December 29, 2006

Mr. Richard M. Rosenblum
Senior Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: FULL-SCOPE IMPLEMENTATION OF AN ALTERNATIVE SOURCE TERM (TAC NOS. MC5495 AND MC5496)

Dear Mr. Rosenblum:

The Commission has issued the enclosed Amendment No. 210 to Facility Operating License No. NPF-10 and Amendment No. 202 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3), respectively. The amendments consist of changes to the Updated Final Safety Analysis Report in response to your application dated December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006.

The amendments revise the SONGS 2 and 3 accident source term (AST) used in the design-basis radiological consequence analyses. These license amendments are in accordance with the requirements of Section 50.67 of Title 10 of the *Code of Federal Regulations*, which addresses the use of an AST at operating reactors, and relevant guidance of Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.” These license amendments represent full-scope implementation of the AST described in Regulatory Guide 1.183.

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/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 210 to NPF-10
2. Amendment No. 202 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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/

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
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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. 210
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 December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006, 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, by Amendment No. 210, the license is amended to approve changes to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application for amendment by SCE dated December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006. SCE shall update the UFSAR to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of issuance. Implementation of the amendment is the incorporation of the UFSAR changes to the description of the facility as described in the licensee's application dated December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006, and evaluated in the staff's Safety Evaluation dated December 29, 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: December 29, 2006

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. 202
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 December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006, 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, by Amendment No. 202, the license is amended to approve changes to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application for amendment by SCE dated December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006. SCE shall update the UFSAR to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of issuance. Implementation of the amendment is the incorporation of the UFSAR changes to the description of the facility as described in the licensee's application dated December 27, 2004, as supplemented by letters dated October 27, 2005, March 10, and October 6, 2006, and evaluated in the staff's Safety Evaluation dated December 29, 2006.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: December 29, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 210 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. NPF-15

FOR IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

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 December 27, 2004 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML043650403), as supplemented by letters dated October 27, 2005 (ADAMS Accession No. ML053040458), March 10 (ADAMS Accession No. ML060750661), and October 6, 2006 (ADAMS Accession No. ML062850525), Southern California Edison Company (SCE or the licensee) requested license amendments to change the Updated Final Safety Analysis Report (UFSAR) for San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3). The proposed amendments would revise the SONGS 2 and 3 design basis to replace the existing accident radiological source term (Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites.") with a full implementation of the alternative source term (AST) pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident Source Term." The supplements dated October 27, 2005, March 10, and October 6, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 1, 2005 (70 FR 5248).

The amendments revise the SONGS 2 and 3 AST used in the design-basis radiological consequence analyses. These license amendments are in accordance with the requirements of 10 CFR 50.67, which addresses the use of an AST at operating reactors, and relevant guidance of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." These license amendments represent full-scope implementation of the AST described in RG 1.183. In the 180-day response to Generic Letter 2003-1, “Control Room Habitability,” the licensee stated in its letter dated August 5, 2003 (ADAMS Accession No. ML032230360), that the current value of assumed unfiltered air inleakage into the control room envelope (CRE) in the design-basis radiological analyses is 10 cubic feet per minute (cfm) assumed for ingress and egress, with no other source of unfiltered inleakage into the CRE (i.e., 0 cfm of unfiltered inleakage), for a total analysis input of 10 cfm unfiltered inleakage to the CRE. The licensee subsequently stated in its letter dated September 17, 2004 (ADAMS Accession No. ML042650353), that the CRE unfiltered air inleakage testing performed at SONGS 2 and 3 in May 2004 provided results that exceeded 0 cfm assumed in the current design-basis radiological analyses and that the licensee performed an operability assessment in accordance with the guidance provided in “Operability Determinations and Degraded/Non-Confirming Condition Resolution,” in an NRC letter to Nuclear Energy Institute (NEI) dated January 30, 2004 (ADAMS Accession No. ML040160868).

Full-scope implementation of an AST requires, at a minimum, reanalysis of the loss-of-coolant accident (LOCA). In its LAR, the licensee reanalyzed the LOCA, the fuel handling accident (FHA) inside containment, FHA in the fuel handling building (FHB), and the Main Steamline Break (MSLB) accident outside containment. As part of implementing the AST, the licensee also requested to depart from the nucleate boiling (DNB) statistical convolution methodology for estimating fuel failure for non-LOCA events (i.e., the MSLB accident).

Based on its operability assessment, the licensee determined that the maximum amount of CRE unfiltered inleakage that would be consistent with an OPERABLE CRE is greater than the maximum unfiltered inleakage that has been demonstrated by CRE unfiltered inleakage testing. The licensee further determined that the CRE is nonconforming, but operable. In the time since the CRE unfiltered inleakage testing in May 2004, subsequent operability assessment was completed, and SONGS 2 and 3 have continued to operate in a nonconforming, but operable status. Accordingly, the licensee submitted this license amendment request (LAR) on December 27, 2004 to restore SONGS 2 and 3 to full qualification for meeting the control room (CR) dose acceptance criteria specified in 10 CFR 50.67 and General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50 and to incorporate an AST into the SONGS 2 and 3 design and licensing bases. There are no physical changes to plant equipment or operation of the plant requested in this LAR and the licensee requested no changes to the SONGS 2 and 3 technical specifications (TSs) in this LAR. Thus, the license amendment fulfills the commitment described in SCE's letter dated September 17, 2004, and establishes use of an AST methodology that documents the acceptability of an assumed increase in SONGS 2 and 3 CRE unfiltered inleakage rate to a value of 1,000 cfm (including ingress and egress related inleakage). As described in the September 17, 2004, letter, this is necessary to restore the CRE to full qualification.

2.0 REGULATORY EVALUATION

In this LAR, the licensee requested a full-scope implementation of the AST pursuant to 10 CFR 50.67, as described in RG 1.183. 10 CFR 50.67 provides a mechanism for licensed power reactors to replace the traditional source term used in their design-basis accident (DBA) radiological consequence analyses.

The NRC staff evaluated the radiological consequences of the proposed DBAs against the dose criteria specified in 10 CFR 50.67, “Accident Source Term”; these criteria are 25 rem, total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ), and 5 rem TEDE in the CR. The TEDE dose includes both noble gas and radioiodine exposure.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, GDC 19, and Standard Review Plan (SRP) Section 15.0.1. Except where the licensee has proposed a suitable alternative, the NRC staff used 10 CFR 50.67, as well as the regulatory guidance in the following documents below in its radiological consequence dose analyses:

- RG 1.23, “Onsite Meteorological Programs”
- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”
- RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors”
- NUREG-0800, “Standard Review Plan,” Section 6.4, “Control Room Habitability Systems”
- SRP Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”
- SRP Section 6.4, “Control Room Habitability Systems” (with regard to control room meteorology)
- SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Term”

The NRC staff also considered relevant information in the SONGS 2 and 3 UFSAR and TSs.

3.0 TECHNICAL EVALUATION

3.1 Atmospheric Dispersion Estimates

The licensee calculated new CR atmospheric dispersion factors (χ/Q values) for use in reanalyzing the bounding UFSAR Chapter 15 DBAs. The resulting SONGS 2 and 3 CR χ/Q values represent a change from those currently presented in Table 15B-4 of the SONGS 2 and 3 UFSAR. The licensee used the existing exclusion area boundary (EAB) and LPZ χ/Q values listed in Table 15B-4 of the SONGS 2 and 3 UFSAR to perform the offsite dose assessments for the bounding UFSAR Chapter 15 DBAs.

3.1.1 Meteorological Data

In its December 27, 2004, submittal, the licensee presented new CR χ/Q values that were generated using site meteorological data collected during the period 1993–2002. The licensee previously used these data to generate FHA CR χ/Q values associated with SONGS 2 License Amendment No. 193 and SONGS 3 License Amendment No. 184. The licensee previously provided a copy of these data in its letter to the NRC staff dated October 6, 2004 (ADAMS Accession No. ML042870468).

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. The combined data recovery of wind speed, wind direction, and stability (delta-temperature) exceeded the RG 1.23 goal of 90 percent during this 10-year period, although the upper level wind data recovery for 1994 and 1995 averaged around 80 percent. Section 2.3.3.1 of the SONGS 2 and 3 UFSAR states that the onsite meteorological measurements system is consistent with the recommendations of RG 1.23.

The NRC staff previously performed a basic review of a subset of the 1993–2002 site meteorological data as discussed in the SE associated with SONGS 2 and 3 License Amendments Nos. 193 and 184. The NRC staff performed a more comprehensive review of these data for this LAR using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. As expected, the NRC staff's examination of the data revealed generally stable and neutral atmospheric conditions at night and unstable and neutral conditions during the day. Wind speed, wind direction, and stability class frequency distributions were reasonably similar from year to year and generally consistent with the 1973–1976 and 1979–1983 data presented in the SONGS 2 and 3 UFSAR, 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).

In its request for additional information (RAI) letter dated October 7, 2005, the NRC staff asked the licensee to explain the abnormal 1999 wind speed data and their impact on the resulting ARCON96 dispersion analyses. In its submittal dated October 27, 2005, the licensee stated that the 1999 wind speed data were in error and revised its accident analyses (including the ARCON96 CR dispersion analyses) to reflect the correct meteorological data. The licensee submitted an electronic copy of the corrected meteorological data in its RAI response letter dated March 10, 2006. The NRC staff has reviewed the corrected 1999 wind speed data and has concluded that the revised 1999 wind speed frequency distributions more closely resemble the wind speed frequency distributions for the remaining 9-year period (1993–1998 and 2000–2002).

3.1.2 CR Atmospheric Dispersion Factors

SONGS 2 and 3 share a combined CR, with one normal mode air intake (located near the northwest corner of the CRE) and two emergency mode air intakes (located near the northwest and southwest corners of the CRE, respectively).

During the CR normal mode of operation, unfiltered outside air is introduced into the CR through the normal mode air intake. During the CR emergency mode of operation, the CR is isolated and filtered outside air is introduced into the CR through the emergency mode air intakes in order to pressurize the CR. The emergency mode of operation can be actuated either manually or automatically following a CR isolation signal (CRIS). The CRIS may be generated automatically by a safety injection actuation signal or by the detection of high radioactivity concentrations in the CR outside air flow.

The emergency mode of operation is facilitated by two 100 percent redundant subsystems, each with its own air intake. The licensee assumed the failure of one emergency train and modeled single emergency train operation throughout the duration of each event. The licensee stated that this single failure results in the largest CR doses. The licensee also assumed that unfiltered inleakage occurs at the beginning of each accident scenario and continues throughout the duration of each event.

The licensee calculated new χ/Q values to evaluate the impact of SONGS 2 and 3 main plant vent, containment shell, containment equipment hatch, main steam safety valve (MSSV), atmospheric dump valve (ADV), steamline break outside containment (SLB-OC), auxiliary feedwater (AFW) turbine steam discharge, refueling water storage tank (RWST) vent, and FHB releases on the SONGS 2 and 3 CR. Each of these nine release scenarios was evaluated for both SONGS 2 and 3 and for each of the three CR air intakes (the one normal mode air intake and the two emergency mode air intakes). For each potential release scenario, the licensee used the maximum χ/Q values calculated for each of these three air intakes to model air intake (including filtered air intake and unfiltered inleakage) into the CR. The licensee provided a plant layout showing the location of potential radiological release points with respect to the CR outside air intakes in its RAI response letter dated March 10, 2006.

In its RAI letter dated October 7, 2005, the NRC staff asked the licensee to confirm that there are no potential unfiltered inleakage pathways during both normal and emergency modes that could result in χ/Q values that are higher than the three CR air intake χ/Q values. In its RAI response dated March 10, 2006, the licensee stated that only the west side of the CRE is exposed to radioactive plumes released to the outside environment and all three CR air intakes are located on the west side of the CRE. The licensee stated that adjacent areas and structures to the north, south, and east of the CRE, and the adjacent areas and structures above and below the CRE, do not contain activity release points. These adjacent areas and locations can become contaminated only with air introduced via intake of infiltration of radioactive material contained in the radioactive plumes released to the outside environment which are then recirculated and diluted throughout these regions. Consequently, the licensee stated that there should be no potential unfiltered inleakage pathways during both normal and emergency modes that could result in χ/Q values that are higher than the three CR air intake χ/Q values.

The licensee used guidance provided in RG 1.194 to generate the new CR atmospheric dispersion factors. The licensee calculated these new CR χ/Q values using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analyses.

The licensee executed ARCON96 using the 1993–2002 hourly data from the site meteorological tower. Wind speed and wind direction data from the tower's 10-meter and 40-meter levels were provided as input and stability class was calculated using the temperature difference between the 40-meter and 10-meter levels. The resulting χ/Q values are presented in Table 2 of this SE. Details on the licensee's assessments of CR post-accident atmospheric dispersion conditions for each release scenario are provided below.

1. <u>Main Plant Vent Releases</u>: Each of the SONGS 2 and 3 units has a main plant vent which is located on top of each unit's containment structure. The licensee modeled the main plant vent releases as point sources using the ARCON96 ground-level release option. The plant vent release heights are both at 53.6 meters above plant grade (APG) as compared to the normal air intake height of 1.7 meters APG and the emergency air intake heights of 4.0 meters APG. The resulting ARCON96 analysis showed that the SONGS 2 main plant vent release to the SONGS 2 emergency air intake was the bounding source-receptor combination that resulted in the highest (i.e., most conservative) χ/Q values.

2. <u>Containment Shell Releases</u>: The licensee modeled the SONGS 2 and 3 containment shell (surface) releases as area (diffuse) sources using the ARCON96 ground-level release option. The area source dimensions were the maximum vertical and horizontal dimensions of the above-grade containment shell cross-sectional area perpendicular to the line of sight from the containment shell center to the CR air intakes. The initial diffusion coefficients (plume dimensions) were determined by dividing the area source dimensions by a factor of six in accordance with RG 1.194. The release heights (24.5 meters APG) were set equal to the mid-height of the containment shells above grade. The resulting ARCON96 analysis showed that the SONGS 2 containment shell release to the SONGS 2 emergency air intake was the bounding source-receptor combination that resulted in conservative χ/Q values.

3. <u>Equipment Hatch Releases</u>: The licensee modeled the SONGS 2 and 3 containment equipment hatch releases as area (diffuse) sources using the ARCON96 ground-level release option. The area source dimensions were based on the 5.8-meter diameter of the hatch openings. The initial diffusion coefficients (plume dimensions) were determined by dividing the area source dimensions by a factor of six in accordance with RG 1.194. The equipment hatch release heights (2.4 meters APG) were set equal to the mid-height of the hatch openings above grade. Since the SONGS 2 and 3 containment equipment hatches are on the opposite side of their respective containment

structures from the CR air intakes, atmospheric dispersion factors were calculated assuming flow both around and over (through) the containment buildings and the higher values were used. The resulting ARCON96 analysis showed that (1) the SONGS 2 containment shell release over (through) the containment building to the SONGS 2 emergency air intake was the bounding source-receptor combination for the 8 to 24-hour time period and (2) the SONGS 2 containment shell release around the containment building to the SONGS 2 emergency air intake was the bounding source-receptor combination for the remaining time periods. In its analyses, the licensee used a composite of these two conservative release pathways to derive conservative equipment hatch release χ/Q values.

4. MSSV Releases: SONGS 2 and 3 are both Combustion Engineering 2-loop pressurized water reactors which have two main steamlines for each unit. Each steamline has a set of nine safety valves centered around a main steamline isolation valve (MSIV). These valves may open either automatically, when the pressure in the main steamline reaches the valve setpoint, or manually by use of a valve lever. The licensee used the center of each MSIV as the MSSV release location for each steamline in determining the horizontal distances and directions to each of the three CR air intakes. Consequently the licensee modeled four MSSV release point locations (i.e., SONGS 2 MSSVs centered at MSIVs 8204 and 8205 and SONGS 3 MSSVs centered at MSIVs 8204 and 8205) to each of the three CR air intakes.

   The licensee modeled the MSSV releases as point sources using the ARCON96 ground-level release option. The MSSV release heights are 13.2 meters APG as compared to the normal air intake height of 1.7 meters APG and the emergency air intake heights of 4.0 meters APG. The resulting ARCON96 analysis showed that the SONGS 2 MSIV 8204 MSSV release to the SONGS 2 emergency air intake was the bounding source-receptor combination.

   RG 1.194 allows the ground level χ/Q values calculated with ARCON96 (on the physical height of the release point) to be reduced by a factor of 5 if (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed. MSSV releases satisfy both criteria, namely, (1) MSSV release points are uncapped and vertically oriented and (2) the calculated minimum MSSV stack exit velocity is 72 meters per second, which is considerably higher than the 10-meter 95th percentile wind speed value of 5.5 meters per second. Consequently, the licensee reduced the resulting ARCON96 MSSV χ/Q values by a factor of 5.[(1)]

5. ADV Releases: Each of the two steamlines for each of the two SONGS units has a power-operated atmospheric ADV. Consequently, the licensee modeled four ADV release points (i.e., SONGS 2 ADVs 606 and 607 and SONGS 3 ADVs 606 and 607) to each of the three CR air intakes.

[(1)]In its RAI letter dated October 7, 2005, the NRC staff asked the licensee whether a stuck-open MSSV concurrent with other DBAs is excluded from the licensing basis for the MSSVs. In its letter response dated March 10, 2006, the licensee responded that the SONGS 2 and 3 licensing basis, as reflected in UFSAR Section 10.3, does not require that a stuck-open MSSV be considered concurrently with other DBAs.

The licensee modeled the ADV releases as point sources using the ARCON96 ground-level release option. The locations of the ADV stacks were used to determine the distance and direction to the CR air intakes. The ADV release heights are 25.6 meters APG as compared to the normal air intake height of 1.7 meters APG and the emergency air intake heights of 4.0 meters APG. No credit was taken for a plume rise associated with the ADV release pathways. The resulting ARCON96 analysis showed that the SONGS 2 607 ADV release to the SONGS 2 emergency air intake was the bounding source-receptor combination[(2)] that resulted in conservative χ/Q values.

6. Steamline Break Outside Containment (SLB-OC) Releases: The locations of the postulated break for each of the two steamlines for each of the two SONGS units are assumed to occur downstream of the MSIVs and the releases are assumed to occur through blowout panels mounted on the roof of the MSIV/MFIV enclosure structures located directly above each main steamline. Consequently, the licensee modeled four SLB-OC release points (i.e., SONGS 2 north and south MSIV/MFIV enclosure roof blowout panels and SONGS 3 north and south MSIV/MFIV enclosure roof blowout panels) to each of the three CR air intakes.

   The licensee modeled the SLB-OC releases as area (diffuse) sources using the ARCON96 ground-level release option. The area source dimensions were based on the width of the area formed by the three blowout panels mounted on the roof of the MSIV/MFIV enclosure structure perpendicular to the line of sight from the MSIVs to the respective CR intake. The initial diffusion coefficients (plume dimensions) were determined by dividing the source dimensions by a factor of six in accordance with RG 1.194. The SLB-OC release heights (10.2 meters AGL) were set equal to the height of the blowout panels and no credit was taken for a plume rise. The resulting ARCON96 analysis showed that the SONGS 2 south MSIV/MFIV enclosure structure roof blowout panel release to the SONGS 2 emergency air intake was the bounding source-receptor combination that resulted in conservative χ/Q values.

7. Auxiliary Feedwater (AFW) Turbine Exhaust Releases: There is one AFW turbine for each of the two SONGS units. Consequently, the licensee modeled two AFW turbine release points (i.e., SONGS 2 turbine stack and SONGS 3 turbine stack) to each of the three CR air intakes.

   The licensee modeled the AFW turbine exhaust releases as point sources using the ARCON96 vent level release option which is not in accordance with RG 1.194. However, the licensee assumed both the vertical exit velocity and stack flow were zero. With both exit velocity and stack flow set to zero, the ARCON96 vent level release

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[(2)]Table 4.4-11 of Enclosure 2 of the licensee's December 27, 2004, submittal (which was modified in the licensee's October 27, 2005, submittal) also presented χ/Q values for ADV releases with plume rise credit. None of the DBA events addressed in the licensee's current AST application use these ADV release χ/Q values with plume rise credit. The licensee presented these ADV release χ/Q values with plume rise credit for use in potential future applications. However, in order to credit plume rise in an ADV release dose analysis, the time period for which the ADV stack vertical flow exit velocity exceeds five times the 95th percentile upper level wind speed would need to be determined before the plume rise adjustment factor could be applied.

option generates the same results as the ground-level release option recommended by RG 1.194. The AFW turbine stack release heights are 8.8 meters APG, as compared to the normal air intake height of 1.7 meters APG and the emergency air intake heights of 4.0 meters APG. The resulting ARCON96 analysis showed that the SONGS 2 AFW turbine exhaust release to the SONGS 2 emergency air intake was the bounding source-receptor combination that resulted in conservative χ/Q values.

8. RWST Releases: There are two RWSTs for each of the two SONGS units. Consequently, the licensee modeled four RWST release points (i.e., SONGS 2 RWSTs T005 and T006 and SONGS 3 RWSTs T005 and T006) to each of the three CR air intakes. The RWST releases are assumed to occur through the roof vent on each RWST.

   The licensee modeled the RWST releases as point sources using the ARCON96 vent release option which is not in accordance with RG 1.194. However, the licensee assumed both the vertical exit velocity and stack flow were zero. With both exit velocity and stack flow set to zero, the ARCON96 vent level release option generates the same results as the ground-level release option recommended by RG 1.194. The RWST vent release heights are approximately 12.7 meters above plant grade as compared to the normal air intake height of 1.7 meters APG and the emergency air intake heights of 4.0 meters APG. The resulting ARCON96 analysis showed that SONGS 2 RWST T006 vent release to the SONGS 2 emergency air intake was the bounding source-receptor combination for the 0–2 hour time period whereas the SONGS 2 RWST T005 vent release to the SONGS 2 emergency air intake was the bounding source-receptor combination for the remaining time periods. In its analyses, the licensee used a composite of these two conservative release pathways to derive conservative RWST χ/Q values.

9. FHB Releases: There is one FHB for each of the two SONGS units. Consequently, the licensee modeled two FHB release points (i.e., SONGS 2 FHB and SONGS 3 FHB) to each of the three CR air intakes. The FHB releases are assumed to occur through the closest and largest cask hatch in each FHB.

   The licensee modeled the FHB releases as point sources using the ARCON96 vent level release option which is not in accordance with RG 1.194. However, the licensee assumed both the vertical exit velocity and stack flow were zero. With both exit velocity and stack flow set to zero, the ARCON96 vent level release option generates the same results as the ground-level release option recommended by RG 1.194. The FHB cask hatch release heights are 10.2 meters APG as compared to the normal air intake height of 1.7 meters APG and the emergency air intake heights of 4.0 meters APG. The resulting ARCON96 analysis showed that the SONGS 2 FHB release to the SONGS 2 emergency air intake was the bounding source-receptor combination that resulted in conservative χ/Q values.

The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the SONGS 2 and 3 site. The NRC staff qualitatively reviewed the inputs to the

ARCON96 calculations and found them generally consistent with site configuration drawings and site practice. The NRC staff made an independent evaluation of the resulting bounding atmospheric dispersion estimates by running the ARCON96 computer code and found that the licensee's results were similar to or more conservative than the NRC staff's results. On the basis of this review, the NRC staff concludes that the χ/Q values for SONGS 2 and 3 DBA releases to the SONGS 2 and 3 CR as presented in Table 2 are acceptable for use in the DBA CR dose assessments performed in support of this license amendment request.

3.1.3 EAB and LPZ Atmospheric Dispersion Factors

The licensee evaluated offsite doses using the EAB and LPZ five percentile χ/Q values presented in the SONGS 2 and 3 UFSAR Appendix 15B Table 15B-4. These χ/Q values, which are presented in Table 3 of this SE, were generally based on the 5 percent overall site χ/Q methodology (excluding the effects of plume meander) described in RG 1.145. Onsite meteorological data from the period 1973 through 1976 were used to derive these χ/Q values. Further details on the calculation of the licensee's EAB and LPZ χ/Q values can be found in SONGS 2 and 3 UFSAR Section 2.3.4.1.

The NRC staff reviewed the licensee's use of existing SONGS 2 and 3 EAB and LPZ χ/Q values and has found them to be appropriate for the applications in which they are being used. On the basis of the review discussed above, the NRC staff concludes that the EAB and LPZ χ/Q values presented in Table 3 are acceptable for use in the design-basis offsite dose assessments performed in support of this license amendment request.

3.2 Radiological Consequences of DBAs

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences of the following three DBAs:

- Large break LOCA
- MSLB outside containment
- FHA in containment and in FHB

In accordance with the guidance provided in RG 1.183, the licensee reanalyzed the LOCA (required as a minimum for full implementation of the AST) and in addition, the FHA inside containment, the FHA in the FHB, and the MSLB accident outside containment. The remainder of the DBA dose analyses applicable to SONGS 2 and 3 were not reanalyzed by the licensee because no other changes were proposed to the SONGS 2 and 3 licensing or design basis that would require analysis of the radiological consequences of those remaining accidents. The licensee's submittal and its supplements reported the results of the radiological consequence analyses for the above DBAs to show compliance with 10 CFR 50.67, or fractions thereof, as defined in SRP Section 15.0.1, for doses offsite and in the CR. Therefore, this LAR is a full implementation of the AST.

The NRC staff performed confirmatory calculations to evaluate the licensee's dose analysis modeling, assumptions and results. The NRC staff used the licensee's analysis assumptions and inputs in its analyses, which were performed using version 3.03 of the computer code described in NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," and its supplements.

3.2.1 Loss-of-Coolant Accident (LOCA)

The current licensing basis radiological consequence analysis for the postulated LOCA is provided in SONGS 2 and 3 UFSAR Section 15.6.3.3, "Loss-of-Coolant Accident," and is based on the traditional accident source term described in TID-14844. To demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at SONGS 2 and 3 will remain adequate after implementing the AST as requested in this LAR, the licensee reanalyzed the offsite and CR radiological consequences of the postulated LOCA.

The licensee submitted the results of its DBA calculations for offsite and CR doses and provided the major assumptions and parameters used in its dose calculations. As documented in its submittal, as supplemented, the licensee has determined that after implementation of the AST, the existing ESF systems at SONGS 2 and 3 will continue to provide reasonable assurance that the radiological consequences of the postulated LOCA at the EAB, in the LPZ, and in the CR will meet the acceptable radiation dose criteria specified in 10 CFR 50.67(b)(2). As part of the implementation of the AST, the TEDE dose reference values of 10 CFR 50.67(b)(2) replace the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and also supplements the dose requirements in GDC 19.

SCE's analyses assumed that the inventory of fission products in the reactor core and available for release into the containment atmosphere is based on the maximum power level of 3,507 MWt, which is 1.02 times the current licensed thermal power level of 3,438 MWt in order to account for the emergency core cooling system (ECCS) evaluation uncertainty. The licensee developed the core inventory of fission products using the SAS2H and ORIGEN-S modules of the SCALE code package developed by the NRC. As discussed in RG 1.183, the NRC staff finds the use of isotope generation and depletion computer codes such as ORIGEN acceptable for developing the core inventory of fission products.

The NRC staff has reviewed the licensee's analyses for the following four potential fission product release pathways:

(1) primary containment leakage

(2) leakage from emergency core cooling systems (ECCSs) outside containment.

(3) RWST release

(4) post-accident sampling system leakage

3.2.1.1 Containment Leakage

The current SONGS 2 and 3 design-basis containment leak rate specified in the SONGS 2 and 3 TS and in the UFSAR is 0.1 percent of the containment-free volume per day (percent per day) at the containment design pressure of 60 pounds per square inch gauge (psig). For the radiological consequence analysis, this rate is reduced to 0.05 percent per day after 24 hours following a LOCA for the remaining duration of the accident (30 days), consistent with the guidance provided in RG 1.183. The licensee has not proposed to change the design basis containment leak rate.

3.2.1.1.1 Radioactivity Removal Inside the Containment

The fission products in the containment atmosphere following the postulated LOCA at SONGS 2 and 3 are mitigated by (1) natural deposition of fission products in aerosol form, and (2) removal by the containment spray system (CSS). SCE's analysis assumed removal of fission products in aerosol form by natural deposition in the containment following the postulated LOCA using Powers simplified natural deposition model in the RADTRAD dose consequences computer code described in NUREG/CR-6604 and its supplements. The Powers simplified natural deposition model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," The licensee used the 10 percentile confidence interval (90 percent probability) removal values implemented in the RADTRAD code. The Powers natural deposition model was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff finds that the use of this model in NRC computer code, RADTRAD, is acceptable, as discussed in RG 1.183. Aerosol removal rates by natural deposition developed and used by the licensee are shown in Table 4.5-3 of the licensee's March 10, 2006, submittal.

The licensee's analysis also assumed removal of elemental iodine by wall deposition using the methodology provided in SRP Section 6.5.2. Inputs to this methodology include a mass transfer coefficient, the wetted surface area inside containment, and the containment building net-free volume. The licensee used the mass transfer coefficient value of 4.9 meters per hour recommended by SRP Section 6.5.2. The NRC staff finds that the use of this methodology according to the guidance in SRP Section 6.5.2 is acceptable. The elemental iodine deposition removal rate value calculated and used by the licensee is 4.26 per hour.

The CSS at SONGS 2 and 3 is an ESF system. When used in conjunction with two containment emergency cooling units (ECUs), each rated at 31,000 cfm, and two dome air circulator units (DACUs), each rated at 37,000 cfm, the CSS is designed to ensure that containment pressure does not exceed the design-basis value of 60 psig and also to remove fission products in the containment atmosphere following the postulated LOCA. To meet the single failure criterion, only one of the two ECUs and one of the two DACUs are assumed to be operational for mixing of air in the containment. The licensee assumes that the ECU and DACU start operation 1 minute after the start of the LOCA. The licensee has determined that 99 percent of the contaminated air in the containment unsprayed region will be replaced with air from the sprayed region within 28 minutes, which equates to approximately four change-outs of the containment unsprayed region prior to the end of the activity releases from the core at conclusion of the early in-vessel phase at 1.8 hours. RG 1.183 assumes that two turnovers of the unsprayed region per hour provide adequate mixing. Therefore, the NRC staff concludes that the licensee has shown that the ECUs and DACUs provide adequate mixing between sprayed and unsprayed regions of the containment atmosphere in accordance with the guidance provided in RG 1.183.

As shown in the UFSAR, the CSS consists of two redundant and independent trains. Each train consists of a pump, spray headers, and associated valves, piping, and instrumentation. Each train receives power from a separate emergency diesel generator and separate actuation signals, and they are physically separated from each other. The CSS is automatically initiated by a containment spray actuation signal that is initiated by the combination of any two high-high containment pressure signals and a safety injection actuation signal. The CSS may also be initiated manually in the CR. The CSS has two phases of operation, an injection phase and a recirculation phase. During the injection phase, the CSS draws spray water from the RWST until it reaches a pre-set nominal low level. Following the injection phase, the CSS enters recirculation phase where spray water is drawn from the containment sump and recirculated through the CSS. The licensee stated that the sprayed volume of the containment is 83.5 percent of the total free volume of the containment.

The CSS is independently capable of delivering minimum flow rate of 1,606 gallons per minute (gpm) of borated water from the RWST during the injection phase and 1,991 gpm of sump water from the containment sump during recirculation phase into the sprayed region of the containment free volume. The licensee stated that it conservatively assumed a flow of 1,600 gpm throughout the CSS operation for determining the fission product removal coefficients by spray. The spray pumps are automatically started whenever two out of four high-high containment pressure signals occur or a manual signal is given. The licensee assumed that one out of two spray pump starts taking suction initially from the RWST and initiates building spray through the spray headers until the water in the RWST reaches a pre-set low level at 20 minutes after the postulated LOCA. This first 20 minutes of containment spray, taking suction from the RWST, is called the injection phase.

The licensee stated that, after the RWST reaches a preset low level at 20 minutes, the spray pump suction is transferred manually to the containment sump and the spray water from the containment sump is recirculated. The recirculation phase starts at 20 minutes and continues for the duration of the accident.

Radioactive iodine is released to the containment atmosphere in three different forms: elemental, particulate, and organic. For iodine in elemental and particulate forms, the licensee considered two distinct mechanisms by which radioactive iodine could be removed: containment sprays and natural deposition in the containment. There is no effective mechanism for removing organic iodine from the containment atmosphere. Removal of iodine in elemental and particulate forms from the containment atmosphere by the CSS is controlled by two types of parameters: those controlling the rates of removal, called removal coefficients or lambdas (λ), and those determining the maximum amount that can be removed, called decontamination factors (DF). The evaluation of the pH control in the containment sump to retain radioiodine in the sump water is addressed in Section 3.3 of this SE.

The removal rate of iodine by spray is a function of the volumetric flow of the spray solution, which is reflected in the corresponding spray removal coefficient λ. For SONGS 2 and 3, there are two different volumetric flow rates: one during the injection phase (1,991 gpm) and one during the recirculation phase (1,606 gpm). Therefore, there are different removal coefficients applied for the injection phase and the recirculation phase. In addition, there are separate sets of removal coefficients for elemental iodine and particulate iodine.

SCE calculated the elemental iodine spray removal coefficients in accordance with the methodology described in Section 6.5.2 of the SRP, which is acceptable to the NRC staff. The licensee calculated elemental iodine removal coefficients using a computer code that incorporates the model in SRP Section 6.5.2 and its basis document NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels." The licensee calculated elemental iodine spray removal rates for each spray ring in each of the two CSS spray headers. The calculations address the fact that each spray ring is at a different height than the others, with its own unique spray flux due to different coverage areas and number of spray nozzles per spray ring. The elemental iodine spray removal rates for each ring in a spray header were summed together, and the lowest header value at each time interval was used in the dose analysis. The licensee's calculations accounted for the changing containment sump water temperature and pH, volumes of the gaseous and liquid phases, and the initial iodine inventory in the containment sump water. The resulting elemental iodine spray removal coefficients and decontamination factors vary over time.

For the duration of the CSS injection phase, SCE calculated an elemental iodine spray removal coefficient of 1.02 per hour, with a decontamination factor of 110. For the CSS recirculation phase, the calculated removal coefficients decrease over time, and vary from 20 per hour to 3.78 per hour. Accordingly, the decontamination factors for the CSS recirculation phase also decrease over time, varying in value from 170 to 25. In accordance with SRP Section 6.5.2, Section III.4.c.(1), the elemental iodine spray removal coefficient is limited to a maximum value of 20 per hour. Therefore, although the licensee's calculation results exceeded that value early in the recirculation phase, the licensee used the limit value of 20 per hour in its dose calculations for those time periods. Because the licensee's calculated elemental iodine removal coefficients were based on the calculated time-dependent airborne aerosol mass, the reduction in removal rate by a factor of 10, when a DF of 50 is achieved, is not required, which is in accordance with RG 1.183, Appendix A, position 3.3. Table 4.5-5 of the licensee's March 10, 2006, submittal lists the licensee calculated elemental iodine spray removal rates per time period.

The licensee calculated the reduction in containment particulate iodine and aerosols by the CSS using the spray removal model for aerosols that was developed by Powers, et al., in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," which is referred to as the Powers spray model in this SE. Appendix A to RG 1.183 identifies the Powers spray model as acceptable to the NRC staff. The licensee used plant-specific input to the spray aerosol removal model with regard to the CSS spray water flux and the fall height of the spray droplets. The plant-specific spray water flux and fall height values are within the applicability range of values for which the Powers spray model is valid. For conservatism, to meet the single failure criterion, the licensee's analysis assumed that only one of the two CSS headers is in operation at a flow rate of 1,600 gpm, which is rounded down from the minimum flow rate during the injection phase for one train. SCE used the Powers spray model 10th percentile correlation, which minimizes the aerosol removal and is appropriately conservative. The licensee's calculated CSS spray aerosol removal coefficient values decrease over time, and vary from 5.15 per hour to 0.5 per hour. Table 4.5-4 of the licensee's March 10, 2006, submittal lists the licensee calculated aerosol spray removal rates per time period.

SCE combined the natural deposition and containment spray removal rates discussed above into an overall effective removal rate per time period for elemental iodine and aerosols for the unsprayed and sprayed regions of the containment. For example, for elemental iodine there is no separate spray removal rate and natural deposition rate for any one time period in the sprayed region of containment. The CSS is assumed to operate for 4 hours after the onset of the LOCA. After 4 hours, only natural deposition is modeled in the sprayed region of containment. Tables 4.5-6 and 4.5-7 of the licensee's March 10, 2006, submittal list the licensee calculated effective iodine and aerosol removal rates per time period.

Use of models for the various mechanisms for iodine removal, when more than one is used simultaneously for the same iodine species in a dose analysis, should consider the effect of one model on the others. Because each model used by the licensee does not account for removal through the other model, the use of both the referenced natural deposition models and spray removal models in the same (sprayed) region of containment for the same time period is recognized as potentially nonconservative. Although both natural deposition and spray removal are acting on the overall in-containment aerosol and elemental iodine source term, the total effect from both removal mechanisms is not the same as would be found by simply adding the removal coefficients for each model for a given time period together. SCE addressed this issue in its March 10, 2006, response to RAI #9. However, the modeled removal of particulate and aerosol iodine by natural deposition is not significant from a radiological consequence analysis perspective. The NRC staff did not find it necessary to have the licensee recalculate the doses using a modified natural deposition model to account for containment spray (or a modified containment spray model to account for natural deposition) because the adjustment to the overall LOCA doses would be negligible.

The NRC staff has reviewed the information provided by the licensee and compared the values used to the guidance in RG 1.183, Appendix A, and has determined that the licensee's assumptions for the containment leakage source term and transport are consistent with the guidance in RG 1.183. The NRC staff also performed an independent calculation of the dose consequences of this LOCA pathway using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results.

3.2.1.2 Post-LOCA Leakage from ESF Outside Containment

As described by the licensee, during the initial phases of the postulated LOCA, both safety injection system (SIS) and CSS draw borated water from the RWST. As early as 20.2 minutes into the accident, the recirculation mode of SIS and CSS starts for a two-train recirculation mode. For the recirculation mode, the spray pump suction is transferred manually to the containment sump and the spray water from the containment sump is recirculated. However, the licensee conservatively assumed the ESF leakage begins at 20 minutes after the start of the postulated LOCA event, representing the earliest possible recirculation start time with the ESF operating at maximum design capacity. The ESF leakage is assumed to continue for the duration of the 30-day LOCA event.

This recirculation flow causes contaminated sump water to be circulated through piping and components outside of the containment where a small amount of system leakage could provide a path for the release of radionuclides to the environment. Consistent with the guidance provided in RG 1.183, the licensee conservatively assumed that all of the radioiodines released from the reactor coolant system (RCS) are instantaneously moved to the containment sump water.

The licensee stated that the maximum expected leakage rate from all ESF components in the recirculation systems is 5,950 cubic centimeters per hour (cc/hr). SONGS 2 and 3 TS Section 5.5.2.8 requires establishment of a program which provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to a level as low as practicable. It further requires the program to include integrated leak test requirements for each system at refueling cycle intervals or less. In its radiological consequence analysis, the licensee doubled the maximum expected leakage rate of 5,950 cc/hr to 11,900 cc/hr consistent with the guidance provided in RG 1.183. The licensee's analysis assumed that 10 percent of the iodine in the ESF leaked fluid becomes airborne in accordance with guidance in Appendix A to RG 1.183.

SCE determined that there are three pathways for radioactivity release from ESF systems outside containment during a postulated LOCA. These release pathways are (1) leakage from ECCSs (2) leakage to the RWST during ESF recirculation, and (3) leakage from the post-accident sampling system (PASS).

3.2.1.2.1 Post-LOCA ECCS Leakage

As described by the licensee, the ECCS automatically delivers cooling water to the reactor core in the event of a LOCA. The ESF recirculation system circulates containment sump liquid for use by the CSS and the ECCS, i.e., high-pressure safety injection (HPSI) and low-pressure safety injection (LPSI). The containment sump liquid is circulated outside of the containment to the ESF pumps. In accordance with RG 1.183, Appendix A, Section 5, ESF systems that circulate sump water outside of the primary containment are assumed to leak during their intended operation. This release includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components.

Consistent with RG 1.183 guidance, the post-LOCA ESF leakage is assumed to start at the earliest time that recirculation flow begins in the ESF recirculation system. The maximum assumed leakage rate from the components in the HPSI, LPSI and CSS during recirculation is a total 5,950 cc/hr. The licensee assumed two times the maximum assumed leakage rate (i.e., 11,900 cc/hr) in the dose analysis, in accordance with guidance in RG 1.183. The licensee assumed that 10 percent of the iodine in the ESF leakage flashed to vapor and in available for release to the outside environment. This assumption of 10 percent is consistent with the guidance in RG 1.183, which states that if the water temperature is less than 212 degrees Fahrenheit (^{o}F), then 10 percent of the iodine in the leakage is assumed to become airborne unless a smaller amount is justified based on actual sump pH history and ventilation rates. The licensee's containment pressure and temperature response analysis for the LOCA event shows that the temperature of the containment sump liquid has been reduced to below 212 ^{o}F when the ESF recirculation mode of operation begins at 20 minutes.

In accordance with RG 1.183 guidance, with the exception of iodine, all radioactive materials in the recirculating fluid are retained in the liquid phase. The licensee further assumed that 100 percent noble gases formed by the decay of the isotopes in the ESF recirculated liquid will become airborne and available for release to the outside environment. This release of noble gases goes beyond the guidance of RG 1.183 and is an additional conservatism. The licensee assumed that iodine species in the airborne release from the ESF leakage were 97 percent elemental iodine and 3 percent organic iodine, in accordance with RG 1.183. The activity released from the ESF recirculation leakage is exhausted to the environment via the main plant vent.

The licensee calculated the doses for the EAB and LPZ for this release pathway and added the results to the EAB and LPZ doses from the other three pathways to give the total offsite radiological consequences of the LOCA. The licensee calculated the doses for the CR for this release pathway and added the results to the doses from the other three pathways and the direct shine dose to give the total radiological consequences of the LOCA in the CR.

The NRC staff has reviewed the information provided by the licensee and compared the values used to the guidance in RG 1.183, Appendix A, and has determined that the licensee's assumptions for the post-LOCA ECCS leakage source term and transport are consistent with, or more conservative than, the guidance in RG 1.183. The NRC staff also performed an independent calculation of the dose consequences of this LOCA pathway using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results.

3.2.1.2.2 Post-LOCA RWST Leakage

As described by the licensee, after 20 minutes post-LOCA, contaminated fluid is circulated outside of the containment via the SIS and CSS pumps during the recirculation mode of operation. The recirculated water may leak past the closed valves that isolate the RWST from the ESF systems. The licensee evaluated two scenarios, dependent on whether the LOCA occurs with or without diesel generator failure. The licensee determined that the scenario that does not assume diesel generator failure leads to more severe dose consequences both offsite and in the CR. Without a diesel generator failure, two ESF leakage pathways to the RWST are likely. The first pathway is flow into the air space of the RWST through the ESF pump minimum flow (mini-flow) isolation valve leakage. The second pathway is flow into the RWST, by backleakage, through the RWST discharge check valve. ESF leakage to the RWST for potential release paths with three or more normally closed isolation valves in series was assumed to be negligible by the licensee, which the NRC staff finds to be a reasonable assumption because of three normally closed valves in series.

The licensee used the guidance in RG 1.183 to perform the dose analysis of this leakage pathway. The licensee assumed that the ESF leakage to the RWST starts at the earliest time that recirculation starts, and continues for the remainder of the 30-day duration of the accident. Consistent with RG 1.183 guidance, the licensee multiplied by two the maximum allowable leak rate through the mini-flow isolation valves and the RWST discharge check valve to account for assumptions for flow into the air space of the RWST of 3 gpm and flow into the water in the RWST of 10 gpm, respectively.

For the 3 gpm that enters the air space of the RWST, the licensee assumed that 10 percent of the iodine in the leakage flashes to vapor, in accordance with RG 1.183 guidance. The licensee addressed this in the dose analysis by modeling 10 percent of the leakage entering the RWST air space, and the remaining 90 percent entering the RWST water space.

For the ESF leakage that has entered the RWST water space, including the 10 gpm from the discharge valve leakage and 90 percent of the 3 gpm from the mini-flow leakage, the licensee applied an iodine partition coefficient to determine the amount of the iodine in the leakage that enters the air space in the RWST and becomes available for release to the outside environment. The iodine partition coefficient is defined as the iodine concentration in the liquid divided by the iodine concentration in the gas. The licensee used a partition coefficient of 200, based on the NUREG/CR-0009 discussion of partition coefficients for borated solutions. The licensee accounted for the pH of the solution as well as air and water space mixing in the RWST due to turbulence from the mini-flow leakage that drops down into the water. No credit was taken by the licensee for mixing due to the RWST discharge valve flow or for mixing by thermal gradients in the water.

In accordance with RG 1.183 guidance, with the exception of iodine, all radioactive materials in the recirculating fluid are retained in the liquid phase. The licensee assumed that 100 percent of the noble gases formed by the decay of the isotopes in the RWST liquid will become airborne and available for release to the outside environment. This release of noble gases goes beyond the guidance in RG 1.183 and is an additional conservatism. The licensee assumed iodine species in the RWST air space were 97 percent elemental iodine and 3 percent organic iodine, in accordance with RG 1.183. The activity released from the RWST is exhausted to the environment through the RWST vent at a rate equal to the sum of the inflow to the RWST from the ESF pump mini-flow isolation valves and RWST discharge check valve.

The licensee calculated the doses for the EAB and LPZ for this release pathway and added the results to the EAB and LPZ doses from the other three pathways to give the total offsite radiological consequences of the LOCA. The licensee calculated the doses for the CR for this release pathway and added the results to the doses from the other three pathways and the direct shine dose to give the total radiological consequences of the LOCA in the CR.

The NRC staff has reviewed the information provided by the licensee and compared the values used to the guidance in RG 1.183, Appendix A, and has determined that the licensee's assumptions for the post-LOCA RWST leakage source term and transport are consistent with, or more conservative than, the guidance in RG 1.183. The NRC staff also performed an independent calculation of the dose consequences of this LOCA pathway using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results.

3.2.1.2.3 Post-LOCA PASS Leakage

The licensee stated that, at SONGS 2 and 3, the post-accident sampling system is maintained for severe accident management only. The PASS samples containment sump liquid, reactor coolant, and containment air. Portions of the PASS that are outside the containment provide a potential leakage pathway. The licensee's analysis assumed that the reactor coolant is the fluid in the PASS that leaks during the DBA LOCA. Reactor coolant has a greater iodine activity concentration than the containment sump liquid or containment air, therefore giving bounding dose results. Although RG 1.183 does not discuss leakage from the PASS in detail, the NRC staff is of the view that guidance given in the regulatory guide on ESF system leakage is applicable to the PASS leakage pathway analysis, because this guidance deals with reactor coolant leakage outside containment.

Consistent with RG 1.183 guidance, the licensee assumed that PASS leakage starts at 30-minutes post-LOCA, which is the earliest time that PASS sampling could start according to plant procedure. The PASS leakage is assumed to continue for the remainder of the 30-day LOCA duration. In the licensee's dose analysis, the PASS leakage rate is assumed to be 700 cc/hr, which is two times the maximum expected leakage from the PASS. The licensee assumes that 10 percent of the iodine in the PASS leakage flashes to vapor and is available for release to the outside environment. This 10 percent flashing assumption is consistent with the guidance in RG 1.183, which states that if the water temperature is less than 212 °F, then 10 percent of the iodine in the water leakage is assumed to become airborne. The PASS sample is cooled by the sample vessel heat exchanger to 120 °F to allow for low temperature sample analysis, and the majority of the leakage from the PASS sample station fittings will be at low temperature.

In accordance with RG 1.183 guidance, with the exception of iodine, all radioactive materials in the recirculating fluid are retained in the liquid phase. The licensee assumed that 100 percent of the noble gases formed by the decay of the isotopes in the PASS liquid will become airborne and become available for release to the outside environment. This release of noble gases goes beyond the guidance in RG 1.183 and is an additional conservatism. The licensee assumed iodine species in the airborne release from the PASS leakage were 97 percent elemental iodine and 3 percent organic iodine, in accordance with RG 1.183. The activity released from the PASS leakage into the radwaste building is exhausted to the atmosphere via the main plant vent.

The licensee calculated the doses for the EAB and LPZ for this release pathway and added the results to the EAB and LPZ doses from the other three pathways to give the total offsite radiological consequences of the LOCA. The licensee calculated the doses for the CR for this release pathway and added the results to the doses from the other three pathways and the direct shine dose to give the total radiological consequences of the LOCA in the CR.

The NRC staff has reviewed the information provided by the licensee and compared the values used to the guidance in RG 1.183, Appendix A, and has determined that the licensee's assumptions for the post-LOCA PASS leakage source term and transport are consistent with, or more conservative than, the guidance in RG 1.183. The NRC staff also performed an independent calculation of the dose consequences of this LOCA pathway using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results.

3.2.1.3 CR Habitability for LOCA

As described by the licensee, the SONGS 2 and 3 CRs are located within a common CRE. The technical support center is located within the CRE. Consistent with the guidance in RG 1.183, the licensee's dose analyses consider the sources of radiation that may cause exposure to CR personnel through intake or infiltration of contaminated air and radiation shine from radioactive material in the containment, release plume, and systems and components external to the CRE.

The CR emergency air cleanup system (CREACUS) is an ESF system. The CREACUS emergency mode of operation pressurizes the CR with filtered outside air and recirculates and filters the air within the CRE. The CREACUS emergency mode may be actuated either automatically following a CR isolation signal or manually. The CR isolation signal may be generated automatically by a safety injection actuation signal (SIAS) or by detection of high radioactivity concentrations in the CR outside air intake flow. The CREACUS is automatically actuated by the SIAS within 10 seconds of the onset of the LOCA, based on high containment pressure. Because the gap release activity is not assumed to be released into the containment until 30 seconds after the onset of the LOCA per RG 1.183, the CREACUS emergency mode of operation is credited from the beginning of the LOCA dose analysis. The NRC staff finds that the CREACUS operation assumption in the dose analysis is acceptable because the CREACUS would be in operation prior to the arrival of any contaminated air at any of the CR ventilation systems outside air intakes or the CRE boundary.

On June 12, 2003, the NRC staff issued Generic Letter (GL) 2003-01, "CR Habitability." This GL identified NRC staff concerns regarding the reliability of current surveillance testing to identify and quantify CR inleakage, and requested licensees to confirm the most limiting unfiltered inleakage into their CRE. In a letter dated August 5, 2003, the licensee submitted a "60-day" response to this GL for SONGS 2 and 3 that included a schedule for testing for the CRE unfiltered inleakage. By letter dated September 17, 2004, SCE provided the results of tracer gas testing performed during May of 2004. The AST dose analyses in the license amendment request reviewed here revised the radiological consequences analyses for both SONGS units to update the plant licensing and design basis with regard to CRE unfiltered inleakage.

Based on information provided by the licensee, tracer gas testing showed that for the CREACUS emergency mode of operation, the SONGS 2 and 3 CRE has 67 standard cubic feet per minute (scfm) of unfiltered inleakage for train A and 65 scfm for train B. Regulatory Position C.1.4 of RG 1.197 provides guidance indicating that it is optional to include the uncertainty for facilities that demonstrate a CRE inleakage rate less than 100 scfm. Since the tracer gas tested unfiltered inleakage rate value is less than 100 scfm, the licensee complies with the guidance of RG 1.197 when using 67 scfm in their calculations for one train operation. The licensee also determined that for the two-train operation of CREACUS, the unfiltered inleakage would be no more than the sum of the two single train measurements or 132 cfm, with an uncertainty of +/- 127 cfm. Consistent with RG 1.197, consideration of uncertainty indicates a maximum dual train unfiltered inleakage rate of 259 cfm. To add additional conservatism, the licensee’s dose calculations assumed a CRE unfiltered inleakage rate of 990 cfm (which is greater than the tracer gas tested value) to evaluate the accident conditions, and added an additional 10 cfm for ingress and egress. Based on these conservatisms, the NRC staff finds that the assumption of total unfiltered air inleakage of 1000 cfm in the licensee’s dose analysis is acceptable.

To model the CREACUS mode of operation, the licensee conservatively assumed a greater outside air intake rate than the nominal rate and a smaller CR recirculation rate than nominal. Filtration of particulates and iodine is credited for the emergency air conditioner (EAC) filters, but not for the emergency ventilation supply filters. The EAC filters are assumed to have 95 percent efficiency for elemental and organic iodine and 99 percent efficiency for particulate iodine and aerosols, consistent with the SONGS 2 and 3 current licensing basis and guidance in RG 1.183 and RG 1.52, as verified by SONGS 2 and 3 TS on filter testing. Major parameters and assumptions used by the licensee and found acceptable by the NRC staff for modeling the CR for DBA dose analyses are listed in Table 7.

The licensee also considered the dose in the CR due to gamma shine from the external radioactive plume (i.e., environmental cloud shine), radioactive material in the CREACUS filters in the CRE (i.e., CR filter shine), radioactive material in the containment (i.e., direct containment shine), and radioactive material in ESF recirculation loop piping outside the CRE (i.e., piping shine). These shine doses are added to the CR dose for the LOCA. The licensee conservatively modeled the source for each shine dose based on the LOCA modeling discussed above for immersion and inhalation. The activity in the CR filters is maximized by assuming that the filters are 100 percent efficient at removing iodine and particulates from the air. The licensee modeled the shielding between the CR dose receptors and the source, including the concrete structures and air space within the walls, floor, and ceilings that are between the source and receptor. The shine dose in the CR from each source is the maximum dose calculated at one of several dose receptors within the CR board area. The NRC staff reviewed the licensee's description of the shine dose calculations and compared them to the analyses currently in the SONGS 2 and 3 licensing and design basis. Based on its evaluation, the NRC staff finds the licensee's modeling assumptions to be reasonable, consistent with the shielding dose calculations previously performed by the licensee and, therefore, acceptable. The total shine dose results are a fraction of the doses due to immersion and inhalation, as listed in Table 4.5-11 of the licensee's letter dated March 10, 2006.

3.2.1.4 LOCA Radiological Consequences and Conclusion

The licensee re-evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria specified in 10 CFR 50.67. The results of the licensee's radiological consequence calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and found acceptable by the NRC staff are listed in Tables 4 and 4a.

The NRC staff performed an independent calculation of the dose consequences of LOCA using the licensee's assumptions and confirmed the licensee's dose results. The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those assumptions stated in the SONGS 2 and 3 UFSAR as design bases. Based on this, the NRC staff concluded that the LOCA radiological consequences are acceptable.

3.2.2 <u>Main Steamline Break</u>

The MSLB accident considered is the complete severance of the largest main steamline outside containment, downstream of an MSIV. Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for the MSLB. The radiological consequences of a MSLB outside containment will bound the consequences of a break inside containment. Thus, only the MSLB outside of containment is considered with regard to radiological consequences. Activity is introduced into the nuclear steam supply system (secondary side) through steam generator tube leakage, also called primary-to-secondary leakage. The licensee's dose analysis assumes a primary-to-secondary leakage rate into any single steam generator of 0.5 gpm, which is the maximum leak rate allowed by TS 3.4.13, and is consistent with RG 1.183 guidance. Activity is released to the outside environment through steaming and release through the steamline break, the MSSVs, the ADVs and the AFW turbine exhaust. The licensee assumed that the MSLB accident is terminated when shutdown cooling is initiated at 13,659 seconds, and all steam releases from both steam generators cease.

The SONGS 2 and 3 current licensing basis evaluates pre-trip and post-trip return to power for the MSLB. The pre-trip return to power may result in no more than 7 percent of the core experiencing fuel failure by damage to the cladding, and is the case evaluated for the DBA dose analysis. The post-trip break does not result in fuel failure. The licensee's analysis conservatively assumes 10 percent fuel failure to bound the fuel failure by the design-basis deterministic departure from nucleate boiling ratio (DNBR) prediction of 7 percent or the fuel failure determined by the proposed DNB statistical convolution methodology. The proposed methodology is addressed in Section 3.4 of this SE. The licensee applied a radial peaking factor of 1.75 to account for differences in power level across the core. Consistent with RG 1.183, Appendix E, because more than minimal fuel failure is postulated, the licensee's dose analysis does not include primary coolant iodine spiking. The initial primary coolant activity concentration is assumed to be at the maximum TS 3.4.16 limit of 1.0 μCi/gm dose equivalent I-131. The licensee assumes that the secondary coolant activity concentration prior to the accident is at the maximum TS 3.7.19 limit of 0.10 μCi/gm dose equivalent I-131.

Leakage from the RCS to the steam generators is assumed to be the maximum value permitted by TSs. Primary-to-secondary leakage is apportioned between faulted and intact steam generators in such a manner that the radiological consequence is maximized. The maximum TS limit for primary-to-secondary leakage to any one steam generator is 0.5 gpm. In accordance with RG 1.183 guidance, the licensee assumed that during periods when the steam generator tubes are covered, the primary-to-secondary leakage is mixed with the secondary water without flashing. The licensee used the CENTS computer code to analyze the secondary side response, including release masses and steam generator tube uncovery. The use of the CENTS code has been previously found acceptable by the NRC staff. The licensee's analysis determined that the tubes in one steam generator are uncovered from 17.3 seconds to 6,620 seconds after the break. The tubes in the other steam generator are uncovered from 17.2 seconds to 6,621 seconds after the break. The licensee conservatively assumed that the tubes in both steam generators are uncovered from 0 seconds to 6,621 seconds.

Consistent with RG 1.183, during periods when the steam generator tubes are uncovered, a portion of the primary-to-secondary leakage flashes to vapor based on the thermodynamic conditions in the reactor coolant and secondary coolant. The licensee conservatively assumed a flashing fraction of 20 during uncovery, which bounds the calculated flashing for the event. The flashed portion of primary-to-secondary leakage is available for release to the outside environment without further reduction for iodine scrubbing. The unflashed portion is assumed to mix with the bulk water. Consistent with RG 1.183, an iodine partition coefficient of 100 is applied to determine the amount of airborne iodine available for release to the environment. All noble gases released from the primary coolant are released to the environment without reduction.

The licensee's dose analysis modeled the SONGS 2 and 3 steam generators moisture carryover of 0.20 percent by assuming a particulate isotope partition coefficient of 500 to determine airborne particulate isotopes available for release to the environment. The release through the break begins at time zero and is terminated at 16.3 seconds when the MSIVs are fully closed. The dose analysis assumes a total mass release through the break of

115,103 pounds mass (lbm), which is 10 percent higher than the calculated mass release which consists of inventory loss from both steam generators and main feedwater flow for 3.83 seconds.

The MSSV and ADV mass releases assumed in the licensee's dose analysis are also increased by 10 percent over the calculated mass releases for the accident to provide analysis margin. The MSSV release is assumed to begin when the MSSVs open at 1,200 seconds and terminate when the MSSVs close at 1,822 seconds. The ADV release is assumed to begin when the ADVs are opened by operator action at 30 minutes and continue for the duration of the accident. The licensee's dose analysis assumed that the total mass release occurs through the MSSVs from 1,200 seconds until 1,800 seconds, and through the ADVs from 1,800 seconds until the end of the accident. The NRC staff finds this assumption to be conservative because the ADV dispersion factors are higher than the MSSV factors, which results in a higher dose.

For the time intervals during which the steam turbine AFW pump is operating, a release of activity occurs through that pathway. The licensee modeled two periods of AFW operation. The first is from 89 seconds to 748 seconds, following the postulated accident. The second is from 1,921 seconds to the end of the accident. The AFW steam turbine mass releases assumed in the licensee's dose analysis are also increased by 10 percent over the calculated mass releases for the accident to provide additional margin in the analysis results.

The licensee assumed that the CR is isolated at 180 seconds based on a high-radiation CR isolation signal. The high-radiation CR isolation signal is to be generated and the normal ventilation system outside air dampers would be closed within 120 seconds per the licensee-controlled Specification 3.3.100 in the Technical Requirements Manual of the UFSAR.. After 180 seconds, the CREACUS emergency mode of operation provides filtered pressurization and recirculation in the CRE as discussed above for the LOCA.

The licensee re-evaluated the radiological consequences resulting from the postulated MSLB accident using the AST and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria specified in 10 CFR 50.67 and the accident specific dose acceptance criteria specified in SRP Section 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the MSLB with fuel failure are a TEDE of 25 rem at the EAB for any two hours, 25 rem at the outer boundary of the LPZ and 5 rem in the CR for the duration of the accident. The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the SONGS 2 and 3 UFSAR as design bases.

The NRC staff also performed an independent calculation of the dose consequences of the MSLB using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The major parameters and assumptions used by the licensee and found acceptable to the NRC staff are presented in Table 5. The results of the licensee's radiological consequence calculation are provided in Table 1. The EAB, LPZ, and CR doses estimated for the MSLB meet the accident dose criteria in 10 CFR 50.67 and are, therefore, acceptable.

3.2.3 Fuel Handling Accidents

The licensee analyzed the radiological consequences of an FHA in two different locations; inside the containment, and in the FHB. RG 1.183, Appendix B, identifies acceptable radiological analysis assumptions for the FHA.

The FHA inside containment involves the inadvertent dropping of a fuel assembly during fuel handing operations inside the reactor vessel and the consequent rupture of fuel rods in both the dropped and impacted fuel assemblies. The number of fuel rods damaged during the accident is based on a conservative analysis that considers the most limiting case, as is discussed in the SONGS 2 and 3 UFSAR. Per the UFSAR, a maximum of 226 fuel rods will fail, which includes 16 rods in the dropped assembly and 210 rods in the impacted assemblies. This fuel failure assumption was previously found acceptable, and nothing in the current license amendment request affects this assumption.

Consistent with guidance in RG 1.183, the licensee applied a radial peaking factor of 1.75 to the average fuel rod isotope inventory to determine the activity inventory in each of damaged fuel rods. The dose analysis models 72 hours of radioactive decay, consistent with the minimum decay time required by SONGS 2 and 3 licensee controlled specification 3.9.101 prior to movement of irradiated fuel in the reactor vessel. The licensee's analysis used the RG 1.183, Table 3, fission product gap fraction values. The licensee stated that the fuel at SONGS 2 and 3 meets the burnup and maximum linear heat generation rate limitations for use of RG 1.183, Table 3, as given in the table's footnote.

Consistent with RG 1.183, the licensee assumed that the gap activity is instantaneously released into the refueling water. Because the depth of water above the damaged fuel is greater than 23 feet, the licensee applied an iodine effective decontamination factor of 200 per the RG 1.183 guidance. Retention of noble gases in the refueling water was assumed to be negligible, and particulate radionuclides are assumed to be retained by the refueling water. The licensee assumed that the iodine released from the water is composed of 57 percent elemental and 43 percent organic species, consistent with RG 1.183.

SONGS 2 and 3 TS 3.9.3 allows the containment personnel airlock to be open under certain conditions during core alterations and movement of irradiated fuel in containment. In addition, the licensee has submitted a license amendment request to allow the containment equipment hatch to be open during core alterations and movement of irradiation fuel in containment. Since the containment may be open during fuel handling operations, the activity that escapes the refueling water is assumed to be released to the environment over a 2-hour period without credit for containment closure. No credit is taken for activity dilution within the air of the containment dome space. Activity may be released to the environment through the containment purge system or by leakage through open containment penetrations, including the personnel airlock or equipment hatch. The CR atmospheric dispersion factors are different for each release point, and are discussed above in Section 3.1.2 of this SE. Since one set of atmospheric dispersion factors for a given source-receptor pair does not consistently give less dispersion than the others over time, the licensee assumed a composite set of limiting atmospheric dispersion factors to give the maximum dose result in the CR.

The FHA in the FHB involves the inadvertent dropping of a fuel assembly during fuel handing operations in the spent fuel pool and the consequent rupture of fuel rods in both the dropped fuel assembly. The number of fuel rods damaged during the accident is based on a conservative analysis that considers the most limiting case, as is discussed in SONGS 2 and 3 UFSAR, Section 15.7.3.4.2.2. Per the UFSAR, a maximum of 60 fuel rods will fail. This fuel failure assumption was previously found acceptable, and nothing in the current LAR affects this assumption.

Consistent with guidance in RG 1.183, the licensee applied a radial peaking factor of 1.75 to the average fuel rod isotope inventory to determine the activity inventory in each of damaged fuel rods. The dose analysis models 72 hours of radioactive decay, consistent with the minimum decay time required by SONGS 2 and 3 licensee controlled specification 3.9.101 prior to movement of irradiated fuel in the reactor vessel. The licensee's analysis used the RG 1.183, Table 3, fission product gap fraction values. The licensee stated that the fuel at SONGS 2 and 3 meets the burnup and maximum linear heat generation rate limitations for use of RG 1.183, Table 3, as given in the table's footnote.

Consistent with RG 1.183, the licensee assumed that the gap activity is instantaneously released into the spent fuel pool water. Because the depth of water above the damaged fuel is greater than 23 feet as assured by TS 3.7.16, the licensee applied an iodine effective decontamination factor of 200 per the RG 1.183 guidance. Retention of noble gases in the spent fuel pool water was assumed to be negligible, and particulate radionuclides are assumed to be retained by the spent fuel pool water. The licensee assumed that the iodine released from the water is composed of 57 percent elemental and 43 percent organic species, consistent with RG 1.183.

The licensee does not take credit for FHB closure for the FHA in the FHB or for operation of the FHB post-accident cleanup unit filter system. Consistent with RG 1.183, the licensee's dose analysis assumes that the activity that escapes the spent fuel pool is released to the environment over a 2-hour period. Activity may be released to the environment by the fuel handling normal ventilation exhaust system through the main plant vent or as leakage through the FHB penetrations such as doors. The CR atmospheric dispersion factors are different for each release point, and are discussed above in Section 3.1.2 of this SE. Since one set of atmospheric dispersion factors for a given source-receptor pair does not consistently give less dispersion than the others over time, the licensee assumed a composite set of limiting atmospheric dispersion factors to give the maximum dose result in the CR.

For both cases, the licensee assumed that the CR is isolated at 180 seconds based on a high-radiation CR isolation signal. The high-radiation CR isolation signal is to be generated and the normal ventilation system outside air dampers would be closed within 120 seconds per licensee-controlled Specification 3.3.100. After 180 seconds, the CREACUS emergency mode of operation provides filtered pressurization and recirculation in the CRE as discussed above for the LOCA.

The licensee re-evaluated the radiological consequences resulting from the postulated FHA inside the containment and the postulated FHA in the FHB using the AST and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria given in SRP 15.0.1 and

RG 1.183. These accident-specific dose acceptance criteria for the FHA are a TEDE of 6.3 rem at the EAB for any 2 hours and 6.3 rem at the outer boundary of the LPZ and 5 rem in the CR for the duration of the accident. The results of the licensee's radiological consequence calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and found acceptable by the NRC staff are listed in Table 6.

The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the SONGS 2 and 3 UFSAR as design bases. The NRC staff also performed an independent calculation of the dose consequences of the FHA using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results.

3.2.4 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the proposed full implementation of an AST and TS changes requested. 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 reasonable assurance that SONGS 2 and 3, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions, and parameters. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with the dose criteria in 10 CFR 50.67. Therefore, the NRC staff concludes that the proposed license amendment is acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the SONGS 2 and 3 design basis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in SRP 15.0.1. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the SONGS 2 and 3 design basis, and modified by the present amendment.

3.3 pH Control in the Containment Sump

3.3.1 Background

In its request for the approval of the license amendment for the application of the AST methodology for SONGS 2 and 3, the licensee included an analysis of its ability to maintain the pH of the sump water at or above 7 for a period of 30 days following a LOCA in order to minimize the amount of radioactive iodine released to the environment. The control of pH in the sump water is to retain radioiodine in the sump water, which effects the radioactivity removal inside containment, is addressed in Section 3.2.1.1.1 of this SE. This subsection reviews the methodology and supporting calculations provided by the licensee.

3.3.2 Evaluation

According to NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” the iodine entering the containment from the damaged core during an accident contains at least 95 percent cesium iodide (CsI). Upon its dissolution in sump water the iodine will be, therefore, predominantly in an iodide form (I^-). However, in the radiation field existing in the containment, some of this iodine will be converted into the molecular form (I_2). This conversion is strongly dependent on pH and will increase with the decreasing value of pH. Since molecular iodine is scarcely soluble in water, some of it will be released to the containment atmosphere and leak to the outside, contributing to the radiation doses. However, if the pH of sump water is maintained at or above 7, formation of the molecular iodine will be impeded and its release to the outside considerably reduced. At SONGS 2 and 3, this is achieved by adding to the sump water, TSP (trisodium phosphate - dodecahydrate, $Na_3PO_4 \bullet 12\ H_2O$). The TSP is a buffering agent and if a sufficient amount added, its buffering action will prevent pH from dropping below 7 with addition of strong acids. The two strong acids of concern that are generated in the containment and released to the containment sump water are hydrochloric acid, which is generated by radiolytic decomposition of Hypalon cable insulation and nitric acid, which is produced by irradiation of the water and air in the containment. Since the concentration of these acids is zero at the beginning of a LOCA and is building up with time, immediately after a LOCA, the TSP has to neutralize only boric acid which accumulates in the containment sump from the RCS, accumulators, and RWST. However, as the concentration of strong acids in the sump is increasing, more TSP is need to maintain pH basic. Therefore, the amount of TSP stored in the containment should be determined by the amount of total acids (boric and strong acids) present in the containment at 30 days after a LOCA.

The licensee calculated this amount using the Polestar STARpH code. The code is based on work performed at Oak Ridge National Laboratory and reported in NUREG/CR-5732, “Iodine Chemical Forms in LWR [Light-Water Reactor] Severe Accidents,” April 1992, and NUREG/CR-5950, “Iodine Evolution and pH Control,” December 1992. The code has been used for safety-related analyses of numerous operating plants under accident conditions. Using this code, the licensee determined that 2.0×10^4 pounds of TSP are required for maintaining basic or neutral pH throughout the 30-day post-LOCA period. In the plant, this TSP is stored in metal baskets located in the sump. After a LOCA, the water collected in the sump dissolves the TSP and forms a solution. Because of the buffering action of the TSP, the sump pH remains at an approximate value of 7, regardless of the concentrations of strong acids present. The NRC staff reviewed the licensee’s analysis and confirmed that the amount of TSP calculated by the licensee will maintain a neutral or basic environment in the sump during the 30-day post-LOCA period.

3.3.3 Summary

The NRC staff reviewed the licensee’s assumptions and calculations for controlling sump pH in order to maintain a neutral or basic environment (pH $\geq$7) in the containment sump. This condition is necessary to keeping dissolved iodine in the containment water for 30 days after a LOCA. The NRC staff’s review indicates that the assumptions and methodologies used by the licensee in its analysis are valid and consistent with the accepted methods, and adequately assure proper control of the sump water pH.

3.4 Departure from Nucleate Boiling (DNB) Statistical Convolution Methodology for Estimating Fuel Failure for non-LOCA Events

This methodology for estimating fuel failures is used in the calculation of the dose consequences for the MSLB in Section 3.2.2 of this SE.

3.4.1 Introduction and Background

In addition to its request to apply an AST methodology for SONGS 2 and 3, the licensee's LAR requested expanded use of fuel failure estimates by DNB statistical convolution methodology to all UFSAR Chapter 15 non-LOCA events that assume a loss of flow (i.e., a loss of alternating current (AC) power) and that fail fuel. The DNB statistical convolution methodology for estimating fuel failures in use at SONGS 2 and 3, is contained in CENPD-183-A, "C-E Methods for Loss of Flow Analysis." Currently, the licensee is only licensed to use the DNB statistical convolution methodology for the reactor coolant pump (RCP) sheared shaft event analysis.

3.4.2 Regulatory Evaluation

Part 50 of 10 CFR, Appendix A, GDC 17, "Electric power systems," prohibits anticipated operational occurrences (AOOs) initiated by or resulting in a loss of flow from exceeding any specified acceptable fuel design limit (SAFDL). Therefore, AOOs initiated by or resulting in a loss of flow are not permitted to result in fuel failures.

According to RG 1.183, Section 3.6, "[t]he amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached." The DNB statistical convolution methodology is an NRC-approved method for estimating the amount of fuel failures for non-LOCA design-basis events.

As previously stated, the licensee is currently only licensed to use the DNB statistical convolution methodology for its RCP sheared shaft event analysis. The proposed change would extend the use of the methodology to all non-LOCA events that assume a loss of flow (i.e., a loss of AC power) and that fail fuel.

3.4.3 Evaluation

DNB is the point at which the heat transfer from a fuel rod rapidly decreases due to the insulating effect of a steam blanket that forms on the rod surface when the temperature continues to increase. Fuel failure is conservatively assumed to occur whenever a fuel rod experiences DNB, even for a short time. To determine when a fuel rod would experience DNB, a DNBR SAFDL is established. DNBR is the ratio of the heat flux to cause DNB to the actual local heat flux of a fuel rod. The DNBR SAFDL is defined such that there is a 95-percent probability with a 95-percent confidence level that the fuel rod will not experience DNB whenever its DNBR is above the DNBR SAFDL.

Historically, a fuel rod was considered to experience DNB if its DNBR fell below the DNBR SAFDL. The number of failed fuel rods was the number with a DNBR below the DNBR SAFDL.

All fuel rods with a DNBR at or above the DNBR SAFDL were considered to not experience DNB and therefore did not fail.

The CENPD-183-A DNB statistical convolution methodology for estimating fuel failures makes use of the fact that the DNBR SAFDL is defined such that there is a 95-percent probability with a 95-percent confidence level that the fuel rod will not experience DNB whenever its DNBR is above the DNBR SAFDL. In other words, there is a 5-percent probability, with a 95-percent confidence level, that DNB will occur at the DNBR SAFDL. Extrapolating further, as DNBR increases above the DNBR SAFDL the probability decreases that DNB will occur, and as DNBR decreases below the DNBR SAFDL, the probability increases that DNB will occur.

The CENPD-183-A procedure for a DNB statistical convolution is to group fuel rods with respect to radial peaking factors; calculate the minimum DNBR in each radial peaking group; and determine the probability of experiencing DNB corresponding to a DNBR value. The number of fuel rods, within a radial peaking group, that are predicted to experience DNB and fail, is the product of the number of fuel rods in the radial peaking group and the probability of experiencing DNB associated with the corresponding minimum DNBR. The total number of fuel failures is the summation of each group's failed fuel.

The key to the use of the DNB statistical convolution methodology is developing the probability distribution of exceeding DNB with respect to DNBR. In the NRC SE report (SER) approving CENPD-183-A, the NRC staff stated:

> Since experimental evidence (Ref. 11) indicates that fuel cladding failure is not necessarily coincident with a short duration of DNB, we conclude that the statistical convolution technique is conservative and acceptable provided that the probability distribution for DNB is acceptable.

CENPD-183-A contained probability distributions for the Combustion Engineering (CE) 14x14 and 16x16 rod matrixes. Those distributions were established using the TORC computer code and the CE-1 heat flux correlation. The NRC SER for CENPD-183-A establishes a clear link between the computer code, critical heat flux correlation, and the probability distributions and requires NRC staff approval for any combination other than that specifically approved in the SER. The NRC SER states:

> The staff has reviewed the Loss-of-Flow [LOF] topical report CENPD-183. The computer codes used for the LOF analysis are acceptable for their assigned purposes. The COAST code can be used for transient system flow rate calculations, and the QUIX code can be used for axial power distributions and reactivity calculations. Either COSMO/W-3 or TORC/CE-1 can be used for steady state hot channel minimum DNBR calculations. The CESEC code is still under review and the comparison, to date, of its analysis with ANO-2 [Arkansas Nuclear One, Unit 2] startup test data indicates that CESEC is acceptable for LOF NSSS [Nuclear Steam Supply System] transient response analysis.
>
> The statistical convolution technique is acceptable for fuel rod failure calculations. The fuel damage probability distributions for CE-1 correlation are listed in Tables 2 and 3, respectively for the standard 16x16 and 14x14 fuel assemblies. If COSMO/W-3 is used

> for DNBR calculations, the applicant is required to submit a fuel damage probability distribution for staff's approval.
>
> In summary, the LOF analysis procedure using the static method of hot channel DNBR calculation is acceptable.

The LAR did not request the use of alternate computer codes or critical heat flux correlations, nor has the licensee provided sufficient information to warrant the approval to use alternate computer codes or critical heat flux correlations. A review of the licensee's UFSAR reveals that the CENPD-183-A listed computer codes are part of the licensee's current licensing basis and that TORC/CE-1 is being used for the DNB analysis for the single RCP sheared shaft transient.

The SER approving CENPD-183-A did not limit it to the sheared shaft event analysis. CENPD-183-A was intended for use with loss of forced flow (LOF) transients at CE plants. Further, the abstract for CENPD-183-A does not limit it to any particular LOF transient. In demonstrating the methodology, CENPD-183-A provided sample analysis for a four pump LOF coast down and a seized shaft LOF. Additionally, while the concept of a DNB statistical convolution methodology is approved in CENPD-183-A, the probability distributions provided therein are computer code, critical heat flux correlation, and CE fuel design 14x14 and 16x16 specific. Therefore, based on the review stated above, the NRC staff is approving the licensee's request to expand the use of of fuel failure estimates by DNB statistical convolution methodology to all USFAR Chapter 15 non-LOCA events that assume a loss of flow. However, the use of any combination of computer code, critical heat flux correlation, or fuel design, other than that explicitly approved by CENPD-183-A, will require submittal of revised probability distributions for NRC staff review and approval.

4.0 REGULATORY COMMITMENTS

In its letter dated December 27, 2004, the licensee has made the following regulatory commitments:

1. Following approval of this license amendment request, future revisions to UFSAR Chapter 15 design basis accident control room and offsite radiological consequence analyses will be performed using AST methodology.

2. Following approval of this license amendment request, the manual dose calculation methodology as described in Emergency Planning Implementation Procedures (EPIPs) and other Emergency Planning guidance documents will be revised to reflect AST methodology.

3. Raddose V dose assessment software will be evaluated by June 30, 2005, to determine what specific changes may be warranted in order to maintain consistency with the manual dose assessment calculation methodology.

The licensee, in an e-mail dated December 27, 2006 (to be added to ADAMS), stated that it completed an evaluation of Raddose V assessment software in April, 2006, to determine the changes necessary to implement the AST PCN. That evaluation identified 6 changes, listed below, needed prior to implementing the AST LAR.

1. Derive new SGTR Iodine removal factors for AST environment.
2. Update Technical Team Notebook.
3. Update 40.200 "don't use Source term capability in a SGTR."
4. Update 40.100 - "new Iodine removal factors."
5. Issue Emergency Planning Bulletin when AST implemented to appropriate Health Physics Emergency Response Organization members.
6. Place Note on RadDose V computers stating "don't use Source term capability in a SGTR."

The licensee sated that these changes will be made prior to implementation of the AST.

4. Following approval of this license amendment request, future revisions to Accident Monitoring setpoint calculations will reflect the AST.

5. Following approval of this license amendment request, SCE will provide the revised UFSAR sections to the NRC as part of its normal UFSAR update required by 10 CFR 50.71 (e).

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements 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 (70 FR 5248, dated February 1, 2005). 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.

7.0 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.

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Table 1

Licensee Calculated Radiological Consequences
TEDE (rem) (1)

DBA	EAB (2)	LPZ (3)	CR
LOCA	5.2	1.9	2.8
Dose acceptance criteria(4)	*25*	*25*	*5.0*
MSLB, 10% Fuel Failure	4.1	0.1	2.2
Dose acceptance criteria	*25*	*25*	*5.0*
FHA, Inside Containment	0.8	<0.1	0.3
Dose acceptance criteria	*6.3*	*6.3*	*5.0*
FHA, FHB	0.2	<0.1	<0.1
Dose acceptance criteria	*6.3*	*6.3*	*5.0*

(1) Total effective dose equivalent
(2) Exclusion area boundary
(3) Low-population zone, maximum 2-hour dose
(4) From RG 1.183 and SRP Section 15.0.1

Table 2

SONGS 2 and 3
CR Atmospheric Dispersion Factors

Release Pathway	DBA	χ/Q Value (sec/m^3)				
		0–2 hrs	2–8 hrs	8–24 hrs	1–4 days	4–30 days
Main Plant Vent	• LOCA ESF Recirc Loop Leakage • LOCA PASS Leakage • FHA–IC[a] • FHA–FHB[b]	1.15×10^{-3}	6.23×10^{-4}	2.14×10^{-4}	2.22×10^{-4}	2.02×10^{-4}
Containment Shell	• LOCA Containment Leakage • FHA–IC[a]	1.01×10^{-3}	6.41×10^{-4}	1.77×10^{-4}	2.36×10^{-4}	2.20×10^{-4}
Equipment Hatch	• FHA–IC[a]	8.01×10^{-4}	6.35×10^{-4}	1.78×10^{-4}	2.23×10^{-4}	2.03×10^{-4}
MSSV	• MSLB (1200 to 1822 sec)	1.22×10^{-3}	7.52×10^{-4}	2.48×10^{-4}	2.86×10^{-4}	2.60×10^{-4}
ADV	• MSLB (1800 sec to end of event)	3.70×10^{-3}	1.99×10^{-3}	6.95×10^{-4}	7.04×10^{-4}	6.34×10^{-4}
SLB-OC	• MSLB (0 to 16.3 sec)	7.78×10^{-3}	4.81×10^{-3}	1.62×10^{-3}	1.83×10^{-3}	1.68×10^{-3}
AFW Turbine Exhaust	• MSLB (89 to 748 sec) • MSLB (1921 sec to end of event)	8.60×10^{-4}	3.70×10^{-4}	1.56×10^{-4}	1.61×10^{-4}	1.30×10^{-4}
RWST	• LOCA RWST Release	5.67×10^{-4}	2.25×10^{-4}	8.84×10^{-5}	8.97×10^{-5}	7.37×10^{-5}
FHB	• FHA–FHB[b]	9.48×10^{-4}	7.61×10^{-4}	1.92×10^{-4}	2.65×10^{-4}	2.43×10^{-4}

[a]For the FHA–IC event, the highest χ/Q value for each time interval among the main plant vent, containment shell, and equipment hatch release pathways is used.

[b]For the FHA–FHB event, the highest χ/Q value for each time interval between the main plant vent and FHB release pathways is used.

Table 3

SONGS 2 and 3
EAB and LPZ Atmospheric Dispersion Factors

Receptor	Time Interval	χ/Q Value (sec/m^3)
EAB	**0–2 hrs**	2.72×10^{-4}
LPZ	**0–8 hrs**	7.72×10^{-6}
	8–24 hrs	4.74×10^{-6}
	1–4 days	3.67×10^{-6}
	4–30 days	2.67×10^{-6}

Table 4

Parameters and Assumptions Used in Radiological Consequence Calculations for LOCA

Parameter	Value
Reactor power, MWt	3,507
Containment volume, ft^3	
Total	2,284,000
Sprayed area	1,907,000
Unsprayed area	377,000
Containment leak rates, % per day	
0 to 24 hour	0.1
24 to 720 hours	0.05
Containment mechanical mixing rate, cfm	
Sprayed to unsprayed	31,000
Unsprayed to sprayed	31,000
Containment iodine and aerosol removal	Variable
Spray	Table 4a
Aerosol natural deposition	Powers 10^{th} percentile
Elemental iodine deposition, per hr	4.26
ESF recirculation volume, ft^3	46,647
ESF leak rates, cfm	
0 to 20 minutes	0
20 minutes to 30 days	0.007
RWST volume, ft^3	
Air space	35,880
Liquid space	7,345
RWST flow rates, cfm	
Mini-flow into RWST, total after 20 min	0.4010
Discharge check valve into RWST	
0 - 1.08 hr	0
1.08 - 2 hr	1.2859
2 - 8 hr	1.2778
8 - 24 hr	0.9622
24 - 96 hr	0.5103
96 - 119.72 hr	0.1078
119.73 - 720 hr	0

Table 4 (cont.)

Parameters and Assumptions Used in Radiological Consequence Calculations for LOCA

RWST flow rates, cfm (cont.)	
From RWST water to RWST air space	0.4010
From RWST air space to environment	
0 - 20 min	0
20 min - 1.08 hr	0.4010
1.08 - 2 hr	1.6869
2 - 8 hr	1.6788
8 - 24 hr	1.3632
24 - 96 hr	0.9113
96 - 119.72 hr	0.5088
119.72 - 720 hr	0.4010
RCS volume, ft^3	10,179
PASS leak rates, cfm	
0 - 30 min	0
30 min - 30 days	4.12E-04
Iodine flashing factors for leakage, %	
ESF recirculation	10
RWST mini-flow leakage to air space	10
PASS	10
RWST iodine partition coefficient	200
Atmospheric dispersion factors	Tables 2 and 3

Table 4a

Elemental Iodine and Aerosol Spray Removal Rates

Elemental Iodine Spray Removal Rates

Time Period (hr)	Elemental Iodine Spray Removal Rate (1/hour)
Prior to injection phase	0
During injection phase	1.02
Start of recirculation to 2 hr	20.00
2 to 4	20.00
4 to 8	18.86
8 to 13.8	16.01
13.8 to 24	12.99
24 to 48	10.09
48 to 96	7.83
96 to 720	3.78

Aerosol Spray Removal Rates

Time Period (hr)	Aerosol Spray Removal Rate (1/hour)
0 to 1.8	5.15
1.8 to 2	3.79
2 to 3.8	1.32
3.8 to 4	0.79
4 to 8	0.62
8 to 13.8	0.52
13.8 to 720	0.50

Table 5

Parameters and Assumptions Used in Radiological Consequence Calculations for MSLB Accident

Parameter	Value
Initial RCS activity, µCi/gm DEI-131	1.0
Secondary coolant activity, µCi/gm DEI-131	0.1
Failed fuel, percent of core	10
Radial peaking factor	1.75
Primary-to-secondary leakage per SG, gpm	0.5
RCS volume, ft^3	10,179
RCS liquid mass, gm	2.015E+08
Secondary coolant mass, lbm	1.59E+05
Shutdown cooling initiation time, sec	13,659
Steamline break mass release, lbm	
0 to 16.3 sec	115,103
16.3 to 13,659 sec	0
Steam release from MSSV, lbm	
0 to 30 min	47,553
30 min to 2 hr	555.5
2 hr to 13,659 sec	0
Steam release from ADV, lbm	
0 to 30 min	8,078
30 min to 2 hr	64,522
2 hr to 13,659 sec	78,944
Steam release from AFW steam turbine, lbm	
0 to 30 min	47,553
30 min to 2 hr	555.5
2 hr to 13,659 sec	0
SG flashing factors and partition coefficients	
SG tube uncovery period, sec	0 to 6,621
Iodine flashing factor during uncovery, %	20
Iodine partition coefficient	100
Particulate partition coefficient	500
Atmospheric dispersion factors	Tables 2 and 3

Table 6

Parameters and Assumptions Used in Radiological Consequence Calculations for FHAs

Parameter	Value
Number of failed fuel rods	
FHA inside containment	226
FHA in FHB	60
Fraction of Core Inventory in Gap	
Kr-85	0.10
I-131	0.08
Alkali metals	0.12
Other noble gases / iodines	0.05
Decay time after reactor shutdown, hr	72
Radial peaking factor	1.75
Minimum water depth above damaged fuel, ft	
Reactor vessel	23
Spent fuel pool	23
Pool effective iodine decontamination factor	200
Iodine species above water, % of iodine	
Elemental iodine	57
Organic iodine	43
Release duration, hr	2
FHA in containment	
Containment closure	Not credited
Release point	Bounding for containment openings
FHA in FHB	
FHB closure	Not credited
Release point	Bounding for FHB
Atmospheric dispersion factors	Tables 2 and 3

Table 7

Parameters and Assumptions Used in Radiological Consequence Calculations for CR Habitability

Parameter	Value
CR net free volume, ft^3	26,920
CR total unfiltered inleakage, cfm (Includes 10 cfm ingress/egress)	1,000
CR ventilation normal mode	
Unfiltered outside air intake, cfm	6402
CREACUS emergency mode initiation	
LOCA (SIAS-induced), sec	0
MSLB, FHA (high-radiation), sec	180
CREACUS emergency mode, one train	
Filtered outside air intake, cfm	2,200
Filtered recirculation, cfm	29,934
CREACUS emergency mode, two trains	
Filtered outside air intake, cfm	4,400
Filtered recirculation, cfm	59,869
CREACUS emergency ventilation supply filters	not credited
CREACUS emergency air conditioner filter efficiency	
Elemental iodine, %	95
Organic iodine, %	95
Particulate iodine and aerosols, %	99