (for Mass and Energy Releases)

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 (MSIV/MFWIV) enclosure area. High energy fluid systems outside containment will not experience any significant increase in energy (pressure or temperature) as a result of the power uprate as noted below. Pipe break locations previously identified and evaluated for potential impact on essential safety-related systems and components remain unchanged.

The bounding pipe break outside containment from the standpoint of mass and energy release is a rupture of a main steam line downstream of the main steam isolation valve in the MSIV/MFWIV enclosure area. The mass and energy release rates for this pipe break, re-evaluated during the Tcold reduction project with the reactor at 102% power and the reactor Tcold at the Technical Specification maximum value of 560°F, bound the mass and energy releases expected with the reactor power uprated by 1.42%. Since the pressure and temperature conditions in other connected systems outside containment are not significantly affected by the power uprate, mass and energy releases for previously evaluated pipe breaks in these other systems outside containment will 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 remain unaffected by the power uprate.

Therefore, mass and energy releases from pipe breaks outside containment and consequential environmental impacts on safety-related equipment required for safe shutdown following the pipe break event will not be impacted by the power uprate.

3.6.5 Flow Induced Vibration

Negligible impact was determined for the BOP heat exchangers, the moisture separator reheater (MSR), main condenser, blowdown heat exchanger, gland steam condenser, steamjet ejector heat exchanger, and feedwater heaters based on the flow induced vibration evaluation for the uprate.

3.6.6 Motor Operated Valves (MOVs)

The MOV program at SONGS was set up in such a way that setpoints were established and MOVs were tested to demonstrate their capability to perform their safety related function. MOV setpoint evaluations include several conservatisms, and small changes in the system operating pressure are not expected to impact the operation of these MOVs. The proposed increase in flow rate has no significant impact on the operation of gate and globe MOVs since the expected changes in the differential pressure are insignificant. A small increase in flow rate would increase the valve sizing coefficients slightly for butterfly valves. However, within the Generic Letter (GL) 89-10 program, butterfly valve application is limited to component cooling water, safety injection system, salt water cooling, chilled water, containment purge, and containment sump suction systems only. These systems are not impacted by the reactor power uprate. Therefore, the existing MOV program is not affected by this change.

3.7 Electrical Systems

3.7.1 Generator and Support Systems

The electrical systems associated with the turbine auxiliary systems are not affected by the uprate.

The steam turbine-driven polyphase generator is a four pole machine with the following nominal ratings:

> 1127 MW at a 0.9 power factor when operated at 60 psig hydrogen pressure (1252.2 MVA and 32,863 Amps)
>
> 1180 MW at a 0.9 power factor when operated at 68 psig hydrogen pressure (1311.1 MVA and 34,409 Amps)

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. This is

achieved by adjusting the operating power factor of the generator within its present operating envelope as defined in the generator capability diagram.

Historically, the generator has operated at a peak of 1175 MW. 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.

A review of applicable calculations identified no need for any changes to equipment protective relay settings for the generator. However, some process alarm setpoints for the generator and the exciter may require adjustment.

To deliver electrical power from the generator to the transmission system, each unit is equipped with an isolated phase (isophase) bus, a main transformer, cabling, and two switchyard breakers. All components are rated to deliver electrical power at or in excess of the main generator rating of 1311 MVA.

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 will permit a total load of 1383 MVA. The isophase bus temperature ratings are well in excess of the anticipated generator output of 1311 MVA. The isophase bus will support the power increase with no modifications.

The main transformer is rated for 1378 MVA and will support the power increase with no modifications.

Standard design practice at SCE requires that switchyard equipment meet or exceed the rated capacity of the main generator. The SONGS switchyard will accept the additional load without the need for any hardware modifications.

In summary, the turbine/generator and major electrical components extending from the isophase bus to the switchyard have adequate design margin to accept the additional power anticipated by the 1.42% uprate.

3.7.2 Onsite Distribution System

The onsite AC power system includes a class 1E system and a non-class 1E system. The onsite AC power system consists of Units 2 and 3 main turbine-generators, unit auxiliary transformers, diesel generators, and AC distribution system with nominal ratings of 6.9 kV, 4.16 kV, 480 volts, and 208/120 volts. The onsite DC system, consisting of class 1E and non-class 1E systems, provides control power for medium voltage and low voltage switchgear, diesel generator controls, and other control systems.

3.7.2.1 Non-Class 1E AC System

The non-Class 1E AC system distributes power at 6.9 kV, 4.36 kV (nominal 4.16 kV switchgear), 480 volts, and 208/120 volts for all non-safety-related loads. The non-Class 1E AC buses normally are supplied through the unit auxiliary transformers from the main generator. However, during plant startup, shutdown, and post-shutdown, power is supplied from the 230 kV preferred offsite power source through the secondaries of the 230kV to 6.9kV and 230kV to 4.36kV reserve auxiliary transformers.

The 4.16 kV non-Class 1E auxiliary system is comprised of four buses (2A03, 2A07, 2A08, and 2A09) for Unit 2, and four buses (3A03, 3A07, 3A08, and 3A09) for Unit 3. The reactor coolant pumps for Unit 2 are fed from non-Class 1E 6.9 kV auxiliary system buses 2A01 and 2A02. Buses 3A01 and 3A02 feed power to the reactor coolant pumps for Unit 3.

In the event of failure of the unit auxiliary transformer, or other failures causing a unit auxiliary transformer breaker trip, a fast transfer to the preferred offsite power source associated with the same respective unit maintains a continuity of power to the 4.16 kV and 6.9 kV non-Class 1E AC buses. However, if power is unavailable from the respective preferred offsite power source, the 6.9 kV buses feeding the reactor coolant pumps will transfer to the companion unit's preferred offsite power source. Permissive interlocks will prevent a transfer to an offsite source if the 6.9 kV bus of the companion unit is already energized from that particular source.

The reserve auxiliary transformers are capable of supplying all of the startup or normally operating loads of one unit simultaneously with the engineered safety feature (ESF) loads associated with each unit.

3.7.2.2 Class 1E AC System

The Class 1E AC system consists of two separate trains and distributes power at 4.36 kV (nominal 4.16 kV switchgear), 480 volts, and 120 volts to safety-related loads. The Class 1E AC buses are normally supplied from the offsite source through their own unit's reserve auxiliary transformers. The Class 1E AC buses may also be supplied from the alternate offsite source through the companion unit's Class 1E AC bus of the same load group. Following unit shutdown, a third offsite power source circuit can be established by manually removing the link in the isolated phase bus between the generator and the main power transformer of the non-operating unit through the supply breaker from the unit auxiliary transformer.

Each safety-related 4.16 kV load group bus of each unit is supplied by two offsite power supply feeders and one standby (diesel generator) supply feeder. In the event of loss of all the offsite power sources, or loss of a single offsite source concurrent with a Safety Injection Actuation Signal (SIAS), the Class 1E AC system will be powered from the emergency diesel generators. In the event that one preferred offsite power feeder fails to function, the safety-related loads connected to it will transfer to the other preferred power feeder via the companion unit through

bus tie circuit breakers only if no SIAS exists.

3.7.2.3 Onsite Distribution System Review

The impact of potential increases in brake horsepower loads on non-safety related pumps (i.e., condensate pumps, heater drain pumps, circulating water pumps, etc.) due to the 1.42% power uprate have been determined to be insignificant. Based on review of the onsite equipment rating, sizing criteria, existing loading, and margins, the electrical equipment powered by the onsite distribution system remains within their respective ratings. Thus, the onsite distribution system is not affected by the uprate.

3.7.3 Grid Stability

Southern California Edison (SCE) performs the grid system analysis. This analysis is reviewed by the California Independent System Operator (ISO) and is updated annually, as required. The grid system analysis was reviewed for a bounding uprate of 50 MW assuming a bounding gross generator output of 1200 MW. The review resulted in the conclusion that there is no impact on grid stability and reliability for a power uprate of 1.42% for both units. Additionally, the SONGS power uprate will not adversely impact the availability of the offsite power source for SONGS house loads in the event of a unit trip.

Based on the review of the current analysis, current grid reliability and stability are not impacted and SONGS continues to be in conformance with the General Design Criterion 17 for the power uprated electrical conditions.

4.0 UFSAR Chapter 15 Accident Analysis

4.1 Transients

This section presents the impact of power uprate on the UFSAR Chapter 15 Transient Analyses. UFSAR Chapter 15 events are discussed in the following paragraphs and in Table 4-2.

4.1.1 Trip Setpoints

The immediate impact of the power uprate on the accident analysis is seen on the initial power assumption for the accident analysis, and on the trip setpoints which are based on a percentage of the rated thermal power (RTP). The reactor trips that are based on a percentage of the RTP are

1) High Log Power Trip
2) High Linear Power Trip
3) Core Protection Calculator System (CPCS) Variable Overpower Trips (VOPT)
    a) CPCS VOPT Setpoint Variable Minimum Value (SPVMIN)
    b) CPCS VOPT Setpoint Variable Maximum Value (SPVMAX)

c) CPCS VOPT "Rate of Change"

Following is a brief discussion of the impact of power uprate on these trip setpoints. The impact of the power uprate on the dynamics of the transients is shown in Table 4-2.

4.1.1.1 High Log Power Trip

The High Log power trip is used as mitigating action against transients starting from a subcritical state (e.g., CEA Withdrawal (CEAW) from subcritical)(Table 4-2, UFSAR Section 15.4.1.1). The impact of power uprate on this trip is the increase in the analysis value of the trip setpoint from 4% of 3390 MWt to 4% of 3438 MWt. The CEAW from subcritical is explicitly performed with the larger trip setpoint in anticipation of the uprate. Therefore, the High Log Power Trip and the Technical Specifications remain unchanged.

4.1.1.2 High Linear Power Trip

The High Linear Power Trip is not explicitly credited in any of the accident analyses. The Technical Specification value of 110% remains unchanged.

4.1.1.3 CPCS VOPT SPVMIN Trip

The CPCS SPVMIN, the floor for the VOPT trip, is used as mitigating action against transients starting from a low power state (e.g., CEAW from Hot Zero Power (HZP))(Table 4-2, UFSAR Section 15.4.1.1). The impact of power uprate on this trip is the increase in trip setpoint from 30% of 3390 MWt to 30% of 3438 MWt. The consequences of the increase in the trip setpoint were analyzed. Margin in the analysis of record was sufficient to bound the changes in peak RCS pressure, peak heat flux, and peak linear heat rate. Therefore, CPCS VOPT SPVMIN Trip setpoint remains unchanged.

4.1.1.4 CPCS VOPT SPVMAX Trip

The CPCS VOPT SPVMAX is a high power trip setpoint. The impact of power uprate on this trip is the increase in trip setpoint from 110% of 3390 MWt to 110% of 3438 MWt. However, current transient analyses that credit this trip add the 2% power uncertainty to the trip setpoint. Therefore, the increase in the trip setpoint as a result of the power uprate has adequately been addressed in the transient analyses.

4.1.1.5 CPCS VOPT "Rate of Change" Trip

The CPCS VOPT trip is used in many of the accident analyses. The trip setpoint is set 10% above the initial power at the start of the transient. This trip moves at a prescribed rate as the transient progresses. The trip is limited to the range of SPVMIN to SPVMAX. The impact of power uprate on this trip is the increase in the offset from 10% of 3390 MWt to 10% of 3438 MWt. This

trip is used in either the CPCS Margin Setting events or the CPC Filter Verification events described below.

4.1.1.5.1 CPCS Margin Setting Events

The CPCS Margin Setting events are the events that determine the amount of thermal margin required in order for the transient to meet its acceptance criteria. These events are simulated using the transient analysis model to determine the final conditions of the event. 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. The required thermal margin is then preserved in COLSS during normal plant operation and CPCS when COLSS is out of service. Therefore, the CPCS VOPT "Rate of Change" trip setpoint remains unchanged.

4.1.1.5.2 CPCS Filter Verification

The CPCS Filter Verification analyses verify dynamic compensation filters for state parameters such as power and temperature. The verification is based on the comparison of the CPCS response to changes in state parameters versus the actual changes in the state parameters as predicted by transient analysis codes. The events are simulated to provide the maximum rate of change for state parameters since the maximum rate of change provides the greatest challenge to the CPCS filters. The VOPT trip is used to provide a reasonable transient duration for which the CPCS filter response is examined. The CPCS filter is verified when the CPCS state parameter response leads the actual state parameter response predicted by the transient analysis model. For these events the choice of initial and final power becomes insignificant. Verification of the CPCS filters assures the conservatism of the CPCS Low DNBR trip. Therefore, the CPCS VOPT "Rate of Change" trip setpoint remains unchanged.

4.1.2 Steam Generator Tube Plugging

The tube plugging assumptions used in the current accident analyses performed for SONGS Units 2 and 3 are based on a range of tubes plugged, from 0 tubes plugged (clean Steam Generator) up to 2000 tubes plugged per SG. This range bounds the current plant values ($\leq$765 for Unit 2 Cycle 11 and $\leq$586 for Unit 3 Cycle 11). The power uprate has no direct impact on the tube plugging assumptions used for the UFSAR Chapter 15 Analyses.

4.2 Radiological Consequences

The radiological consequences of the power uprate relative to events described in UFSAR Chapter 15 are discussed in the following paragraphs and in Table 4-2.

4.2.1 Moderate Frequency and Infrequent Events

In all cases, the moderate frequency events yield radiological consequences that are enveloped by another more severe event.

Many of the infrequent events yield radiological consequences that are enveloped by another more severe event. The following discusses the impact of power uprate on each infrequent event reporting dose consequences in UFSAR Chapter 15.

4.2.1.1 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Single Active Failure

In the absence of fuel failure, the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Single Active Failure (IOSGADV/SAF) (Table 4-2, UFSAR Section 15.1.2.4) is the most severe infrequent event. From a radiological standpoint, the IOSGADV/SAF also bounds all moderate frequency events. This event is evaluated assuming a primary reactor coolant system activity concentration profile at the Technical Specification LCO 3.4.16 limit of 1.0 micro Curie/gram Iodine-131 dose equivalency. Since the basic sequence of events for this accident analysis is unchanged for the power uprate, the LCO limit is unchanged.

4.2.1.2 Increased Main Steam Flow with a Single Active Failure

The Increased Main Steam Flow with a Single Active Failure (IMSF/SAF) is the most severe infrequent event (Table 4-2, UFSAR Section 15.1.2.3). The Units 2 and 3 IMSF/SAF dose analyses of record models a fuel rod activity inventory profile (i.e., the non-LOCA source term). The fuel rod activity profile for 3458 MWt was compared to the profile of the analyses of record. The modeled activity inventories of some fuel rod isotopes (primarily iodine isotopes) are greater than, and the activity inventories of other fuel rod isotopes (mainly noble gas isotopes) are less than, the fuel rod activity inventories associated with 3458 MWt. An evaluation of the thyroid and whole body gamma dose contributions of iodine and noble gas core isotopes has determined that the non-LOCA source term modeled in the dose analyses of record bounds (i.e., is equal to, or more severe than) any non-LOCA source term for a reactor core power less than 3458 MWt (e.g., 3438 MWt). Therefore, all radiological consequences continue to meet acceptance criteria.

4.2.2 Limiting Faults

Many of the limiting faults have radiological consequences that are enveloped by another more severe event. The following discusses the impact of power uprate on each of the limiting faults stated to have dose consequences in UFSAR Chapter 15.

4.2.2.1 Events with No Fuel Failure

In the absence of fuel failure, the Post-Trip Return-to-Power Steam Line Break, the Primary Sample or Instrument Line Break, the Steam Generator Tube Rupture, and the Radioactive Waste Gas System Leak or Failure events are evaluated assuming primary reactor coolant system activity concentration profiles at Technical Specification LCO 3.4.16 limits or greater, considering cases with and without iodine spiking. Since these accident analyses are unchanged for the power uprate, the LCO is unchanged and all radiological criteria continue to be met (Table 4-2, UFSAR Sections 15.1.3.1B, 15.6.3.1, 15.6.3.2, and 15.7.3.1).

4.2.2.2 Pre-Trip Steam Line Break

The pre-trip Steam Line Break (pre-trip SLB) event is characterized by fuel failure (Table 4-2, UFSAR Section 15.1.3.1A). The Units 2 and 3 pre-trip SLB dose analyses of record model a fuel rod activity inventory profile (i.e., the non-LOCA source term). As in the case of the IMSF/SAF event, an evaluation of the thyroid and whole body gamma dose contributions of iodine and noble gas core isotopes has determined that the non-LOCA source term modeled in the dose analyses of record bounds (i.e., is equal to, or more severe than) any non-LOCA source term for a reactor core power less than 3458 MWt (e.g., 3438 MWt). Therefore, all radiological consequences are unchanged from the analyses of record.

4.2.2.3 Reactor Coolant Pump Sheared Shaft

The Reactor Coolant Pump Sheared Shaft (RCP SS) event is characterized by fuel failure (Table 4-2, UFSAR Section 15.3.3.2). The Units 2 and 3 RCP SS dose analyses of record model a fuel rod activity inventory profile (i.e., the non-LOCA source term). As in the case of the IMSF/SAF event, an evaluation of the thyroid and whole body gamma dose contributions of iodine and noble gas core isotopes has determined that the non-LOCA source term modeled in the dose analyses of record bounds (i.e., is equal to, or more severe than) any non-LOCA source term for a reactor core power less than 3458 MWt (e.g., 3438 MWt). Therefore, all radiological consequences are unchanged from the analyses of record.

4.2.2.4 Control Element Assembly Ejection

The Control Element Assembly Ejection (CEA-ej) Event is characterized by fuel failure (Table 4-2, UFSAR Section 15.4.3.2). The source term used in the Units 2 and 3 CEA-ej dose analysis is based on the Units 2 and 3 Cycle 1 core activity inventory profile. The activity inventories of some Cycle 1 core isotopes (primarily some iodine and noble gas isotopes) are greater than the core activity inventories associated with 3458 MWt. An evaluation of the noble gas dose contributions of iodine and noble gas core isotopes has determined that the Cycle 1 source term modeled in UFSAR Chapter 15 CEA-ej dose analysis of record bounds (i.e., is equal to, or more severe than) any non-LOCA source term for a reactor core power less than 3458 MWt (e.g., 3438 MWt). A similar evaluation of the thyroid dose contributions of iodine core isotopes has determined that the

Cycle 1 source term modeled in UFSAR Chapter 15 CEA-ej dose analysis of record is approximately one percent less severe than the 3458 MWt non-LOCA source term. However, the CEA-ej dose analysis of record is excessively conservative in its evaluation of thyroid dose due to its modeling of Regulatory Guide 1.109 Revision 0 thyroid inhalation Dose Conversion Factors (DCF) rather than the International Commission on Radiological Protection (ICRP) Publication 30 DCFs which are approved for use at SONGS (reference 8.12). Use of the ICRP-30 DCFs would reduce the thyroid dose by almost 30 percent, which is significantly greater than the one percent non-conservatism in the source term as it relates to thyroid dose. Therefore, all radiological consequences continue to meet acceptance criteria.

4.2.2.5 Loss of Coolant Accident

The large break Loss of Coolant Accident (LOCA) event is characterized by fuel failure (Table 4-2, UFSAR Section 15.6.3.3). The source term used in the Units 2 and 3 LOCA dose analyses of record is based on the Units 2 and 3 Cycle 9 core activity inventory profile. The activity inventories of some Cycle 9 core isotopes are greater than, and the activity inventories of other Cycle 9 core isotopes are less than, the core activity inventories associated with a reactor power of 3458 MWt. An evaluation of the thyroid, whole body gamma, and beta-skin dose contributions of iodine, noble gas, and particulate core isotopes has determined that the Cycle 9 LOCA source term modeled in the UFSAR Chapter 15 LOCA dose analyses of record would not bound the 3458 MWt LOCA source term. Based on this determination, new LOCA dose analyses modeling the 3458 MWt source term were performed. The analyses show an insignificant increase (up to 1.5 Rem) in the exclusion area boundary (EAB), low population zone (LPZ), and control room (CR) thyroid inhalation doses, and no more than a slight increase (up to 0.2 Rem) in the EAB, LPZ, and CR whole body (WB) gamma and beta-skin doses (see Table 4-1). The revised doses continue to meet the offsite dose criteria of 10 CFR 100 and the control room dose criteria of 10 CFR 50 Appendix A General Design Criterion 19.

4.2.2.6 Fuel Handling Accidents

The Fuel Handling Accident occurring inside the Fuel Handling Building (FHA-FHB), the Fuel Handling Accident occurring inside the Containment Building (FHA-IC), and the Spent Fuel Pool Gate Drop Accident (GDA) events are all characterized by fuel failure (Table 4-2, UFSAR Sections 15.7.3.4, 15.7.3.9, and 15.7.3.6 respectively). The source term used in the Units 2 and 3 FHA-FHB, FHA-IC, and GDA dose analyses of record is based on a fuel rod activity inventory profile decayed for 72 hours, which is the earliest time fuel handling operations may begin (i.e., the FHA source term). The modeled FHA source term was scaled up by two percent to generate the fuel rod activity inventories defining a reactor power of 3458 MWt. As such, the dose analysis of record FHA source term does not bound the 3458 MWt FHA source term. However, the existing dose analyses of record model conservative pairings of radial peaking factors and iodine fuel rod gap release fractions (e.g., high burnup 12 percent release fraction for all iodine isotopes, with a 1.75 radial peaking factor) which result in excessively conservative dose consequences. Analysis has shown that removal of this excess conservatism would reduce the dose consequences by an

amount significantly greater than the two percent non-conservatism in the modeled source term.

Table 4-1 LOCA Event Dose Changes Due to Power Uprate

DOSE RECEPTOR	DESIGN BASIS DOSE (Rem)	POWER UPRATE DOSE (Rem)
EAB - Thyroid Dose	73.5	75.0
EAB - WB Gamma Dose	1.5	1.4
EAB - Beta-Skin Dose	0.7	0.7
LPZ - Thyroid Dose	34.3	35.2
LPZ - WB Gamma Dose	0.2	0.2
LPZ - Beta-Skin Dose	0.1	0.1
CR - Thyroid Dose	25.1	25.7
CR - WB Gamma Dose	3.4	3.6
CR - Beta-Skin Dose	23.0	22.9

4.2.2.7 Spent Fuel Pool Boiling

The Spent Fuel Pool Boiling (SFP boiling) event is not characterized by fuel failure (Table 4-2, UFSAR Section 15.7.3.8). The source term used in the Units 2 and 3 SFP boiling dose analyses of record is based on the same FHA source term used in the Units 2 and 3 FHA-FHB, FHA-IC, and GDA dose analyses of record. As in those events, the modeled FHA source term does not bound the 3458 MWt FHA source term. The SFP boiling dose analysis of record also models an increased SFP heat load based on 3444 MWt and an increased initial SFP water temperature which both increase as a consequence of the power uprate. Based on these issues, a revised SFP boiling dose analysis modeling the 3458 MWt power source term and related increased heat load and SFP water temperature was performed. The analysis shows no increase in the reported offsite thyroid, whole body gamma, and beta-skin doses. This is due to the differences in heat-up causing a change in the timing when the iodine spiking occurs. The doses continue to meet the offsite dose criteria of 10 CFR 100. The analysis shows a slight reduction in the elapsed time to SFP boiling, of less than 0.2 hours.

4.2.3 Other Radiological Consequences

4.2.3.1 Equipment Qualification Dose Analyses

The UFSAR Section 3.11 Equipment Qualification (EQ) Dose Analyses address event duration doses to safety related electrical equipment primarily in the event of a large break LOCA. For equipment present in the Fuel Handling Building, the events of interest include the fuel handling accident (FHA in FHB) and the spent fuel pool gate drop accident (GDA).

The source term used in the EQ LOCA dose analyses of record is based on the Units 2 and 3 Cycle 1 core activity inventory profile. The activity inventories of some Cycle 1 core isotopes (primarily some iodine and noble gas isotopes) are greater than the core activity inventories associated with a reactor power of 3458 MWt. An evaluation of the dose contributions of iodine, noble gas, and particulate core isotopes has determined that the Cycle 1 LOCA source term modeled in UFSAR Section 3.11 EQ dose analyses of record bound (i.e., are equal to, or more severe than) any LOCA source term for a reactor core power less than 3458 MWt (e.g., 3438 MWt).

The source term used in the EQ FHA-FHB and EQ GDA dose analyses of record is based on the Units 2 and 3 Cycle 1 fuel rod activity inventory profile decayed for 72 hours, which is the earliest time fuel handling operations may begin (i.e., the FHA source term), with isotopic inventory scaled to reflect the increased inventory associated with extended burnup to 60 gigawatt days per metric ton uranium. As in the EQ LOCA Dose Analyses, the activity inventories of some Cycle 1 core isotopes (primarily some iodine and noble gas isotopes) are greater than the core activity inventories associated with the 102% power after 72 hours of decay. An evaluation of the dose contributions of iodine and noble gas fuel rod isotopes has determined that the Cycle 1 FHA fuel rod gap activity release profile modeled in UFSAR Section 3.11 EQ dose analyses of record bound (i.e., are equal to, or more severe than) any FHA fuel rod gap activity release source term for a reactor core power less than 3458 MWt (e.g., 3438 MWt).

4.2.3.2 Radioactive Waste Management Analyses and Radiation Protection Analyses

The UFSAR Chapter 11, Radioactive Waste (Radwaste) Management, addresses system/component activity inventories and activity releases associated with the liquid, gaseous, and solid waste management systems, as well as the process and effluent radiological monitoring and sampling systems. The UFSAR Chapter 12 Radiation Protection Analyses address As Low As Reasonably Achievable (ALARA) compliance, radiation sources, radiation protection design features (e.g., plant shielding), and dose assessment.

The radwaste management and radiation protection analyses are based on core and system activity profiles featuring the Units 2 and 3 Cycle 1 core activity inventory profile with one percent fuel cladding defects. The activity inventories of some Cycle 1 core isotopes (primarily some iodine

and noble gas isotopes) are greater than the core activity inventories associated with a reactor power of 3458 MWt. Based on the evaluation of the dose contributions of iodine, noble gas, and particulate core isotopes as previously discussed in the EQ evaluation, the Cycle 1 one percent fuel failure core and system activity profiles modeled in UFSAR Chapter 11 bound (i.e., are equal to, or more severe than) the 3438 MWt one percent fuel failure core and system activity source terms for a reactor core power less than 3458 MWt (e.g., 3438 MWt).

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1 Increase in Heat Removal by the Secondary System			
15.1.1.1	Decrease in Feedwater Temperature	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Increased Main Steam Flow (Section 15.1.1.3).
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	
15.1.1.2	Increase in Feedwater Flow	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Increased Main Steam Flow (Section 15.1.1.3).
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	
15.1.1.3	Increased Main Steam Flow	Peak RCS Pressure ≤ 110% of Design	Peak Pressure criteria are not challenged for this event. CPCS filters, see Section 4.1.1.5.2, are set to ensure DNBR trip to preclude fuel failure. The filter verification is impacted by the rate of change of Tcold and is not impacted by the power uprate. Therefore, the power uprate has no adverse impact on all criteria for this event.
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.1.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve	Peak RCS Pressure ≤ 110% of Design	Peak Pressure and Fuel Performance criteria are bounded by Increased Main Steam Flow (Section 15.1.1.3). The most adverse offsite dose consequence for this event occurs at Hot Zero Power (HZP) and there is no trip credited for this event. Therefore, the power uprate has no impact on any of the acceptance criteria. The radiological consequences are bounded by the IOSGADV with Single Active Failure (SF) (Section 15.1.2.4).
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	
		Offsite Dose ≤ 0.5 Rem whole body	
15.1.2.1	Decrease in Feedwater Temperature With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Increased Main Steam Flow with Single Active Failure (Section 15.1.2.3).
		Peak Secondary Pressure ≤ 110% of Design	
		Maintain coolable geometry	
		Offsite Doses well within 10CFR100 guidelines	
15.1.2.2	Increase in Feedwater Flow With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Increased Main Steam Flow with Single Active Failure (Section 15.1.2.3).
		Peak Secondary Pressure ≤ 110% of Design	
		Maintain coolable geometry	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.2.3	Increased Main Steam Flow With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Peak Pressure criteria are not challenged for this event. A combination of Preserved DNBR margin (Section 4.1.1.5.1) and the CPCS filter settings minimize fuel failures. The filter verification is impacted by the rate of change of Tcold and not the initial power, see Section 4.1.1.5.2, and is thus not adversely impacted by power uprate. The trips credited for this event are based on the relative change of power and CPCS low DNBR trip. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		Maintain coolable geometry	
		Offsite Doses well within 10CFR100 guidelines	As discussed in Section 4.2.1.2, all radiological consequences continue to meet the acceptance criteria.
15.1.2.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Peak Pressure and Fuel Performance criteria are bounded by Increased Main Steam Flow with Single Active Failure (Section 15.1.2.3). The most adverse offsite dose consequence for this event occurs at HZP and there is no trip credited for this event. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		Maintain coolable geometry	As discussed in Section 4.2.1.1, all radiological consequences continue to meet the acceptance criteria.
		Offsite Doses well within 10CFR100 guidelines	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.1.3.1A	Steam System Piping Failures (Pre-trip power excursion)	Maintain coolable geometry Offsite Doses within 10CFR100 guidelines.	A combination of preserved DNBR margin (Section4.1.1.5.1) and the reactor trips are set to minimize fuel failures. The reactor trips are based on the Reactor Protection System (RPS) Low Steam Generator (SG) pressure trip and the VOPT. The Low SG pressure trip is not impacted by the power uprate. The CPCS VOPT trip is based on the relative change of power not the initial power value (Section 4.1.1.5). Therefore, the power uprate has no impact on any of the acceptance criteria. As discussed in Section 4.2.2.2, all radiological consequences continue to meet the acceptance criteria.
15.1.3.1B	Steam System Piping Failures (Post - trip return to power)	Maintain coolable geometry Offsite Doses a small fraction of 10CFR100 guidelines (with no iodine spike). Offsite Doses within 10CFR100 guidelines (with pre-existing iodine spike).	The most adverse consequence for this event occurs at HZP. The RPS trip is based on the Low SG Pressure trip which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria. As discussed in Section 4.2.2.1, all radiological consequences continue to meet the acceptance criteria.

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.2 Decrease in Heat Removal by the Secondary System			
15.2.1.1	Loss of External Load	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum (Section 15.2.1.3).
		Peak Secondary Pressure ≤ 110% of Design	
15.2.1.2	Turbine Trip	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum (Section 15.2.1.3).
		Peak Secondary Pressure ≤ 110% of Design	
15.2.1.3	Loss of Condenser Vacuum (LOCV)	Peak RCS Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the High Pressurizer Pressure Trip (HPPT), which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria. The MSSV Inoperable Analysis (LCO 3.7.1 and LCO Table 3.7.1-1) was also performed based on a 3458 MWt power definition as an initial condition (e.g., 98.6% of 3458 MWt).
		Peak Secondary pressure ≤ 110% of Design	
15.2.1.4	Loss of Normal AC Power	Peak RCS Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the CPCS Low Pump Speed Trip, which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	
		Adequate SG inventory to maintain adequate heat sink	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES

FSAR SECTION	TITLE	ACCEPTANCE CRITERIA	IMPACT OF POWER UPRATE
15.2.2.1	Loss of External Load with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum with Single Active Failure (Section 15.2.2.3).
		Peak Secondary Pressure ≤ 110% of Design	
15.2.2.2	Turbine Trip with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all criteria are bounded by Loss of Condenser Vacuum with Single Active Failure (Section 15.2.2.3).
		Peak Secondary Pressure ≤ 110% of Design	
15.2.2.3	Loss of Condenser Vacuum with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the High Pressurizer Pressure Trip (HPPT), which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary pressure ≤ 110% of Design	
15.2.2.4	Loss of all Normal AC Power with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the CPCS Low Pump Speed Trip, which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		Adequate SG inventory to maintain adequate heat sink	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.2.2.5	Loss of Normal Feedwater Flow	Peak RCS Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the Low SG Level Trip (LSGLT), which is not impacted by the power uprate. The initial SG level is the maximum SG level which is not impacted by the uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		Adequate SG inventory to maintain adequate heat sink	
15.2.3.1	Feedwater System Pipe Breaks	Peak RCS Pressure ≤ 120% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the High Pressurizer Pressure Trip (HPPT), which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		No Liquid release through the PSV for peak RCS pressure case	
		Adequate SG inventory to maintain adequate heat sink	
15.2.3.2	Loss of Normal Feedwater Flow with a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	The analysis is performed at 102% power. Furthermore, the mitigating action is the Low SG Level Trip (LSGLT), which is not impacted by the power uprate. The initial SG level is the maximum SG level which is not impacted by the uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		Adequate SG inventory to maintain adequate heat sink	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.3 Decrease in Reactor Coolant Flow Rate			
15.3.1.1	Partial Loss of Forced Reactor Coolant Flow	Peak RCS Pressure ≤ 110% of Design	The Partial Loss of Forced Flow was not analyzed because it is bounded by the Total Loss of Flow (Section 15.3.2.1).
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	
15.3.2.1	Total Loss of Forced Reactor Coolant Flow	Peak RCS Pressure ≤ 110% of Design	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. Furthermore, the mitigating action is the CPCS Low Pump Speed Trip, which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Peak Secondary Pressure ≤ 110% of Design	
		No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	
15.3.2.2	Partial Loss of Forced Reactor Coolant Flow with Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	Not analyzed since all acceptance criteria are bounded by the Single Reactor Coolant Pump Sheared Shaft event (Section 15.3.3.2).
		Maintain Coolable Geometry	
		Peak Secondary Pressure ≤ 110% of Design	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.3.3.1	Single Reactor Coolant Pump Shaft Seizure	Peak RCS Pressure ≤ 110% of Design	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 mitigating action is the CPCS Low Pump Speed trip which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria. The radiological consequences are bounded by Single Reactor Coolant Pump Sheared Shaft (Section 15.3.3.2).
		Peak Secondary Pressure ≤ 110% of Design	
		Maintain Coolable Geometry	
		Offsite Doses a small fraction of 10CFR100 guidelines	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.3.3.2	Single Reactor Coolant Pump Sheared Shaft	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain Coolable Geometry Offsite Doses a small fraction of 10CFR100 guidelines	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 mitigating action is the PPS Low Flow trip which is not impacted by the power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria. As discussed in Section 4.2.2.3, all radiological consequences continue to meet the acceptance criteria.
15.3.3.3	Total Loss of Forced Reactor Coolant Flow with Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design Peak Secondary Pressure ≤ 110% of Design Maintain Coolable Geometry	Not analyzed since all acceptance criteria are bounded by Single Reactor Coolant Pump Sheared Shaft (Section 15.3.3.2).
15.4 Reactivity and Power Distribution Anomalies			
15.4.1.1	Uncontrolled CEA Withdrawal at Subcritical and Low Power	Peak RCS Pressure ≤ 110% of Design No Fuel Failure (Minimum DNBR ≥1.31 and Peak LHR ≤ 21 kw/ft)	The analysis is performed at subcritical and low power. The mitigating actions are the CPCS VOPT SPVMIN (Section 4.1.1.3) and the High Log Power Trips (Section 4.1.1.1) for the low power and subcritical conditions, respectively. The impact of power uprate was evaluated and margin in the analysis of record was sufficient to bound the change in peak heat flux, peak linear heat, and peak pressure. Therefore, the power uprate has no impact on any of the acceptance criteria.

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. Therefore, 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. 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.

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.4.3.2	Control Element Assembly (CEA) Ejection	Peak RCS Pressure ≤ 110% of Design	This analysis was performed at 102% power. Due to the fast nature of the event, the timing of the trip is insensitive to the apparent 1.42% increase in the trip setpoint. Therefore, the power uprate has no impact on any of the acceptance criteria. As discussed in Section 4.2.2.4, all radiological consequences continue to meet the acceptance criteria.
		Centerline enthalpy of hottest fuel pellet ≤ 280 cal/gm (SONGS fuel failure threshold: total average enthalpy of hottest fuel pellet ≤ 200 Cal/gm, total centerline enthalpy of hottest fuel pellet ≤ 250 Cal/gm)	
		Offsite Doses within 10CFR100 guidelines	
15.5 Increase in Reactor Coolant Inventory			
15.5.1.1	Chemical and Volume Control System Malfunction	Peak RCS Pressure ≤ 110% of Design	The transient was performed at 102% power. The mitigation action was a High Pressurizer Pressure Trip (HPPT), which is not affected by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		No liquid flow through PSVs for peak RCS pressure case	
15.5.1.2	Inadvertent Operation of the ECCS During Power Operation	Peak RCS Pressure ≤ 110% of Design	Not analyzed since this event is bounded by CVCS malfunction (Section 15.5.1.1).
		No liquid flow through PSVs for peak RCS pressure case	
15.5.2.1	Chemical and Volume Control System Malfunction With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design	The transient was performed at 102% power. The mitigation action was a HPPT, which is not affected by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria.
		No liquid flow through PSVs for peak RCS pressure case	

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.5.2.2	Inadvertent Operation of the ECCS During Power Operation With a Concurrent Single Failure of an Active Component	Peak RCS Pressure ≤ 110% of Design No liquid flow through PSVs for peak RCS pressure	Not analyzed since this event is bounded by CVCS malfunction with concurrent Single Failure (SF) of an active component (Section 15.5.2.1).
15.6 Decrease in Reactor Coolant Inventory			
15.6.3.1	Primary Sample or Instrument Line Break	Offsite Doses a small fraction of 10CFR100 guidelines	The transient was performed at 102% power. There is no credible trip in the first 30 minutes. Therefore, the power uprate has no impact on any of the acceptance criteria. As discussed in Section 4.2.2.1, all radiological consequences continue to meet the acceptance criteria.
15.6.3.2	Steam Generator Tube Rupture	Offsite Doses a small fraction of 10CFR100 guidelines (with no iodine spike). Offsite Doses within 10CFR100 guidelines (with pre-existing iodine spike). Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	The transient was performed at 102% power. The mitigation action was a CPCS Auxiliary trip (e.g., Pressurizer Pressure Range), which is not affected by power uprate. Therefore, the power uprate has no impact on any of the acceptance criteria. As discussed in Section 4.2.2.1, all radiological consequences continue to meet the acceptance criteria.

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.6.3.3	LOCA	10CFR50.46	The analysis was performed at 102% power. Therefore, the power uprate has no impact on any of the acceptance criteria.
		Offsite Doses within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	As discussed in Section 4.2.2.5, the radiological source term modeled in the analysis of record for this event does not bound the power uprate source term. The power uprate results in a slight increase in the offsite and control room dose consequences of this event. The revised doses continue to meet the offsite dose criteria of 10 CFR 100 and the control room dose criteria of 10 CFR 50 Appendix A GDC 19.
15.6.3.4	Inadvertent Opening of a Pressurizer Safety Valve	Bounded by LOCA analysis	Not analyzed since this event is bounded by LOCA.
15.7 Radioactive Release from a Subsystem or Component			
15.7.3.1	Radioactive Waste Gas System Leak or Failure	Offsite Dose ≤ 0.5 Rem whole body	As discussed in Section 4.2.2.1, all radiological consequences continue to meet the acceptance criteria.
15.7.3.2	Radioactive Waste System Leak or Failure (Release to Atmosphere)	Offsite Dose ≤ 0.5 Rem whole body Thyroid Dose Inhalation ≤ 1.5 Rem	As discussed in Section 4.2.2.1, all radiological consequences continue to meet the acceptance criteria.

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.7.3.3	Postulated Radioactive Releases due to Liquid Tank Failures	Offsite Dose ≤ 0.5 Rem whole body	Not analyzed since this event has no dose consequences.
15.7.3.4	Design Basis Fuel Handling Accident Inside Fuel Building	Offsite Doses well within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	As discussed in Section 4.2.2.6, all radiological consequences continue to meet the acceptance criteria.
15.7.3.5.1	Spent Fuel Cask Drop into Spent Fuel Pool	Offsite Doses well within 10CFR100 guidelines.	Not analyzed since this event has no dose consequences.
15.7.3.5.2	Spent Fuel Cask Drop to Flat Surface	Offsite Doses well within 10CFR100 guidelines.	Not analyzed since this event has no dose consequences.
15.7.3.6	Spent Fuel Pool Gate Drop Accident	Offsite Doses well within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	As discussed in Section 4.2.2.6, all radiological consequences continue to meet the acceptance criteria.
15.7.3.7	Test Equipment Drop	Offsite Doses well within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	Not analyzed since this event has no dose consequences.

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.7.3.8	Spent Fuel Pool Boiling Accident	Offsite Doses well within 10CFR100 guidelines.	As discussed in Section 4.2.2.7, the radiological source term modeled in the analysis of record for this event does not bound the power uprate source term. However, the power uprate results in no change in the UFSAR reported dose consequences of this event. The analysis shows a slight reduction in the elapsed time to SFP boiling.
15.7.3.9	Design Basis Fuel Handling Accident Inside Containment	Offsite Doses well within 10CFR100 guidelines. Control Room Doses within 10CFR100 Appendix A GDC 19 guidelines.	As discussed in Section 4.2.2.6, all radiological consequences continue to meet the acceptance criteria.
15.7.3.10.1	Spent Fuel Assembly Drop onto Reconstitution Station	Offsite Doses well within 10CFR100 guidelines.	Not analyzed since this event has no dose consequences.
15.7.3.10.2	Spent Fuel Assembly Drop onto CEA Bearing Spent Fuel Assemblies	Offsite Doses well within 10CFR100 guidelines.	Not analyzed since this event has no dose consequences.
15.7.3.11	Use of Miscellaneous Equipment Under 2000 lbs	Offsite Doses well within 10CFR100 guidelines.	Not analyzed since this event has no dose consequences.
15.8 Anticipated Transient Without Scram (ATWS)			
15.8.1	ATWS	10 CFR 50.62	SONGS has installed a Diverse Scram System (DSS) and Diverse Emergency Feedwater Actuation System (DEFAS) as a response to the generic ATWS report. The power uprate does not impact any of the setpoints used in the DSS. Therefore, the power uprate has no impact on this event.

TABLE 4-2 - IMPACT OF POWER UPRATE ON THE UFSAR CHAPTER 15 ACCIDENT ANALYSES			
FSAR SECTION	**TITLE**	**ACCEPTANCE CRITERIA**	**IMPACT OF POWER UPRATE**
15.9 Miscellaneous			
15.9.1.1	Asymmetric Steam Generator Transient (ASGT)	No Fuel Failure (Minimum DNBR $\geq$1.31 and Peak LHR $\leq$ 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 mitigation action was a CPCS Auxiliary trip (e.g CPCS ΔT trip) which is not impacted by power uprate (Section 4.1.1.5.2). Therefore, the power uprate has no impact on any of the acceptance criteria.

5.0 Miscellaneous

5.1 Station Blackout (SBO) Analysis

SCE submitted a response to 10CFR50.63, Station Blackout, on September 12, 1991 (reference 8.13). As part of that response, SCE committed to follow the criteria for satisfactory performance as outlined in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors."

The general criteria states that procedures and equipment relied upon in a station blackout should ensure that satisfactory performance of necessary decay heat removal systems is maintained for the required 4 hour coping duration. The core must remain covered and containment integrity should be provided to the extent that isolation valves perform their intended functions without AC power.

Therefore, an element of concern for this uprate from 100% to 101.42% (3390 to 3438 MWt) for SBO is a slight increase in decay heat generation (slightly higher cooling load during cooldown). Containment pressure and temperature profiles will continue to be bounded by the LOCA profiles which were generated at 3458 MWt (102% of 3390 MWt).

The proposed operating conditions are bounded by the existing plant analysis. The water inventory and the ability to keep the core covered is not a concern because the volume of borated water in the boric acid tanks and RWSTs is 10 times that needed for shrinkage from normal operating temperature to cold shutdown.

The total condensate inventory is based on RSB 5-1 sizing requirements. Combustion Engineering calculation S-PEC-221 (reference 8.14) assumes an initial RTP condition of 102% power (100% power + 2% uncertainty in core thermal power). Also ten full power seconds of additional heat load is assumed. The calculation provides for 24 hours of steaming with one ADV and loss of offsite power. This 24 hours of operation includes 4 hours of hot standby, cooldown to 400°F (the maximum cut-in conditions for the shutdown cooling system), and continued decay heat removal until 24 hours of steaming is attained. An assessment was performed of other criteria such as system leakage to determine a minimum required storage inventory to assure 24 hours of operation in accordance with RSB 5-1.

The Atmospheric Dump Valves were designed to provide a means of decay heat removal and plant cooldown during loss of condenser vacuum from a steady state power of 100 % RTP + 2 % instrument uncertainty. This design bounds the power uprate.

Other elements of the SBO analysis have not significantly changed: Plant Lighting, RCS Inventory Loss, Shutdown Margin, Containment Isolation, Loss of Ventilation, Compressed Air, Battery Capacity, Coping Period, Diesel Generator Reliability, or equipment required operable for Station Blackout. None of the associated instruments require control setpoint changes, and none of the associated instruments exceed design basis due to the power uprate. Therefore, the SBO analysis

is not affected by this power uprate.

5.2 Integrated Plant Evaluation

Although this is not a risk informed submittal, SCE has reviewed the impact of the proposed power uprate on the overall Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) results. The results of that evaluation indicate that the power uprate would not have a measurable impact on the IPE and IPEEE results. The evaluation included a comprehensive review of: (1) accident mitigating system success criteria (e.g., number of emergency core cooling flow paths required to mitigate loss of coolant accidents, adequacy of auxiliary feedwater bypass valves to provide sufficient flow to prevent steam generator dryout in all events); (2) human reliability analyses (e.g., time available to establish emergency condensate to avoid steam generator dryout with unavailability of auxiliary feedwater, time to cross-tie emergency diesel generators from the other unit to prevent steam generator dryout in the event of multiple diesel generator and auxiliary feedwater pump unavailabilities); and (3) core damage progression timing leading to a containment challenge.

5.3 Operations Impact

5.3.1 Control Room

Control Room alarms will be affected due to the installation of the CROSSFLOW system. These alarms will be addressed in the appropriate operating procedures per SONGS standard design change implementation procedures and deal exclusively with actions to take in accordance with problems with the CROSSFLOW measuring system. The operator displays, instrumentation, and control features in the Control Room will remain unchanged as a result of the installation of the CROSSFLOW system.

Displays for power will show 100% power for the new 3438 MWt power level. Other plant operating parameters will have minor changes. Those parameters determined to be outside of the existing zone markings on the control board indicators will be addressed as a part of the normal SONGS "Green Band" process for determining appropriate indicator operating band zones.

The SONGS Safety Parameter Display System is composed of the Critical Function Monitoring System (CFMS) and the Qualified Safety Parameter Display System (QSPDS). The 1.42% power uprate will have negligible effects on the associated parameter displays of these two systems. All points will remain within existing ranges, and the new affected operating values, including RCS temperature, SG pressures, and associated flows will be addressed within applicable operating procedures.

5.3.2 Emergency Operating Instructions

The power uprate will not change the type and scope of plant emergency and abnormal operating procedures. The procedures are based on design analysis conditions for accident scenarios, and those conditions do not change with this power uprate. A review of procedures found no additional scope or change in type or scope of abnormal operating actions based upon this power uprate. The plant response and subsequent operator actions will remain unchanged.

5.3.3 Operator Training and Simulator

SONGS will provide classroom and simulator training on all changes that affect operator performance caused by the power uprate modification. SONGS will complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." All control room and plant process computer system changes as a result of the power uprate will be completed. Additionally, operator training and the plant simulator will be modified, as required, to address all related issues and discrepancies that are identified during the startup testing program.

6.0 Environmental Evaluation

An environmental impact analysis was performed to evaluate the effects of the increased rated thermal power on the environment and on the existing plant environmental analysis and to identify actions that might be required to permit increasing power level under existing licenses, permits, and agreements (reference 8.15).

The units are cooled by once-through cooling water systems, withdrawing cooling water from the Pacific Ocean and discharging it to the ocean through separate underwater diffusers on the ocean bottom. The differential temperature developed by the cooling system will increase by approximately 0.3°F, increasing the calculated differential to approximately 19.2°F (reference 8.16). The limit on differential temperature is 25°F and includes an allowance of 0.4°F for increases in thermal power level (reference 8.16).

Other environmental discharges, for instance air quality due to operating diesel generators, will not be increased because the small increase in reactor power will not affect the operation or surveillance of the diesel generators.

SCE has evaluated the environmental impact of operation at a power level of 3438 MWt and concluded that no undue environmental effects are expected.

7.0 Significant Hazards Analysis

San Onofre Nuclear Generating Station (SONGS) has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10CFR50.92(c). The following information is provided to address the three questions required for the 10 CFR 50.92 evaluation.

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   The comprehensive analytical efforts performed to support the proposed change included a review of the Nuclear Steam Supply System (NSSS) systems and components that could be affected by this change. All systems and components will function as designed, and the applicable performance requirements have been evaluated and found to be acceptable.

   The primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The Leak Before Break analysis conclusions remain valid, and thus the limiting break sizes determined in this analysis remain bounding. All of the NSSS will still perform the intended design functions during normal and accident conditions. The auxiliary systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The main steam safety valves (MSSVs) will provide adequate relief capacity to maintain the steam generator pressures within design limits. The atmospheric dump valves and steam bypass valves meet design sizing requirements at the uprated power level. The current Loss of Coolant Accident (LOCA) hydraulic forcing functions are still bounding for the proposed 1.42% increase in power.

   Because the integrity of the plant will not be affected by operation at the uprated condition, it is concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. 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 either bound operation at the 1.42% uprated condition, or new analyses were performed to verify all acceptance criteria continue to be met.

Based on the foregoing, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. The new installation of the CROSSFLOW system has been analyzed, and failures of this system will have no effect on any safety related system or any systems, structures or components required for transient mitigation. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria. Therefore, it is concluded that the proposed changes do not involve a significant reduction in a margin of safety.

Based on the responses to these three criteria, Southern California Edison (SCE) has concluded that the proposed amendments involve no significant hazards consideration.

Environmental Consideration:

SCE has determined that the proposed amendments involve no changes in the amount or type of effluent that may be released offsite, and result in no increase in individual or cumulative occupational radiation exposure. As described above, the proposed License amendments involve no significant hazards consideration and, as such, meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10).

8.0 References

8.1 10CFR50 Appendix K - Emergency Core Cooling System Evaluation Models, effective date: July 31, 2000.

8.2 Letter from Stuart A. Richards, NRC, to Ian C. Rickard, A.B. Combustion Engineering dated March 20, 2000. Subject: Acceptance for Referencing of CENPD-397-P, Revision-01-P, "Improved Flow Measurement Accuracy using CROSSFLOW Ultrasonic Flow Measurement Technology" (TAC No. MA6452).

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

8.4 Southern California Edison Company Amendment Applications 179 and 165 for San Onofre Units 2 and 3 RCS Temperature Reduction, dated June 19, 1998, as supplemented by letters dated December 4, 1998, and January 13, 1999.

8.5 Topical Report CENPD-397-P-A, Revision 1, Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology, May, 2000.

8.6 Letter from M. B. Fields (NRC) to H. B. Ray (SCE), dated April 11, 1996; Subject: "Application of Leak-Before-Break Technology to Reactor Coolant System piping at San Onofre Nuclear Generating Station, Units 2 and 3 (TAC Nos. M92949 and M92950)."

8.7 NEI 97-06, "Steam Generator Program Guidelines Revision 1," dated January 2001.

8.8 Addendum to CENC 1272 and 1298, Analytical Reports for SONGS 2&3 Steam Generators, dated 4/21/99 (SONGS CDM number SO23-915-C220; Vendor number DR-SONGS-9449-1207)

8.9 Combustion Engineering Analytical Report CENC-1297 and addendum, (SO23-915-112) for the Model 3410 Steam Generators.

8.10 Combustion Engineering Calculation No. 2007701-CEAE-602, Rev 0 (SO23-915-212-0).

8.11 ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition up to Summer 1974 Addenda.

8.12 Letter from Dwight E. Nunn (SCE) to Document Control Desk (NRC), dated May 7, 1998. Subject: Docket Nos. 50-361 and 50-362, Amendment Application Nos. 177 and 163 DEFINITION OF DOSE EQUIVALENT I-131, San Onofre Nuclear Generating Station, Units 2 and 3.

8.13 Letter from R. M. Rosenblum (SCE) to the Document Control Desk, dated September 12, 1991. Subject: Docket Nos. 50-361 and 50-362, Station Blackout Analysis for San Onofre Nuclear Generating Station Units 2 and 3.

8.14 S-PEC-221, Combustion Engineering Calculation "Feedwater Requirements For Plant Cooldown Following Loss Of Offsite AC Power (CDDC#39999)," CDM File No. C921216S5546.

8.15 California State Water Board Resolution No. 99-028, April 14, 1999.

8.16 "Request for Thermal Plan Exception," Southern California Edison Company, San Onofre Nuclear Generating Station, Prepared for the California Regional Water Quality Control Board, San Diego Region, May 5, 1997.

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Attachment A

(Existing Pages)

SONGS Unit 2

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C(1), 2.C(2), and 2.C(5) of Facility Operating License No. NPF-10 are hereby amended to read as follows:

   (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 (3390 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. 171, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   (5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

   This paragraph intentionally deleted.

3. In addition, paragraphs 2.C(23), 2.C(24), and 2.C(25) to Operating License No. NPF-10 are hereby added, to read as follows:

   (23) Emergency Preparedness Conditions

   a. Conditions of ASLB Initial Decision of May 14, 1982

   Within five (5) months of initially exceeding five (5) percent power, SCE shall:

   i. Demonstrate that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (See Initial Decision, Section IV, Paragraph D1.12, pp. 136-140),

1.1 Definitions

OPERABLE - OPERABILITY (continued)	equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3390 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. b. In MODES 1 and 2, the fuel and moderator

(continued)

Attachment B

(Existing Pages)

SONGS Unit 3

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

2. Paragraphs 2.C(1), 2.C.(2) and 2.C(20) of Facility Operating License No. NPF-15, are hereby amended to read as follows:

(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 (3,390 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. 162, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(20) Qualification of Auxiliary Feedwater (AFW) Pump Motor Bearings

Prior to startup following the first refueling outage, SCE shall install and make operational the lubrication oil cooling system for the auxiliary feedwater pump motor bearings described in SCE's letter of March 7, 1983. Prior to installation of the lube oil cooling system, SCE shall perform daily visual inspection of the steam lines in the AFW pump room in accordance with SCE's letter of July 12, 1982.

3. Paragraphs 2.C(23) through 2.C(26) are hereby Facility Operating License NPF-15, as follows:

(23) Fuel Assembly Shoulder Gap Clearance (SCE letter of July 25, 1983)

Prior to entering Startup (Mode 2) after each refueling, SCE shall either provide a report that demonstrates that the existing fuel element assembly (FEA) has sufficient available shoulder gap clearance for at least the next cycle of operation, or identify to the NRC and implement a modified FEA design that has adequate shoulder gap clearance for at least the next cycle of operation. The commitment will apply until the NRC concurs that the shoulder gap clearance provided is adequate for the design life of the fuel.

(24) Isolation Capability for Primary EOF

By January 1, 1984 the primary EOF ventilation system shall be modified to provide isolation capability as described in the SCE letter of July 22, 1983.

1.1 Definitions

OPERABLE – OPERABILITY (continued)	equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3390 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. b. In MODES 1 and 2, the fuel and moderator

(continued)

Attachment C

(Proposed Pages)

(Redline and Strikeout)

SONGS Unit 2

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C(1), 2.C(2), and 2.C(5) of Facility Operating License No. NPF-10 are hereby amended to read as follows:

(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 (~~3390~~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. ~~171~~, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

This paragraph intentionally deleted.

3. In addition, paragraphs 2.C(23), 2.C(24), and 2.C(25) to Operating License No. NPF-10 are hereby added, to read as follows:

(23) Emergency Preparedness Conditions

a. Conditions of ASLB Initial Decision of May 14, 1982

Within five (5) months of initially exceeding five (5) percent power, SCE shall:

i. Demonstrate that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (See Initial Decision, Section IV, Paragraph D1.12, pp. 136-140),

1.1 Definitions

OPERABLE – OPERABILITY (continued)	equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3390~~3438 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. b. In MODES 1 and 2, the fuel and moderator

(continued)

Attachment D

(Proposed Pages)

(Redline and Strikeout)

SONGS Unit 3

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

2. Paragraphs 2.C(1), 2.C.(2) and 2.C(20) of Facility Operating License No. NPF-15, are hereby amended to read as follows:

(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 (~~3,390~~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. ~~162~~, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(20) Qualification of Auxiliary Feedwater (AFW) Pump Motor Bearings

Prior to startup following the first refueling outage, SCE shall install and make operational the lubrication oil cooling system for the auxiliary feedwater pump motor bearings described in SCE's letter of March 7, 1983. Prior to installation of the lube oil cooling system, SCE shall perform daily visual inspection of the steam lines in the AFW pump room in accordance with SCE's letter of July 12, 1982.

3. Paragraphs 2.C(23) through 2.C(26) are hereby Facility Operating License NPF-15, as follows:

(23) Fuel Assembly Shoulder Gap Clearance (SCE letter of July 25, 1983)

Prior to entering Startup (Mode 2) after each refueling, SCE shall either provide a report that demonstrates that the existing fuel element assembly (FEA) has sufficient available shoulder gap clearance for at least the next cycle of operation, or identify to the NRC and implement a modified FEA design that has adequate shoulder gap clearance for at least the next cycle of operation. The commitment will apply until the NRC concurs that the shoulder gap clearance provided is adequate for the design life of the fuel.

(24) Isolation Capability for Primary EOF

By January 1, 1984 the primary EOF ventilation system shall be modified to provide isolation capability as described in the SCE letter of July 22, 1983.

1.1 Definitions

OPERABLE – OPERABILITY (continued)	equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3390~~3438 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. b. In MODES 1 and 2, the fuel and moderator

(continued)

Attachment E

(Proposed Pages)

SONGS Unit 2

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C(1), 2.C(2), and 2.C(5) of Facility Operating License No. NPF-10 are hereby amended to read as follows:

(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. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(5) Environmental Qualification (Section 3.11, SER, SSER #3, SSER #4)

This paragraph intentionally deleted.

3. In addition, paragraphs 2.C(23), 2.C(24), and 2.C(25) to Operating License No. NPF-10 are hereby added, to read as follows:

(23) Emergency Preparedness Conditions

a. Conditions of ASLB Initial Decision of May 14, 1982

Within five (5) months of initially exceeding five (5) percent power, SCE shall:

i. Demonstrate that both meteorological towers and the Health Physics Computer System are fully installed and operational. SCE shall maintain offsite assessment and monitoring capabilities, essentially as described in the hearing (See Initial Decision, Section IV, Paragraph D1.12, pp. 136-140),

1.1 Definitions

OPERABLE – OPERABILITY (continued)	equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3438 MWt.
REACTOR PROTECTIVE SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
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(continued)

Attachment F

(Proposed Pages)

SONGS Unit 3

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

2. Paragraphs 2.C(1), 2.C.(2) and 2.C(20) of Facility Operating License No. NPF-15, are hereby amended to read as follows:

(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

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(20) Qualification of Auxiliary Feedwater (AFW) Pump Motor Bearings

Prior to startup following the first refueling outage, SCE shall install and make operational the lubrication oil cooling system for the auxiliary feedwater pump motor bearings described in SCE's letter of March 7, 1983. Prior to installation of the lube oil cooling system, SCE shall perform daily visual inspection of the steam lines in the AFW pump room in accordance with SCE's letter of July 12, 1982.

3. Paragraphs 2.C(23) through 2.C(26) are hereby Facility Operating License NPF-15, as follows:

(23) Fuel Assembly Shoulder Gap Clearance (SCE letter of July 25, 1983)

Prior to entering Startup (Mode 2) after each refueling, SCE shall either provide a report that demonstrates that the existing fuel element assembly (FEA) has sufficient available shoulder gap clearance for at least the next cycle of operation, or identify to the NRC and implement a modified FEA design that has adequate shoulder gap clearance for at least the next cycle of operation. The commitment will apply until the NRC concurs that the shoulder gap clearance provided is adequate for the design life of the fuel.

(24) Isolation Capability for Primary EOF

By January 1, 1984 the primary EOF ventilation system shall be modified to provide isolation capability as described in the SCE letter of July 22, 1983.

1.1 Definitions

OPERABLE – OPERABILITY (continued)	equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the SONGS Units 2 and 3 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
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(continued)

Attachment G

(Proposed BASES Page)

(Redline and Strikeout)

San Onofre Unit 2

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

during an MSLB). Both results are within the design. (See the Bases for Specifications 3.6.4, "Containment Pressure," and 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 102% of 3390 MWt ~~RTP~~(100% + 2% for instrument error of the original RTP of 3390 MWt. Increased instrument accuracy has allowed an increase to the Licensed RTP to the current level of 3438 MWt), one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of 120°F and 14.7 psia. The analyses also assume a response time delayed initiation in order to provide a conservative calculation of peak containment pressure and temperature responses.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation reduces the containment pressure to -4.2 psig due to the sudden cooling effect in the interior of the air tight containment. Additional discussion is provided in the Bases for Specification 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint coincident with an SIAS to achieve full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

The performance of the containment cooling train for post accident conditions is given in Reference 2. The result of the analysis is that each train can provide 50% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 2.

The modeled Containment Cooling System actuation from the containment analysis is based upon the unit specific response time associated with exceeding the CCAS to achieve full Containment Cooling System air and CCW System water flow.

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC Policy Statement.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Nine MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 5.2 (Ref. 1). The MSSVs' rated capacity passes the full steam flow at 102% of 3390 MWt~~RATED THERMAL POWER (RTP)~~ (100% + 2% for instrument error of the original RATED THERMAL POWER [RTP] of 3390 MWt. Increased instrument accuracy has allowed an increase to the Licensed RTP to the current level of 3438 MWt) with the valves full open. This meets the requirements of Section III of the ASME Code (Ref. 2).

The ASME requirement that MSSVs lift settings should be within 1% of the specified setpoint reflects two separate objectives: the objective to maintain lift setpoints within the bounds of the Safety Analysis and an objective to minimize the number of valves which operate to mitigate an event by staggering the valve setpoints.

This second requirement to stagger setpoints reflects good engineering design, but not safety requirements. The objective to stagger valve setpoints constrains the less restrictive Safety Analysis requirement as a condition of Operability.

The radiological release assumptions used in the Steam Generator Tube Rupture dose assessment bound the source terms which are based on a low MSSV setpoint of 1100 psia with 15% MSSV blowdown, and considering the appropriate setpoint tolerance.

(continued)

BASES (continued)

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 30 minutes following a reactor trip from 102% of 3390 MWt (100% + 2% for instrument error of the original RTP of 3390 MWt. Increased instrument accuracy has allowed an increase to the Licensed RTP to the current level of 3438 MWt) ~~RTP~~, and then cool down the RCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during the cooldown, as well as to account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

The combined volume of CST ensures that sufficient water is available to maintain the unit in MODE 3 for 24 hours including cooldown to shutdown cooling initiation.

OPERABILITY of the CST is determined by maintaining the tank volume at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODES 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST volume is not within the limit, the OPERABILITY of the backup water supply must be verified by administrative means within 4 hours.

OPERABILITY of the backup feedwater supply must include verification of the OPERABILITY of flow paths from the backup supply to the AFW pumps, and availability of the required volume of water in the backup supply. The CST volume must be returned to OPERABLE status within 7 days, as the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

(continued)