UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 25, 2009

Mr. Ross T. Ridenoure
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: TECHNICAL SPECIFICATION CHANGES IN SUPPORT OF STEAM GENERATOR REPLACEMENT (TAC NOS. MD9160 AND MD9161)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. NPF-10 and Amendment No. 213 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 27, 2008, as supplemented by letters dated August 13, 2008, and February 5, 2009.

The amendments revise TSs 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.2.11, "Steam Generator (SG) Program," 5.5.2.15, "Containment Leakage Rate Testing Program," and 5.7.2.c, "Special Reports," and support of the replacement of the steam generators at SONGS, Units 2 and 3.

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,

James R. Hall

James R. Hall, Senior 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. 220 to NPF-10
2. Amendment No. 213 to NPF-15
3. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
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 June 27, 2008, as supplemented by letters dated August 13, 2008, and February 5, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 220, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to entry into Mode 4 during the Unit 2 Cycle 16 refueling outage return-to-service.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. NPF-10
and Technical Specifications

Date of Issuance: June 25, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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

Facility Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.4-51	3.4-51
3.4-52	3.4-52
5.0-13	5.0-13
5.0-15	5.0-15
5.0-15a	-------
5.0-16	5.0-16
5.0-20a	5.0-20a
5.0-29	5.0-29

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

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

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

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 2 and by the decommissioning of San Onofre Nuclear Generating Station Unit 1.

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

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 220, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
OR	AND	
SG tube integrity not maintained.	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.8 Primary Coolant Sources Outside Containment Program (continued)

system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path). The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.2.9 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containment, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. Program itself is relocated to the LCS.

5.5.2.10 Inservice Inspection and Testing Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components and Code Class CC and MC components including applicable supports. The program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program itself is located in the LCS.

5.5.2.11 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.

In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

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5.5 Procedures, Programs, and Manuals (continued)

5.5.2.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exception:

NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the March 31, 1995 Type A Test shall be performed no later than March 30, 2010.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 48.0 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage rate acceptance criteria are:

a. The Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 9.0 psig.

(continued)

5.7 Reporting Requirements (continued)

5.7.2 Special Reports (continued)

1. The scope of inspections performed on each SG,
2. Active degradation mechanisms found,
3. Nondestructive examination techniques utilized for each degradation mechanism,
4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
5. Number of tubes plugged during the inspection outage for each active degradation mechanism,
6. Total number and percentage of tubes plugged to date,
7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
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 June 27, 2008, as supplemented by letters dated August 13, 2008, and February 5, 2009, 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.

Enclosure 2

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 213, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to entry into Mode 4 during the Unit 3 Cycle 16 refueling outage return-to-service.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. NPF-15
and Technical Specifications

Date of Issuance: June 25, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 213

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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

Facility Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.4-51	3.4-51
3.4-52	3.4-52
5.0-13	5.0-13
5.0-15	5.0-15
5.0-15a	-------
5.0-16	5.0-16
5.0-20a	5.0-20a
5.0-29	5.0-29

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

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

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

(6) SCE, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of San Onofre Nuclear Generating Station, Units 1 and 3 and by the decommissioning of San Onofre Nuclear Generating Station Unit 1.

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

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 213, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5.2.8 Primary Coolant Sources Outside Containment Program (continued)

system (post-accident sampling return piping only until such time as a modification eliminates the post-accident piping as a potential leakage path). The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.2.9 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containment, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. Program itself is relocated to the LCS.

5.5.2.10 Inservice Inspection and Testing Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components and Code Class CC and MC components including applicable supports. The program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program itself is located in the LCS.

5.5.2.11 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.11 Steam Generator (SG) Program (continued)

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 35% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.

In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

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5.5.2.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 as modified by the following exception:

NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the September 10, 1995 Type A Test shall be performed no later than September 9, 2010.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 48.0 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage rate acceptance criteria are:

a. The Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 9.0 psig.

(continued)

5.7 Reporting Requirements (continued)

5.7.2 Special Reports (continued)

1. The scope of inspections performed on each SG,
2. Active degradation mechanisms found,
3. Nondestructive examination techniques utilized for each degradation mechanism,
4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
5. Number of tubes plugged during the inspection outage for each active degradation mechanism,
6. Total number and percentage of tubes plugged to date,
7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated June 27, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081830421), as supplemented by letters dated August 13, 2008, and February 5, 2009 (ADAMS Accession Nos. ML082280080 and ML090400654, respectively), Southern California Edison Company (SCE or the licensee), submitted a license amendment request for changes to the Technical Specifications (TSs) for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The supplemental letters dated August 13, 2008, and February 5, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination, as published in the *Federal Register* on September 23, 2008 (73 FR 54867). The February 5, 2009, supplemental letter provided the licensee's response to the NRC staff's requests for additional information (RAIs) dated November 4 and December 8, 2008 (ADAMS Accession Nos. ML083050122 and ML083170553, respectively).

The proposed changes would revise TSs 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.2.11, "Steam Generator (SG) Program," 5.5.2.15, "Containment Leakage Rate Testing Program," and 5.7.2.c, "Special Reports." The proposed changes reflect revised steam generator (SG) inspection and repair criteria and revised peak containment post-accident pressures resulting from the planned installation of the replacement steam generators (RSGs) at SONGS, Units 2 and 3.

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in the licensee's June 27, 2008, submittal and its supplements. The detailed evaluation below supports the conclusion 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

2.0 REGULATORY EVALUATION

Revised Steam Generator Inspection and Repair Criteria

Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50) establishes the fundamental regulatory requirements with respect to the integrity of the steam generator (SG) tubing. SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 establish more specific design requirements that apply to SG tubing, as follows:

- GDC 14, "Reactor coolant pressure boundary," states that the RCPB shall have "an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- GDC 15, "Reactor coolant system design," states that the reactor coolant system (RCS) "shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

- GDC 30, "Quality of reactor coolant pressure boundary," states that "Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards possible."

- GDC 31, "Fracture prevention of reactor coolant pressure boundary," states that the RCPB "shall be designed with sufficient margin to assure that...(1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized."

- GDC 32, "Inspection of reactor coolant pressure boundary," states that "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."

The regulations in 10 CFR 50.55a, "Codes and standards," specify that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized-water reactor (PWR) facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. ASME Code, Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional SG tube surveillance requirements in the TSs for SONGS, Units 2 and 3.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing that may occur during these events, and the analyses must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100, "Reactor site criteria," guidelines for offsite doses (or 10 CFR 50.67, "Accident source term," as appropriate); GDC 19, "Control room," criteria for control room operator doses, or some fraction thereof as appropriate to the accident; or the NRC-approved licensing basis.

The SONGS, Units 2 and 3 TSs are modeled after TS Task Force (TSTF) change traveler TSTF-449, Revision 4, "Steam Generator Tube Integrity." TS 5.5.2.11 for SONGS requires that an SG program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity consistent with the plant design and licensing bases. TS 5.5.2.11 requires a condition monitoring assessment to be performed during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met. TS 5.5.2.11 also includes provisions regarding the scope, frequency, and methods of SG tube inspections. The staff also considered the guidance of NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976, in its evaluation of the application.

Containment Analysis

The Updated Final Safety Analysis Report (UFSAR) for SONGS, Units 2 and 3 states that the design of these units conforms with the GDC of 10 CFR Part 50, Appendix A, with no exceptions (other than NRC-approved exceptions). The 10 CFR 50, Appendix A criteria applicable to the proposed changes to the containment analysis are listed below:

- GDC 16, "Containment design," requires that the containment and its associated systems (e.g., penetrations) be "provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

- GDC 38, "Containment heat removal," requires that the reactor containment be provided with a system to "reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

- GDC 50, "Containment design basis," requires that the containment and its penetrations "accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident."

Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," specifies leakage test requirements for periodic testing of the containment and its penetrations. As discussed in TS 5.5.2.15, the SONGS Containment Leakage Rate Testing Program utilizes Option B to Appendix J for the Type A, B, and C containment leakage testing.

In addition, the NRC staff evaluated the proposed changes using the guidance of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," dated March 2007; specifically, SRP Section 6.2.1, "Containment Functional Design," and SRP Section 6.2.2, "Containment Heat Removal Systems."

3.0 TECHNICAL EVALUATION

SONGS, Units 2 and 3 are two-loop Combustion Engineering (CE) PWRs, with each loop consisting of two reactor coolant pumps and an SG. The current SGs (referred to as original SGs or OSGs) are also designed and manufactured by CE. The licensee is planning to replace the OSGs with new SGs (referred to as replacement SGs or RSGs) designed and manufactured by Mitsubishi Heavy Industries, Ltd. (MHI). The replacement SGs are scheduled to be installed during the SONGS, Unit 2 fuel cycle 16 refueling outage, currently scheduled to begin in October 2009, and the SONGS, Unit 3 fuel cycle 16 refueling outage, currently scheduled to begin in October 2010. In support of the SG replacements, the licensee is proposing changes to the TSs for SONGS, Units 2 and 3 associated with SG inspection and repair, and peak containment post-accident pressure. The TS sections affected by the SG inspection and repair criteria are 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.2.11, "Steam Generator (SG) Program," and 5.7.2.c, "Special Reports." The TS section affected by the containment post-accident pressure is 5.5.2.15, "Containment Leakage Rate Testing Program."

3.1 Revised SG Inspection and Repair Criteria

SONGS, Units 2 and 3 currently have CE designed and manufactured Model 3410 SGs. Each SG contains 9350 mill-annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.75 inches and nominal wall thickness of 0.048 inches. The tubes were explosively expanded into the tubesheet.

The NRC has approved several previous amendments related to the original SONGS, Units 2 and 3 SGs. The licensee is currently permitted to repair tubes by sleeving and is permitted to limit the extent of inspection in the tubesheet region, thereby allowing flaws to remain in service based on a methodology referred to as C* (C-star).

The replacement SGs manufactured by MHI differ from the existing SGs in that the tube material is thermally-treated Alloy 690 in the replacement SGs, versus the mill-annealed Alloy 600 in the existing SGs. In addition, the tubes are hydraulically expanded into the tubesheet in the RSGs, instead of explosively expanded in the existing SGs.

The licensee is proposing to remove the TS requirements associated with the C-star methodology, since this methodology is based on the design of the tube-to-tubesheet joint in the original SGs (i.e., an explosively expanded joint), and does not apply to the replacement SGs. In addition, the licensee is proposing to remove the TS requirements associated with sleeving, since these requirements were developed based on the tube material (Alloy 600) of the original SGs. The affected TS requirements are contained in TS 3.4.17 (limiting condition for operation and associated action and surveillance requirements), TS 5.5.2.11.a (condition monitoring), TS 5.5.2.11.c (tube repair criteria), TS 5.5.2.11.d (tube inspections), TS 5.5.2.11.f (tube repair methods), and TS 5.7.2 (reporting requirements). The changes would remove references to the term "repair," and other requirements, as they relate to the use of the sleeving method. Those portions of the affected TSs describing criteria applicable to the inspection and repair (i.e., plugging) of tubes for the RSGs will be retained.

The NRC staff concludes that these changes are acceptable, since the requirements to be removed were developed for the original SGs at SONGS, Units 2 and 3. With the planned replacement of the original SGs, these requirements for sleeving repairs and use of the C-star methodology are no longer needed. In addition, given the design differences between the original and replacement SGs, these requirements are not applicable to the replacement SGs. Further, the licensee is proposing to replace the current SONGS SG inspection requirements with those requirements specific to SGs with thermally-treated Alloy 690 tubes, the material used in the replacement SGs. The staff finds these proposed changes to revise the inspection requirements acceptable, since the licensee's replacement SGs have thermally-treated Alloy 690 tubes, and the proposed changes are consistent with TSTF-449.

Lastly, the licensee has proposed to revise the tube repair (plugging) criterion in TS 5.5.2.11.c. The criterion for requiring a tube with an identified flaw to be plugged will be reduced from a flaw depth of 44 percent of the nominal wall thickness to 35 percent of the nominal wall thickness. The NRC staff concludes that the proposed change is acceptable, since the licensee indicated that this tube repair criterion was determined using the methodology in NRC Regulatory Guide 1.121, and such a repair criterion is generally consistent with that used at other similarly designed and operated plants.

In summary, the NRC staff finds that the proposed changes to the TS requirements for SG inspection and repair criteria for SONGS, Units 2 and 3 are acceptable, since the revised TSs are consistent with the staff positions in TSTF-449, and are appropriate for the new tube material in the replacement SGs. These changes provide reasonable assurance that the licensee can implement an effective SG inspection and repair program such that the replacement SGs will be capable of performing their design function to maintain the integrity of the RCPB, in accordance with the requirements of 10 CFR 50.55a and GDCs 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A.

3.2 Containment Analysis

With respect to the revised containment analysis associated with the replacement SGs, the licensee stated that the following changes to the SG design affect the calculation of post-accident containment pressure:

- Additional RSG primary coolant inventory due to larger SG tube bundle.
- Additional RSG secondary side inventory.
- Changes to the SG tube heat transfer area.
- Use of a flow restrictor in the RSG steam outlet nozzle.

The licensee performed mass and energy release, and containment response analyses for the design-basis LOCA and MSLB accident to determine a revised value for the limiting containment post-accident pressure.

3.2.1 LOCA Containment Analysis

The containment is the final barrier against the release of fission products in the event of an accident. Design-basis events are analyzed to demonstrate that the containment structure can withstand the pressure and temperature conditions resulting from a LOCA or an MSLB inside the containment, and that the equipment needed to mitigate these events remains functional during and after the events. The containment is provided with automatic protective features and engineered safety feature (ESF) systems to accomplish this function. When the containment pressure exceeds the containment high-pressure setpoint (5.0 pounds per square inch gauge (psig) analysis value, plus a time delay for actuation signal processing), a safety injection actuation signal (SIAS), containment isolation actuation signal (CIAS), and a containment cooling actuation signal (CCAS) occur. In addition, the automatic reactor protection system (RPS) trips the reactor. The SIAS adds borated water to the RCS and initiates a 10-second (± 2.5 second) sequencer to start the containment spray pumps. The CIAS isolates the non-essential lines penetrating the containment and closes the main steam isolation valves (MSIVs), the main feedwater isolation valves (MFIVs), and the main feedwater block valves. The CCAS actuates the containment emergency air cooling units (ECUs). When containment pressure exceeds the containment high-high (CPHH) setpoint (20 psig analysis value, plus a time delay for signal processing), a containment spray actuation signal (CSAS) initiates the opening of the containment spray isolation block valves. The safety injection system (SIS) and the containment spray system (CSS) initially take suction from the refueling water tank (RWT). When the water level in the RWT reaches a certain low level, a recirculation actuation signal (RAS) is generated, at which time the source of water for the safety injection pumps and containment spray pumps transfers to the containment sump. In addition, component cooling water (CCW) is directed to the shutdown cooling heat exchangers (SDCHX) to cool the containment spray water.

The revised analysis modifies the input parameters to reflect the RSGs and updates any other applicable parameters due to changes that had occurred at SONGS, Units 2 and 3 since the previous evaluation. The licensee stated that the previous post-accident containment analysis was reviewed by the NRC staff as part of SONGS, Units 2 and 3 Amendment Nos. 182 and 173, dated January 24, 2002 (ADAMS Accession No. ML013340271).

The LOCA event is assumed to be initiated with the reactor operating at full thermal power (including measurement uncertainty), and is analyzed in four distinct phases known as blowdown, reflood, post-reflood, and long-term cooldown.

3.2.1.1 Blowdown Phase

The licensee simulated the blowdown phase of the LOCA using the NRC staff-approved CEFLASH-4A code methodology. The mass and energy release calculations with the RSGs were done in general accordance with the performance analyses methods of 10 CFR 50, Appendix K, "ECCS [Emergency Core Cooling System] Evaluation Models," with additional conservatism to maximize the release to containment, consistent with the current analysis described in UFSAR Section 6.2.1.3. The additional conservatisms assumed by the licensee are as follows:

- To maximize the energy available for release from the core, the Appendix K model for fuel clad swelling and rupture was not considered.
- To enhance the energy transfer from core to coolant, calculations of heat transfer assumed nucleate boiling even if conditions may warrant departure from nucleate boiling, except for conditions of single-phase steam.
- RCS volume was conservatively calculated based on the expansion of the RCS loop from cold to hot operating conditions at rated thermal power, plus uncertainty.
- Main feedwater addition was included to account for the hot feedwater added to the SGs.
- To maximize the severity of the blowdown, a simplistic core nodalization was used.
- Decay heat used was the American Nuclear Society (ANS) Standard ANS 5 – 1971 (+20 percent) decay heat standard.

3.2.1.2 Reflood and Post-Reflood Phases

The LOCA mass and energy analyses for the core reflood and post-reflood phases were performed using the FLOOD3 computer code, consistent with methodology described in UFSAR Section 6.2.1.3.4. Following initial blowdown, the reactor is first filled by the incoming safety injection flow, including safety injection tanks (SITs), and then reflooded as the core becomes quenched. However, the refill phase, which is the time period to fill the reactor vessel to the bottom of the active core, is conservatively omitted for the containment calculations, as provided in the review guidelines of Section 6.2.1.3 of NUREG–0800. The reflood phase is defined as the time period during which the coolant in the reactor vessel accumulates to 2 feet below the top of the core. The end of post-reflood occurs when the RCS and SGs are essentially in temperature equilibrium, at which time the core is considered to be quenched and the liquid entrainment reduces significantly. To maximize the rate of energy release to the containment, the following conservatisms were included:

- Heat transfer from the core to RCS is considered to be always in the nucleate boiling regime.

- Decay heat model was the ANS 5 – 1971 (+ 20 percent) decay heat standard, same as in blowdown phase.

- A Carryout Rate Fraction (CRF) of 1.0 was modeled after the core level increases above the elevation corresponding to 2 feet below the top of the active core. This is a conservative deviation from Section 6.2.1.3 of NUREG-0800, which states that a CRF of 0.05 may be used.

- The model assumes that only 50 percent of the steam condenses for the time period after the annulus is full and the SITs are injecting, although test data indicates that significantly greater condensation of steam occurs at the safety injection location.

The FLOOD3 code is not used for hot-leg breaks, since there is no viable means for the exiting break flow to pass through the SGs prior to exiting to containment. Therefore, the mass and energy analysis ends at the end of blowdown phase.

3.2.1.3 Long-Term Cooldown Phase

The LOCA mass and energy analysis for the long-term phase makes use of the COPATTA and CONTRANS codes, consistent with the current licensing basis methodology described in UFSAR Section 6.2.1.3.5. During this phase, COPATTA also calculates the mass and energy release data in parallel with the transient containment pressure/temperature calculation. The long-term phase follows the blowdown mass and energy release calculation for the hot-leg LOCA and post-reflood mass and energy release calculation for the cold-leg LOCA. The CONTRANS code is used to calculate the residual heat addition from primary and secondary metal and the SG inventory during the long-term phase. The time-dependent energy addition due to this sensible heat was input to the COPATTA code and added to the reactor vessel or directly to the atmosphere. Consistent with the current licensing basis, the decay heat input to COPATTA during the long-term cooling phase was based on Branch Technical Position (BTP) ASB 9-2 in Section 9.2.5 of NUREG-0800.

3.2.1.4 LOCA Containment Response

As stated in UFSAR Section 6.2.1.1.3.c, the COPATTA code predicts both the pressure and temperature within the containment regions and the temperatures in the containment structures. The code models the containment and the heat transfer surfaces following design-basis accidents. The model includes ESF system parameters and analytical techniques that enable calculation of the effects upon the containment.

The licensee's revised model also incorporated updated parameters not related to the RSGs. These parameters are provided in the licensee's letter dated February 5, 2009, in response to the staff's RAI dated November 4, 2008. Significant changes are the reduction of CCW flow to the SDCHX, reduction in CSS flow rate, reduction in CCW flow rate to the ECUs, additional delay in starting CCS, and heat sinks. These updated parameters reflect implemented or planned changes to address operational concerns, including flow-induced vibration limits, future flow rate degradation, post-RAS flow alignments, potential variations in emergency diesel generator (EDG) starting time delay, and heat sink changes.

3.2.1.5 Break Locations and Single Failures

Consistent with the current licensing basis as described in UFSAR Section 6.2.1.1, all breaks analyzed were double-ended slot breaks. The break locations are the reactor coolant pump discharge and suction legs (i.e., cold legs) and the RCS hot leg. Consistent with UFSAR Section 6.2.1.1.1, offsite power was assumed to be lost at the initiation of the LOCA to maximize the delay in starting the containment heat removal systems. The first single failure considered is the failure of an EDG to start, resulting in the failure of one train of containment spray, one train of ECUs, and one train of the safety injection system. This failure results in one train of safety injection flow available. The second failure assumed a failure of either one train of containment spray or one train of ECUs. The second failure results in two trains of safety injection flow available. The COPATTA results show that the failure of one EDG is limiting.

3.2.1.6 LOCA Results

The maximum containment pressure and temperature results for all LOCA cases analyzed are shown in Table 4.2-2 of the licensee's evaluation provided in its application dated June 27, 2008. The maximum pressure calculated was 48.0 psig for a double-ended slot break in the hot leg with an assumed failure of one EDG. The licensee provided the pressure and temperature profile for the limiting case in Figure 4.2-1 of the enclosure to the application. The containment pressure and temperature peak at 48.0 psig and 273 degrees Fahrenheit (°F) at 16 seconds, which is before the containment spray begins operation. The profile also indicates that at 24 hours post-LOCA, the pressure has dropped to below half the peak pressure.

3.2.2 <u>MSLB Containment Analysis</u>

The location of the postulated break for the MSLB event is at one of the RSG outlet nozzles. The limiting break size is the largest break area that results in an all-steam blowdown. A significant difference between the RSG and the OSG is that the RSG is equipped with a flow-limiting device installed in the outlet nozzle, whereas the OSG does not have a flow restrictor. The flow-limiting device consists of seven 8-inch inner diameter venturi nozzles installed in the holes of the steam outlet nozzle. The steam flow limiting devices limit a 7.406 square feet (ft^2) double-ended guillotine break of the steamline as seen by the RSGs to 2.8 ft^2, resulting in an all-steam blowdown. Therefore, the licensee has modeled all MSLB sizes as 7.406 ft^2 double-ended guillotine breaks.

The analytical response of the plant protection systems for the MSLB is the same as for the LOCA. Once the MSIVs are closed by the initiation of CIAS, steam flow into containment from the intact SG and the isolated steamline ceases. The mass and energy release to the containment continues until the faulted SG is blown down. No auxiliary feedwater (AFW) flow is assumed to the faulted SG because the SG delta-pressure comparison within the emergency feedwater actuation signal (EFAS) prevents it.

3.2.2.1 MSLB Mass and Energy Analysis

The revised MSLB mass and energy analysis was performed consistent with the current licensing basis methodology as described in UFSAR Section 6.2.1.4. The licensee stated that the MSLB analysis was performed in two parts, similar to the LOCA analysis. The SGNIII computer code was used to determine the mass and energy discharged from each SG into the containment and the mass and energy data was then used to determine the containment response using the COPATTA code. To maximize the rate of energy release to the containment, the following conservatisms were included:

- The Moody critical flow correlation was used to determine break flow rate. Mass and energy release to the containment is calculated independent of the containment pressure.

- No SG tube plugging was assumed.

- To calculate the contribution of main feedwater, including flashing, to the affected and intact SGs, an existing hydraulic pressure-balance feedwater model developed for the OSGs was used. The licensee stated that this was conservative because the OSG full-power feedwater flow rate is slightly higher than that for the RSGs. Additional conservatism is provided by stepping feedwater flow to peak flow and maintaining peak flow until the feedwater isolation valve closes.

- Peak pressure MSLB cases use the maximum initial containment pressure value.

- Turbine stop valves are assumed to close immediately (0.01 seconds).

3.2.2.2 MSLB Containment Response

The methodology used for the revised MSLB containment response analysis is consistent with the methodology described in UFSAR Section 6.2. The containment pressure and temperature response is calculated using the COPATTA code.

3.2.2.3 Break Locations and Single Failures

The licensee stated that due to the steam flow restrictor installed in the RSG outlet nozzle, a double-ended guillotine break at all initial power levels produced no entrainment in the break flow. Since the limiting break size is the largest break for which there is no entrainment in the break flow, there was no need to evaluate smaller break sizes.

Offsite power was assumed to be available throughout the transient, because the impact of running the RCPs on the transfer of RCS energy to the faulted SG has a more prominent effect on containment pressure than the delays in the actuation of containment heat removal systems.

The analysis for peak containment pressure was run at multiple power levels (0, 20, 50, 80, and 100.58 percent power). The current authorized full power rating is 3438 megawatts thermal (MWth) versus 3390 MWth in the previous evaluation. However, the sum of the licensed power limit and the power measurement uncertainty between the two power ratings did not change as a result of the NRC staff's approval of an exception from the 10 CFR Part 50, Appendix K requirement to use 2-percent power measurement uncertainty associated with the current power rating (Amendment Nos. 180 and 171, dated July 6, 2001, for SONGS, Units 2 and 3, respectively, ADAMS Accession No. ML012180237).

The failure of an MSIV to close, failure of an MFIV to close, and the failure of a containment cooling train to activate were also evaluated. The limiting failure was determined to be the failure of an MSIV to close for an MSLB accident initiated at 0 percent power.

3.2.2.4 MSLB Results

The maximum containment pressure and temperature results for all MSLB cases analyzed are shown in Table 4.2-4 of the licensee's evaluation enclosed with their letter dated June 27, 2008. The limiting case is an MSLB-initiated while at 0 percent power, with a failure of an MSIV to close. The maximum containment atmosphere temperature of 380 °F occurred at 36 seconds when containment spray flow is ramping to full flow. The peak calculated pressure of 51.5 psig occurred at 168 seconds. The pressure dropped to well below half the maximum pressure at 5000 seconds (1.39 hours). The pressure and temperature profile for the limiting case is provided in Figure 4.2-3 of the enclosure to the licensee's letter dated June 27, 2008. The peak temperature is above 300 °F for a short duration of approximately 90 seconds. The licensee stated that the calculated peak temperature is lower than the peak temperature calculated for this accident for the existing SGs and further noted that the NRC staff has previously reviewed and approved MSLB analyses for temperatures exceeding 300 °F for a short duration in the "Safety Evaluation Report [SER] related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3," dated February 6, 1981, for SONGS, Units 2 and 3 (NUREG-0712).

3.2.3 Containment Liner Temperature

The calculated RSG LOCA containment liner temperature is 251 °F, which is 1 °F higher than the OSG LOCA results. The calculated RSG MSLB containment liner temperature is also 251 °F, which is 18 °F higher than the OSG MSLB analysis.

The containment design temperature is shown as 300 °F in UFSAR Table 6.2-3, "Principal Containment Design Parameters." The licensee stated that the temperature limit of 300 °F is not a vapor temperature limit, but pertains to the containment structure such as the containment liner plate and concrete. The calculated liner temperature of 251 °F is still considerably below the limit. Therefore, the NRC staff concludes that the revised peak containment temperatures associated with the installation of the RSGs will have no adverse affect on the containment liner plate.

3.2.4 Environmental Qualification

As stated in Section 4.2.4.3 of the enclosure to the licensee's letter dated June 27, 2008, a bounding environmental qualification (EQ) case was also run for the MSLB for the RSGs to determine the maximum pressure and temperature for EQ assessment. The primary differences between the bounding EQ case and the non-EQ case runs are provided in Table 4.2-3 of the enclosure of the licensee's letter. They are:

- Per the NRC Information Notice (IN) 84-90, “Main Steam Line Break Effect on Environmental Qualification of Equipment,” dated December 7, 1984 (ADAMS Legacy Library No. 8412050090), steam superheating upon uncovering of the SG tubes is assumed for all the MSLB EQ cases.

- Consistent with the current methodology from NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," dated July 31, 1981 (ADAMS Accession No. ML031480402), the containment MSLB EQ cases take credit for 0.08 re-evaporation of condensate from the heat sinks.

- MSLB cases are initialized with the lowest allowable containment pressure. Minimizing the amount of air inside containment reduces the heat capacity of the vapor and maximizes the temperature response of containment to the MSLB event.

The results of the EQ bounding case are provided in Table 4.2-4 of the enclosure to the licensee's letter dated June 27, 2008. The maximum pressure and temperature are essentially the same as those for the non-EQ case governing the MSLB. The revised calculated maximum containment atmosphere temperature of 380 °F exceeded the containment design temperature of 300 °F provided in UFSAR Table 6.2-3. However, the licensee noted that this peak is lower than the corresponding peak previously calculated for the OSGs, and further stated that the MSLB event exceeding 300 °F for a short duration has been previously reviewed and approved by the NRC for SONGS in the NUREG-0712 SER. Initial temperatures of the structures and components inside containment will be well below the peak containment temperature. Considering that the containment atmosphere temperatures will exceed 300 °F for only a brief time, and the fact that condensate will form on the subcooled containment structures and components when in contact with superheated steam, the resulting peak temperature for any structure or component is expected to be below the corresponding EQ temperature limit. The licensee further referenced UFSAR Section 6.2.2.1.3, which states that “although the containment atmosphere may exceed 300 °F, the calculated equipment temperature of environmentally qualified equipment inside the containment is bounded by environmental qualification test temperatures.” The staff concludes that the change will have no impact on EQ of equipment inside containment and is, therefore, acceptable.

3.2.5 Summary of Containment Analysis

The LOCA and MSLB containment analyses performed by the licensee followed the current licensing basis methodology and employed the same computer codes that were previously used at SONGS, Units 2 and 3 and approved by the NRC staff. The mass and energy release analysis for the LOCA followed the guidelines in Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs),” in the SRP, NUREG-0800. The mass and energy release analysis for the MSLB followed the guidelines in Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures,” in the SRP, NUREG-0800. The decay heat input during the long-term cooling phase followed the guidance in BTP ASB 9-2.

The containment design parameters are provided in UFSAR Table 6.2-3. Per these parameters, the reactor containment is designed for a maximum internal pressure of 60 psig and containment air temperature of 300 °F. The revised analysis shows that the maximum calculated containment pressure and the maximum design temperature remain below the containment design limits, except that the temperature exceeds 300 °F for a short duration during MSLB. The licensee stated that this short duration has been previously reviewed and approved by the NRC in the SER for SONGS (NUREG-0712). The NRC staff concludes that with the proposed changes, SONGS, Units 2 and 3 will continue to meet the requirements of GDCs 16, 38, and 50, and Appendix J to 10 CFR Part 50, as follows:

- GDC 16 is satisfied since the proposed change would not result in pressure and temperatures exceeding the containment design limits, and since surveillance testing will continue to demonstrate that the containment is "essentially leak-tight."

- GDCs 38 and 50 are satisfied since the proposed changes would not result in pressures and temperatures exceeding the containment design limits. The containment response analysis for a LOCA assumes a loss-of-offsite power and a postulated failure of an entire train of ESF equipment. In addition, the analysis has shown that the containment pressure for a design-basis LOCA was reduced to less than the peak calculated pressure within 24 hours after the postulated accident, thus complying with review guidance in SRP Section 6.2.1.1.A. The MSLB analysis was based on the most severe single active failure in the containment heat removal system (loss of a train of containment ESF equipment) or the loss of secondary system isolation provisions (e.g., MSIV failure, MFIV failure). The licensee has identified a spectrum of pipe breaks resulting in the highest containment pressure and temperature, pipe break locations and reactor power levels and analyzed the containment for such breaks. The results indicate that pressures and temperatures for both LOCA and MSLB remain below the containment design limits. Therefore, the NRC staff concludes that the proposed change satisfies the requirements of GDCs 38 and 50, and is consistent with SRP Section 6.2.1.1.A.

- TS Section 5.5.2.15, "Containment Leakage Rate Testing Program," states the 10 CFR Part 50, Appendix J Type A test pressure, $P_{a,}$ is conservatively set equal to the calculated MSLB peak containment pressure, which is greater than the calculated LOCA peak pressure. Since the peak calculated pressures change for both accidents as a result of the RSGs, the P_a value is affected. However, the revised MSLB peak pressure still bounds the LOCA peak pressure, and the post-LOCA containment leakage will still be limited to less than 0.1 percent containment air weight per day, as required by the current TS 5.5.2.15. Therefore, the NRC staff concludes that the change to TS 5.5.2.15 to revise the test value for P_a will still provide an appropriate Type A test value to demonstrate that SONGS, Units 2 and 3 will meet the requirements for containment leakage testing of 10 CFR Part 50, Appendix J.

3.2.6 Technical Specification Changes

In relation to peak containment post-accident pressure, TS 5.5.2.15 currently states,

> The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 45.9 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (56.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The licensee is proposing to revise the TS 5.5.2.15 to state,

> The calculated peak containment internal pressure related to the design basis loss-of-coolant accident, P_a, is 48.0 psig (P_a will conservatively be assumed to be equal to the calculated peak containment internal pressure for the design basis Main Steam Line Break (51.5 psig) for the purpose of containment testing in accordance with this Technical Specification).

The proposed changes increase the calculated peak containment post-accident LOCA pressure from 45.9 psig to 48.0 psig, and decrease the post-MSLB peak containment pressure from 56.5 psig to 51.5 psig. The revised post-LOCA peak containment pressure is still bounded by the revised post-MSLB peak containment pressure and the containment design pressure, though the margin to the containment test pressure P_a (taken as post-MSLB peak pressure) is reduced. Despite the reduction in margin, any post-LOCA containment leakage will be still limited to less than 0.1 percent containment air weight per day as required by TS 5.5.2.15. Therefore, there is no increase in the radiological consequences of a LOCA as a result of these changes. Therefore, the NRC staff concludes that the proposed changes to TS 5.5.2.15 are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on September 23, 2008 (73 FR 54867). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 FR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributors: A. Obodoako
N. Karipineni

Date: June 25, 2009

June 25, 2009

Mr. Ross T. Ridenoure
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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: TECHNICAL SPECIFICATION CHANGES IN SUPPORT OF STEAM GENERATOR REPLACEMENT (TAC NOS. MD9160 AND MD9161)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. NPF-10 and Amendment No. 213 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 27, 2008, as supplemented by letters dated August 13, 2008, and February 5, 2009.

The amendments revise TSs 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.2.11, "Steam Generator (SG) Program," 5.5.2.15, "Containment Leakage Rate Testing Program," and 5.7.2.c, "Special Reports," and support of the replacement of the steam generators at SONGS, Units 2 and 3.

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/

James R. Hall, 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. 220 to NPF-10
2. Amendment No. 213 to NPF-15
3. Safety Evaluation

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