December 23, 2004

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

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS ON EQUIPMENT HATCH OPEN DURING REFUEL OPERATIONS (TAC NOS. MC0317 AND MC0318)

Dear Mr. Ray:

The Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. NPF-10 and Amendment No. 184 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 4, 2003, as supplemented by letters dated December 24, 2003, and June 3, August 24, and October 6 and 22, 2004.

The amendments revise TS 3.9.3, "Containment Penetrations," by adding a note to the limiting condition for operation to permit the containment equipment hatch to be open during core alterations and movement of irradiated fuel inside containment during refueling operations.

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

Sincerely,

/RA/

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

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 193 to NPF-10
2. Amendment No. 184 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. NPF-10

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

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee), dated August 4, 2003, as supplemented by letters dated December 24, 2003, and June 3, August 24, and October 6 and 22, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 193, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance, including the incorporation of the changes to the Technical Specification Bases as described in the licensee's letters dated August 4 and December 24, 2003, and June 3, August 24, and October 6 and 22, 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 23, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 193

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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

REMOVE	INSERT
3.9-4	3.9-4

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. NPF-15

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

   A. The application for amendment by Southern California Edison Company, et al. (SCE or the licensee) dated August 4, 2003, as supplemented by letters dated December 24, 2003, and June 3, August 24, and October 6 and 22, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 184, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance, including the incorporation of the changes to the Technical Specification Bases as described in the licensee's letters dated August 4 and December 24, 2003, and June 3, August 24, and October 6 and 22, 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Robert Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: December 23, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 184

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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

REMOVE	INSERT
3.9-4	3.9-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By application dated August 4, 2003, as supplemented by letters dated December 24, 2003, and June 3, August 24, and October 6 and 22, 2004, Southern California Edison Company (the licensee) requested changes to the Technical Specifications (TSs) for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The proposed amendments would revise TS 3.9.3, "Containment Penetrations," by adding a note to the limiting condition for operation to permit the containment equipment hatch to be open during core alterations and movement of irradiated fuel inside containment during refueling operations.

The supplemental letters dated December 24, 2003, and June 3, August 24, and October 6 and 22, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 18, 2003 (68 FR 54752).

2.0 REGULATORY EVALUATION

The licensee's description of the proposed amendment, and the technical and regulatory analyses in support of its proposed amendment are described in Sections 2.0, 4.0 and 5.2 of Enclosure 2 to the licensee's August 4, 2003, application.

The NRC staff finds that the licensee in Section 5.2 of its submittal identified applicable regulatory requirements. The proposed amendments would allow the equipment hatch to be open during refueling operations when there are core alterations or movement of irradiated fuel assemblies inside containment. Based on this, the proposed amendments involve the staff’s evaluation of the licensee’s design basis fuel handling accident (FHA) inside containment, and containment integrity (i.e., the equipment hatch is part of the containment pressure boundary), during refueling operations or Mode 6.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 19, “Control Room,” requires that licensees maintain the control room in a safe condition under accident conditions. Under these conditions, the licensee must provide adequate radiation protection to permit access and occupancy of the control room. 10 CFR 100.11, “Determination of exclusion area, low population zone and population center distance,” on the other hand, establishes the dose limits for the exclusion area and for the public.

In order to show that the radiation doses, onsite and offsite, will meet the above regulatory requirements, licensees have performed evaluations of their accident radiation doses. Regulatory guidance for these evaluations is provided in the form of regulatory guides (RGs) and standard review plans (SRPs). The regulatory requirements from which the NRC staff based its review are contained in 10 CFR Part 50, Appendix A, GDC 19 and 10 CFR 100.11, as supplemented by SRP 6.4, "Control Room Habitability System." Except where the licensee proposed a suitable alternative, the NRC staff used the regulatory guidance provided in the following documents in performing this review.

- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.”

- RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

- SRP 6.4, "Control Room Habitability System."

The NRC staff also considered relevant information in the SONGS Updated Final Safety Analysis Report (UFSAR), TSs, responses to Generic Letter (GL) 2003-01, “Control Room Habitability,” and the licensee’s August 4, 2003, submittal and its supplements.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendments which are described in Sections 2.0 and 4.0 of Enclosure 2 to the licensee's August 4, 2003, submittal and its supplements.

As stated in its application, the licensee proposes to add the following note to LCO 3.9.3:

> The equipment hatch may be open if all of the following conditions are met:
>
> 1. The Containment Structure Equipment Hatch Shield Doors are capable of being closed within 30 minutes,

2. The plant is in Mode 6 with at least 23 feet of water above the reactor vessel flange,
3. A designated crew is available to close the Containment Structure Equipment Hatch Shield Doors,
4. Containment purge is in service, and
5. The reactor has been subcritical for at least 72 hours.

The licensee stated that the proposed change is to permit the containment equipment hatch to remain open during core alterations and movement of irradiated fuel assemblies inside containment during refueling outages. This is currently not permitted by TS 3.9.3.

The postulated accident that could result in a release of radioactive material through the equipment hatch would be an FHA inside containment, as discussed in the evaluation below.

3.1 Administrative Controls

The licensee has proposed to have both the equipment hatch and the missile shield doors open during core alterations or fuel movement inside containment. However, the missile shield doors, under administrative controls, would be maintained in an isolable condition (i.e., capable of being closed) and such controls would require the following in place:

- The missile shield doors are capable of being closed in 30 minutes.

- A designated crew is available to close the containment structure equipment hatch shield doors.

- Flashing would be added to the top and sides of the shield doors to retard or restrict a release of post-accident fission products when the doors are closed.

- The capability to close the shield doors includes requirements that any cables or hoses across the opening have quick disconnects to ensure the doors are capable of being closed in the 30 minutes.

- The 30-minute closure time is considered to begin when the control room communicates the need to shut the containment structure equipment hatch shield doors.

In its application, the licensee explained that the licensed operator supervising the movement of the irradiated fuel assemblies is in constant communication with the control room and is procedurally required to inform the control room that the containment evacuation alarm be sounded in the event of an FHA inside containment which requires the personnel inside containment to evacuate.

In its supplemental letter of December 24, 2003, the licensee stated that the designated crew would be part of the routine crew used to take in and out equipment from containment through the equipment hatch, and the requirements for this crew would be specified in administrative procedures. The crew would be stationed in the vicinity of the open equipment hatch for the unit in refueling. This may include the area as far away as the opposite unit equipment hatch where equipment is stored waiting to go in containment or place equipment being moved out of containment. These storage areas are within a few hundred feet of the open equipment hatch and the time it takes to respond from these locations will not significantly impact the time it takes to achieve closing of the missile shield doors.

A description of the administrative controls is given in the licensee's application and will be added to the Bases of the TSs.

In justifying the 30-minute closure time for the shield doors, the licensee stated that, at the conclusion of the last Unit 3 refueling outage on February 12, 2003, engineering and licensing personnel observed maintenance personnel closing the missile shield doors. The licensee states that this closure was completed within 30 minutes under what the licensee called the extremely adverse conditions of a rainstorm without the pre-staging of manual chainfalls used to close the doors.

3.2 Tornado Missiles

The missile shield doors and the equipment hatch provide missile protection for inside the containment. During Modes 1 through 4, when containment integrity is required, the missile shield covers the equipment hatch. The equipment hatch shield doors are designed for protection against tornado generated missiles. The tornado-generated missiles considered are provided in UFSAR Table 3.5-6.

In addressing what will happen on site during refueling with severe weather in the vicinity of the plant, the licensee, by letter dated August 24, 2004, stated that abnormal operating instructions are in place to verify that missile barrier doors are closed, including the missile shield doors for severe weather. The licensee stated that the conditions that require entry into this procedure include tornado warning, hurricane watch, flash flood watch or warning, or tsunami warning.

3.3 Postulated Accidents

The limiting event during refueling when there are core alterations or movement of irradiated fuel inside containment is the FHA inside containment. The licensee has described this event in Sections 15.7.3.9 and 15.10.3.9 of the UFSAR and the NRC staff's acceptance criteria is given in Sections 6.4 and 15.7.4 of the SRP. The dose models used by the licensee to evaluate the consequences of accidents, including the FHA inside containment, are contained in UFSAR Appendices 15B and 15.10B.

The licensee's calculated potential dose consequences for the FHA inside containment at the exclusion area boundary (EAB) and the assumptions used for the calculated dose consequences are in Tables 1 and 2, respectively, attached to this safety evaluation (SE).

By letters dated February 28, 1995, and October 8, 1996, the NRC staff issued SEs for SCE's amendment requests to allow both doors of the containment personnel hatch to be open during refueling operations and to allow an upgrade or replacement of containment area and airborne radiation monitoring instrumentation, respectively. In both of these amendments, the NRC staff conducted reviews of SONGS's FHA analysis for dose consequences. However, SCE has since re-analyzed the FHA described in the SONGS UFSAR Chapter 15 accidents. The re-analyses include changes in the release characteristics from the containment, and an increase the amount of unfiltered inleakage assumed to enter into the control room. These changes alter the releases during a FHA and the offsite and control room doses, and is therefore, evaluated in this SE.

3.3.1 FHA Radiological Consequence Analysis

Prior to January 2000, SONGS's FHA licensing basis assumed that a fuel bundle dropped onto the ground would result in all 236 fuel pins of the bundle failing, releasing fission product gases from all 236 failed fuel pins. In 2000, SCE updated its FSAR, revising its FHA assumptions to reflect a more mechanistic approach in determining the number of failed pins. The current SONGS FHA analysis postulates that a more realistic worst-case FHA scenario would result from a spent fuel assembly dropped onto a partially filled core during refueling, causing a total of 226 fuel pins to fail. This analysis is described in Section 15.10.7.3.9 of the licensee's UFSAR.

In determining the consequences of a FHA, the determination of the amount of iodine contained in the damaged fuel assemblies is highly dependent on the design and operation of the reactor core. In SCE's FHA analysis of record (AOR), the licensee applies conservative and bounding values for the radial peaking factor (RPF) and relative power density (RPD). The RPF and RPD reflect the overall power production in individual fuel rod pins and entire fuel assemblies, respectively. For its FHA, the licensee's AOR assumes the bounding RPF for the dropped assembly is 1.71 and the maximum RPD for the impacted assembly is 1.37.

During each reload analysis, the licensee calculates the maximum cycle-specific RPF and RPD and verifies that each is lower than the value assumed in the FHA AOR. The licensee uses an NRC-approved reload analysis methodology (Reference 1) which incorporates the NRC-approved ROCS-MC computer code for determining the cycle-specific maximum RPF and RPD. The licensee controls the reload analysis process through procedure SO23-XXXVI-2.10, "Core Reload Analyses and Activities Checklist." The licensee stated that Section 1.5 of this procedure addresses the reload cycle dose analysis validation for the FHA. Additionally, the licensee stated that the reload cycle dose analysis validation includes verification of the input parameters, including the peaking factors modeled in the current FHA dose analysis.

The licensee's FHA AOR assumes that the dropped assembly will fail 16 fuel rods due to the impact and conservatively applies the bounding 1.71 RPF to all 16 rods. In its reload analysis, the licensee verifies that the maximum cycle-specific RPF calculated is bounded by the 1.71 value assumed in the AOR. Likewise, the licensee's FHA AOR assumes, that in the impacted assembly, an additional 210 fuel rods will fail due to the force of the impact. For these fuel rods, the licensee calculates and verifies that the maximum cycle-specific RPD for any fuel assembly loaded in the core is less than 1.37.

In 1972, the NRC published RG 1.25, "Assumption Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." RG 1.25 states that the appropriate iodine gap release fraction to be assumed in a FHA is 10 percent. In 1988, the NRC issued NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," which concluded that the 10 percent assumption listed RG 1.25 was appropriate for low burnup fuel but that the iodine gap release fraction should be increased by 20 percent–to 12 percent–for high burnup fuel. The licensee's FHA AOR assumes and applies the more conservative 12 percent iodine gap release fraction to both low and high burnup fuel. This provides additional margin in assuring that the licensee will satisfy the 10 CFR Part 100 dose requirements.

In SONGS's FHA analysis, the fuel rods, as described above, are conservatively assumed to rupture, releasing the radionuclides within the fuel rod to the reactor cavity water. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity depending on their physical and chemical form. The licensee assumed no decontamination for noble gases, an overall effective decontamination factor of 100 for radioiodines, and retention of all particulate fission products. SCE also assumed that essentially 100 percent of the fission products released from the reactor cavity are released to the environment in 2 hours without any credit for filtration.

The assumptions provided by SCE are presented in Table 2, and the EAB and control room doses estimated by the licensee for the FHA were found to be acceptable. The NRC staff performed independent calculations using the SCE assumptions and confirmed the licensee's conclusions.

3.3.1.1 Atmospheric Dispersion Estimates

SCE performed the reanalyses of the FHA using the same control room and EAB atmospheric dispersion factors (χ/Q values) used in the previous FHA analyses described in SONGS UFSAR Chapter 15. A description of the development of these χ/Q values is provided in SONGS UFSAR Chapter 2.3.4. The UFSAR control room atmospheric dispersion factor of 3.1×10^{-3} sec/m^3 is based on the Murphy & Campe diffuse source-point receptor model (Reference 2) whereas the UFSAR EAB atmospheric dispersion factor of 2.72×10^{-4} sec/m^3 is generally based on the five percent overall site χ/Q value (excluding the effects of plume meander) described in RG 1.145.

In Question 17 of the request for additional information (RAI) letter dated November 7, 2003, the NRC staff commented on the use of the UFSAR Murphy & Campe diffuse source-point receptor control room atmospheric dispersion factor for the FHA open containment scenario. The NRC staff stated that this χ/Q value (3.1×10^{-3} sec/m^3) is applicable when activity is assumed to leak from many points on the surface of the containment in conjunction with a single point receptor (e.g., control room air intake); that is, the activity is assumed to be homogeneously distributed throughout the containment and the release rate is assumed to be reasonably constant over the surface of the building. This is not the situation in the SONGS's postulated FHA scenario, where the release is assumed to occur through the open containment equipment hatch. Consequently, the licensee was asked to justify the use of the UFSAR Murphy & Campe diffuse source-point receptor χ/Q value of 3.1×10^{-3} sec/m^3 in its FHA open containment dose analysis.

In its RAI response to the NRC staff, dated June 3, 2004, SCE stated that a new set of control room χ/Q values were calculated for comparison with the UFSAR control room χ/Q value of 3.1×10^{-3} sec/m^3 using the guidance in RG 1.194. This new set of control room χ/Q values was calculated using the ARCON96 computer code (Reference 3).

The licensee used 10 years of hourly onsite meteorological data collected during calendar years 1993 to 2002 in order to generate the ARCON96 χ/Q values. Wind speed and wind direction were measured at 10 and 40 meters above ground level and atmospheric stability classification was based on temperature difference measurements between these two levels. Section 2.3.3.1 of the SONGS UFSAR states that the onsite meteorological measurement system is consistent with the recommendations of RG 1.23, "Onsite Meteorological Programs."

SCE provided an electronic copy of the hourly meteorological data used as input to the ARCON96 computer runs as well as copies of the resulting ARCON96 output in its RAI response Letter dated October 6, 2004. The NRC staff performed a perfunctory review of a subset of the ARCON96 hourly meteorological data base using the methodology described in NUREG-0917 (Reference 4). Further review was performed using computer spreadsheets. The data recovery rate during the ten-year period 1993–2002 exceeded the RG 1.23 goal of 90 percent. Examination of the data revealed that stable and neutral atmospheric conditions were generally reported to occur at night and unstable and neutral conditions were generally reported to occur during the day, as expected. Wind speed, wind direction, and stability class frequency distributions were reasonably similar from year to year, with the exception that the average lower and upper level wind speeds in 1999 were approximately 1.8 times higher than the lower and upper level wind speeds averaged over the remaining 9-year period (1993–1998 and 2000–2002). This discrepancy in wind speed values does not have a significant impact on the conclusion of this analysis, since the resulting bounding ACRON96 χ/Q value is only approximately 25 percent of the 3.1×10^{-3} sec/m^3 UFSAR control room χ/Q value for this license amendment request.

The NRC staff qualitatively reviewed the inputs to the ARCON96 computer runs and found them generally consistent with site configuration drawings and NRC staff practice. Six release-receptor combinations representing two release locations (Units 2 and 3 containment equipment hatches) and three receptors (control room normal, Unit 2 and 3 emergency air intakes) were evaluated. The containment equipment hatch releases were modeled as a ground level area (diffuse) source. Leakage was assumed to occur through the open hatch and the source dimensions were based on the face area of the equipment hatch. The initial diffusion coefficients were determined by dividing the source dimensions by a factor of six in accordance with RG 1.194. The release height was set to the mid-height of the equipment hatch and the distance-to-receptor was set as the shortest path around the containment. Atmospheric dispersion factors were calculated assuming flow both around and over (or through) the containment building and the resulting highest χ/Q value was used.

A comparison of the resulting bounding (highest) 0-2 hour ARCON96 χ/Q value (7.99×10^{-4} sec/m^3) with the UFSAR control room χ/Q value of 3.1×10^{-3} sec/m^3 shows that the use of the UFSAR control room χ/Q value in the dose analysis for this license amendment request is conservative. However, because of the potential discrepancy in wind speed data revealed during the NRC staff's perfunctory review of a subset of the 1993–2002 onsite meteorological data, the NRC staff does not endorse the use of this data set in other future licensing actions without further review. Notwithstanding this, the NRC staff concludes that the conservatism of SONGS's UFSAR control room χ/Q value bounds the ARCON96 results and is conservative.

With respect to the EAB χ/Q value of 2.72×10^{-4} sec/m^3, the licensee had previously used this value in its FHA dose analyses as contained in the SONGS UFSAR Chapter 15. The NRC staff has reviewed the licensee's use of this existing UFSAR EAB χ/Q value and has found it to be appropriate for the application in which it is being used without change. On the basis of this review, the staff concludes that this EAB χ/Q value is acceptable for use in this license amendment request.

3.3.1.2 Control Room Doses and Unfiltered Inleakage

The NRC staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room inleakage rates assumed by licensees in analyses of control room habitability. The NRC staff issued GL2003-01, "Control Room Habitability," on June 12, 2003. SCE responded to this GL by letter dated September 17, 2004. In its response, SCE reported that inleakage testing using the ASTM E741 tracer gas methodology determined a control room unfiltered inleakage rate of 259 cfm during the duel train pressurization mode (Train A and B both operating). The proposed values assumed for the FHA are provided in Table 2. These values plus 10 cfm for ingress and egress are larger than the measured values reported in the licensee's tracer gas test results.

The NRC staff is still reviewing SCE's September 17, 2004, response for final resolution of GL 2003-01. However, the NRC staff has determined that there is reasonable assurance that the SONGS control room would be habitable during the design basis FHA, and that an evaluation of the licensee's current amendment requests can be made prior to final resolution of the generic issue. The NRC staff made this determination based on (1) the results of the tracer gas testing at SONGS, (2) the independent confirmatory calculations performed by the NRC staff, and (3) the available margin between the licensee's FHA assumed inleakage (1000 cfm) and the actual measured inleakage (259 cfm). The NRC staff believes the margin gap was necessary to adequately demonstrate conservatism in the licensee's assumption, since the 259 cfm was actually calculated based on single train inleakage measurements rather than through direct measurement during dual train operation (i.e., in the case that the most limiting condition existed in dual train operation). This SE's finding, that SCE's assumptions for control room doses and unfiltered inleakage during the FHA are acceptable, is limited to only the scope of this amendment's request. As the NRC staff continues its review of GL 2003-01, additional information may be necessary to supplement the licensee's September 17, 2004, response letter. Any future resolution will be addressed in separate correspondence once review of the generic issue for SONGS is complete.

3.3.1.3 Offsite Doses

The EAB and control room doses estimated by SCE for the FHA were found to be acceptable. The NRC staff performed independent calculations and confirmed the licensee's conclusions.

3.3.1.4 Conclusions

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed change to the SONGS TSs. Based on its review, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0, above. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the control room doses would continue to comply with GDC 19 (5 rem whole body or 30 rem thyroid). The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB doses would continue to be well within 10 CFR Part 100 (6.3 rem whole body and 75 rem thyroid). Therefore, the proposed license amendment is acceptable with regard to the radiological consequences of the postulated FHA.

The NRC staff has reviewed the description of the administrative controls in the licensee's application and concludes that the description is acceptable. In its June 3, 2004, supplemental letter, the licensee agreed to add this description to the TS Bases during the implementation of the amendments. This will be a condition of the amendment to the operating licenses. Therefore, when the amendments are incorporated into the TSs, the description of the administrative controls will become a part of the Bases of the TSs. Any changes to the description of the administrative controls will be controlled by Section 5.5.14 of the Administrative Controls Section of the TSs.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published September 18, 2003 (68 FR 54752). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Document SCE-9801-P-A, “Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3,” June 1999.

2. Murphy, K. G. and Campe, K. M., “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19,” Proceedings of the 13th AEC Air Cleaning Conference held August 12–15, 1974, CONF 740807, Vol. I, pp. 401–430.

3. Ramsdell, Jr., J. V. and Simonen, C. A., "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997.

4. Snell, W., "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," NUREG-0917, July 1982.

Attachments: 1. Table 1, Calculated Radiological Dose Consequences
2. Table 2, Parameters and Assumptions Used in Analysis of Radiological Dose Consequences

Principal Contributors: J. Donohew
W. Blumberg
B. Harvey
R. Taylor

Date: December 23, 2004

Table 1
Calculated Radiological Dose Consequences

Exclusion Area Boundary	**Dose**	**SRP 15.7.4 Limits**
Whole Body	0.3 Rem	6 Rem
Thyroid	56.4 Rem	75 Rem
Control Room Operator	**Dose**	**GDC 19 Limits**
Whole Body	0.3 Rem	5 Rem
Thyroid	25.4 Rem	Equivalent to 5 Rem Whole Body (30 Rem per SRP Section 6.4)

Table 2 (sheet 1 of 2)
Parameters and Assumptions Used in Analysis of FHA
(Accident in Containment with Equipment Hatch Open)

Core thermal power, MWt	3458
Time between plant shutdown and accident, hrs.	72
Fraction of Gap Activity Released to the Refueling Cavity Water, %	100
Minimum Water Depth Above Reactor Vessel Flange (and damaged fuel rods), feet	23
Refueling Cavity Water Decontamination Factors:	
Noble Gases	1
Iodine	100
Airborne Iodine Forms, %	
Elemental	75
Organic	25
Exclusion Area Boundary Parameters	
Atmospheric Relative Concentration, sec/m^3	2.72E-4[a]
Dose Conversion Factors	ICRP-30
EAB Breathing Rate, m^3/sec	3.47E-4
EAB Occupancy Factor (0-2 hours)	1.0

[a]2.72E-4 is to be read 2.72×10^{-4}

Table 2 (sheet 2 of 2)
Parameters and Assumptions Used in Analysis of FHA
(Accident in Containment with Equipment Hatch Open)

Control Room Parameters	
Atmospheric Relative Concentration, sec/m^3	3.1E-3
Dose Conversion Factors	ICRP-30
Control Room Breathing Rate m^3/sec	3.47E-4
Control Room Occupancy Factor	1.0
Control Room Volume, ft^3	266,920
Unfiltered ingress/egress rate, ft^3/min	10
Unfiltered Inleakage Rate, ft^3/min	990[b]
Control Room Normal HVAC System Operation (0 - 3 min):	
Normal Operation Unfiltered Inflow Rate, ft^3/min	5820
Control Room Isolation (switchover to CREACUS), min.	3
Control Room CREACUS Operation (3 min. - 8 hours):	
Filtered Inflow Rate, cfm	4,400
Inflow and Recirculation Filter Efficiencies, %	
Elemental Iodine	95
Organic Iodide	95
Particulate	99

[b] 990 ft^3/min does not include unfiltered inleakage due to ingress and egress.

cc:
Mr. Raymond Waldo, Plant Manager
Nuclear Generation
Southern California Edison Company
San Onofre Nuclear Generating Station
P. O. Box 128
San Clemente, CA 92674-0128

Mr. Douglas K. Porter
Southern California Edison Company
2244 Walnut Grove Avenue
Rosemead, CA 91770

Mr. David Spath, Chief
Division of Drinking Water and
Environmental Management
P. O. Box 942732
Sacramento, CA 94234-7320

Chairman, Board of Supervisors
County of San Diego
1600 Pacific Highway, Room 335
San Diego, CA 92101

Eileen M. Teichert, Esq.
Supervising Deputy City Attorney
City of Riverside
3900 Main Street
Riverside, CA 92522

Mr. Gary L. Nolff
Power Projects/Contracts Manager
Riverside Public Utilities
2911 Adams Street
Riverside, CA 92504

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

Mr. Michael Olson
San Diego Gas & Electric Company
P.O. Box 1831
San Diego, CA 92112-4150

Mr. Ed Bailey, Chief
Radiologic Health Branch
State Department of Health Services
Post Office Box 997414 (MS7610)
Sacramento, CA 95899-7414

Resident Inspector/San Onofre NPS
c/o U.S. Nuclear Regulatory Commission
Post Office Box 4329
San Clemente, CA 92674

Mayor
City of San Clemente
100 Avenida Presidio
San Clemente, CA 92672

Mr. Dwight E. Nunn, Vice President
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

Mr. James D. Boyd, Commissioner
California Energy Commission
1516 Ninth Street (MS 31)
Sacramento, CA 95814

Mr. Joseph J. Wambold, Vice President
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92764-0128

Mr. Steve Hsu
Department of Health Services
Radiologic Health Branch
MS 7610, P.O. Box 997414
Sacramento, CA 95899

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