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

July 17, 2015

Mr. Thomas J. Palmisano
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 AMENDMENT FOR PERMANENTLY SHUTDOWN AND DEFUELED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS (TAC NOS. MF3774 AND MF3775)

Dear Mr. Palmisano:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 230 to Facility Operating License No. NPF-10, and Amendment No. 223 to Facility Operating License No. NPF-15, for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively. The amendments consist of changes to the SONGS facility operating licenses and the Technical Specifications (TSs) in response to your application dated March 21, 2014, as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015.

The proposed amendments revise the operating licenses and associated TSs to reflect the permanent cessation of reactor operations and the permanently defueled condition of the reactor vessels at SONGS Units 2 and 3. In general, the changes eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of irradiated fuel assemblies. Changes were also made to the TS definitions, administrative controls, and related to programs and procedures. The proposed amendments also revise the facility operating licenses to clarify or remove certain conditions no longer relevant and add conditions consistent with other permanently shutdown and defueled reactors. Related Amendment Nos. 227 and 220 for SONGS Units 2 and 3, respectively, were issued on September 30, 2014, to revise and remove certain requirements from Section 5.0, "Administrative Controls," of the SONGS Units 2 and 3 TSs to reflect the permanently shutdown and defueled staffing and training requirements for SONGS Units 2 and 3 operations staff.

T. Palmisano

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

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 230 to NPF-10
2. Amendment No. 223 to NPF-15
3. Safety Evaluation

cc w/enclosures: 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. 230
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 March 21, 2014, as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015, 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, Facility Operating License No. NPF-10 is hereby amended to read, as follows, as indicated in the attachment to this license amendment.

Paragraph 2.B.(2) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess and use the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

Paragraph 2.B.(3) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

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

Paragraph 2.B.(4) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(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 sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

Paragraph 2.C.(1) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(1) Deleted

Paragraph 2.C.(2) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 230, 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.

Paragraph 2.C.(14) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(14) Deleted

Paragraph 2.C.(27) of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

(27) Deleted

New License Condition 2.C.(28) of Facility Operating License No. NPF-10 is hereby added to read, as follows:

(28) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 2 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

Paragraph 2.J of Facility Operating License No. NPF-10 is hereby amended to read, as follows:

J. Deleted

New License Condition 3 of Facility Operating License No. NPF-10 is hereby added to read, as follows:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
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: July 17, 2015

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. 223
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 March 21, 2014, as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015, 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, Facility Operating License No. NPF-15 is hereby amended to read, as follows, as indicated in the attachment to this license amendment.

   Paragraph 2.B.(2) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

   (2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", to possess and use the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

   Paragraph 2.B.(3) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

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

   Paragraph 2.B.(4) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

   (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 sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

   Paragraph 2.C.(1) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

   (1) Deleted

   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. 223, 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.

Paragraph 2.C.(12) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(12) Deleted

Paragraph 2.C.(28) of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

(28) Deleted

New License Condition 2.C.(29) of Facility Operating License No. NPF-15 is hereby added to read, as follows:

(29) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 3 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 3 until such time that all spent fuel has been removed from the Unit 3 spent fuel pool.

Paragraph 2.J of Facility Operating License No. NPF-15 is hereby amended to read, as follows:

J. Deleted

New License Condition 3 of Facility Operating License No. NPF-15 is hereby added to read, as follows:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
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: July 17, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 230

TO FACILITY OPERATING LICENSE NO. NPF-10

AND LICENSE AMENDMENT NO. 223

TO FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NOS. 50-361 AND 50-362

Replace the following pages of the Facility Operating License Nos. NPF-10 and 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 No. NPF-10

| Remove | Insert |
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Facility Operating License No. NPF-15

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| -9- | -9- |

Technical Specifications

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# Facility Operating License No. NPF-10

# Revised License Pages

G. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-10, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings and the Partial Initial Decision issued by the Atomic Safety and Licensing Board on January 11, 1982, regarding this facility, Facility Operating License No. NPF-10 is hereby issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report as supplemented and amended, and the Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess and use the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license;

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[1]The City of Anaheim has transferred its ownership interests in the facility, and entitlement to facility output, to Southern California Edison Company, except that it retains its ownership interests in its spent nuclear fuel and the facility's independent spent fuel storage installation located on the facility's site. In addition, the City of Anaheim retains financial responsibility for its spent fuel and for a portion of the facility's decommissioning costs. The City of Anaheim remains a licensee for purposes of its retained interests and liabilities.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, 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 sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

(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) Deleted

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 230, 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.

(14) Deleted

(15) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No. 185

(16) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No. 185

(17) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No. 185

(18) Initial Test Program (Section 14, SER)

Deleted by Amendment No. 185

(19) NUREG-0737 Conditions (Section 22)

a. Shift Technical Advisor (I.A.1.1, SSER #1)
Deleted by Amendment No. 185

b. Shift Manning (I.A.1.3. SSER #1, SSER #5)
Deleted by Amendment No. 147

c. Independent Safety Engineering Group (1.B.1.2. SSER #1)

Deleted by Amendment No. 185

d. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

Deleted by Amendment No. 185

6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(27) Deleted

(28) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 2 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. SCE shall fully implement and maintain in effect all provisions of the Commission- approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21 is entitled: "San Onofre Nuclear Generating Station Security, Training and Qualification, and Safeguards Contingency Plan, Revision 2" submitted by letter dated May 15, 2006. SCE shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The SONGS CSP was approved by License Amendment No. 225.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. DELETED

H. SCE shall notify the Commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. Deleted

*On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

   This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

   A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

   B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A (Technical Specifications)
2. Appendix B (Environmental Protection Plan)
3. Appendix C (Antitrust Conditions)

Date of Issuance: FEB 16 1982

Facility Operating License No. NPF-15

Revised License Pages

F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-15, subject to the condition for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings and the Partial Initial Decision issued by the Atomic Safety and Licensing Board on January 11, 1982, and the Initial Decision issued by the Atomic Safety and Licensing Board on May 14, 1982 regarding this facility, Facility Operating License No. NPF-15 is hereby issued to the Southern California Edison Company, the San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to read as follows:

A. This license applies to the San Onofre Nuclear Generating Station, Unit 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by the licensees. The facility is located in San Diego County, California, and is described in the Final Safety Analysis Report, as amended, through Amendment 30, and the Environmental Report, as amended, through Amendment 6.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) Southern California Edison Company, San Diego Gas and Electric Company, the City of Riverside, California, and the City of Anaheim, California[1] to possess the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license;

(2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess and use the facility at the designated location in San Diego County, California in accordance with the procedures and limitations set forth in this license;

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[1]The City of Anaheim has transferred its ownership interests in the facility, and entitlement to facility output, to Southern California Edison Company, except that it retains its ownership interests in its spent nuclear fuel and the facility's independent spent fuel storage installation located on the facility's site. In addition, the City of Anaheim retains financial responsibility for its spent fuel and for a portion of the facility's decommissioning costs. The City of Anaheim remains a licensee for purposes of its retained interests and liabilities.

(3) SCE, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, 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 sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

(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) Deleted

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 223, 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) Antitrust Conditions

SCE shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Containment Tendon Surveillance (Section *3.8.1, SER, SSER #5)

Deleted by Amendment No. 26

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

Deleted by Amendment No. 49

(6) High Burnup Fission Gas Release (Section 4.2.2.2, SER)

Deleted by Amendment No. 176

(7) Low Temperature Overpressurization Protection (Section 5.2.2.2, SER)

Deleted by Amendment No. 176

(8) Volume Control Tank Control Logic (Section 7.3.5, SSER #4)

Deleted by Amendment No. 176

(9) Compliance with Regulatory Guide 1.97 (Section 7.5.1, SER, SSER #5)

Deleted by Amendment No. 176

(10) Control System Failures (Section 7.7, SER, SSER #4)

Deleted by Amendment No. 176

(11) Diesel Generator Modifications (Section 8.3.1, SER)

Deleted by Amendment No. 176

(12) Deleted

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(13) Turbine Disc Inspection (Section 10.2.2, SER)

Deleted by Amendment No. 176

(14) Radioactive Waste System (Section 11.1, SER, SSER #5)

Deleted by Amendment No. 176

(15) Purge System Monitors (Section 11.3, SER, SSER #5)

Deleted by Amendment No. 176

(16) Initial Test Program (Section 14, SER)

Deleted by Amendment No. 176

(17) NUREG-0737 Conditions (Section 22)

Deleted by Amendment No. 176

a. Procedures for Transients and Accidents (I.C.1, SSER #1, SSER #2, SSER #5)

   Deleted by Amendment No. 176

b. Procedures for Verifying Correct Performance of Operating Activities (I.C.6, SSER #1)

   Deleted by Amendment No. 176

c. Control Room Design Review (I.D.1, SSER #1)

   Deleted by Amendment No. 176

d. Post Accident Sampling System (NUREG-0737 Item II.B.3)

   Deleted by Amendment No. 169

e. Direct Indication of Safety Valve Position (II.D.3, SSER #1)

   Deleted by Amendment No. 176

f. AFW Pump 48-hour Endurance Test (II.E.1.1, SSER #11)

   Deleted by Amendment No. 176

g. Emergency Power Supply for Pressurizer Heaters (II.E.3.1, SSER #1, SSER #5)

   Deleted by Amendment No. 176

h. ICC Instrumentation (II.F.2, SSER #1, SSER #2, SSER #4)

   Deleted by Amendment No. 176

(27) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(28) Deleted

(29) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 3 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 3 until such time that all spent fuel has been removed from the Unit 3 spent fuel pool.

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation

J. Deleted

K. Deleted by Amendment No. 176

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Attachment 1 - Deleted by Amendment No. 176
2. Appendix A (Technical Specifications)
3. Appendix B (Environmental Protection Plan)
4. Appendix C (Antitrust Conditions)

Date of Issuance: NOV 15 1982

# APPENDIX A

# TO THE

# FACILITY OPERATING LICENSE NPF-10

# AND

# FACILITY OPERATING LICENSE NPF-15

# TECHNICAL SPECIFICATIONS FOR

# SAN ONOFRE NUCLEAR GENERATING STATION
# UNIT 2 AND UNIT 3

TABLE OF CONTENTS

# 1.0 USE AND APPLICATION

## 1.1 Definitions

---------------------------------------------------------NOTE---------------------------------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

---

| Term | Definition |
| --- | --- |
| ACTIONS | ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times. |
| CERTIFIED FUEL HANDLER | A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2. |
| OPERABLE–OPERABILITY | A system, subsystem, train, component, or device shall be OPERABLE when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). |

# 1.0 USE AND APPLICATION

## 1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLE

The following example illustrates the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. LCO not met. | A.1 Verify . . .<br><br>AND<br><br>A.2 Restore . . . | |

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

# 1.0 USE AND APPLICATION

## 1.3 Completion Times

| | |
|---|---|
| PURPOSE | The purpose of this section is to establish the Completion Time convention and to provide guidance for its use. |
| BACKGROUND | Limiting Condition for Operation (LCOs) specify minimum requirements for ensuring safe storage of fuel assemblies. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s). |
| DESCRIPTION | The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability. |

EXAMPLE

The following example illustrates the use of Completion Times.

EXAMPLE 1.3-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required Action and associated Completion Time not met. | B.1 Verify . . .<br><br>AND<br><br>B.2 Restore . . . | 6 hours<br><br><br><br>36 hours |

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to perform the verification within 6 hours AND perform the restoration within 36 hours. A total of 6 hours is allowed for performing the verification and a total of 36 hours (not 42 hours) is allowed for performing the restoration from the time that Condition B was entered. If verification is performed within 3 hours, the

1.3 Completion Times

| | |
|---|---|
| EXAMPLE (continued) | |
| | time allowed for performing the restoration is the next 33 hours because the total time allowed for performing the restoration is 36 hours. |
| IMMEDIATE COMPLETION TIME | When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. |

# 1.0 USE AND APPLICATION

## 1.4 Frequency

| | |
|---|---|
| PURPOSE | The purpose of this section is to define the proper use and application of Frequency requirements. |
| DESCRIPTION | Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.<br><br>The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. |
| EXAMPLES | The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) occurs whenever any fuel assembly is stored in the fuel storage pool. |

<u>EXAMPLE 1.4-1</u>

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| Verify . . . | 7 days |

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (7 days) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 7 days, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.

1.4 Frequency

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

<u>EXAMPLE 1.4-2</u>

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| Verify . . . | Prior to moving a fuel assembly . . . |

Example 1.4-2 illustrates a one time performance Frequency.

This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

## 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

| | |
|---|---|
| LCO 3.0.1 | LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2. |
| LCO 3.0.2 | Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.<br><br>If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated. |

## 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

| | |
|---|---|
| SR 3.0.1 | SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits. |
| SR 3.0.2 | The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. |
| SR 3.0.3 | If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.<br><br>If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period.<br><br>When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance. |
| SR 3.0.4 | Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS. |

## 3.1 PLANT SYSTEMS

### 3.1.1 Fuel Storage Pool Water Level

LCO 3.1.1 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of fuel assemblies in the fuel storage pool.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Fuel storage pool water level not within limit. | A.1 | Suspend movement of fuel assemblies in fuel storage pool. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.1.1 | Verify the fuel storage pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks. | 7 days |

## 3.1 PLANT SYSTEMS

### 3.1.2 Fuel Storage Pool Boron Concentration

LCO 3.1.2 The fuel storage pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Fuel storage pool boron concentration not within limit. | A.1 | Suspend movement of fuel assemblies in the fuel storage pool. | Immediately |
| | AND | | |
| | A.2 | Initiate action to restore fuel storage pool boron concentration to within limit. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.2.1 | Verify the fuel storage pool boron concentration is within limit. | 7 days |

## 3.1 PLANT SYSTEMS

### 3.1.3 Spent Fuel Assembly Storage

LCO 3.1.3

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region I shall be within the acceptable burnup domain of Figure 3.1.3-1 or Figure 3.1.3-2 or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

The combination of initial enrichment and burnup of each SONGS 2 and 3 spent fuel assembly stored in Region II shall be within the acceptable burnup domain of Figure 3.1.3-3 or Figure 3.1.3-4, or the fuel assembly shall be stored in accordance with Technical Specification 4.3.1.1.

Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Technical Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 Initiate action to bring the noncomplying fuel assembly into compliance. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.3.1 | Verify by administrative means the initial enrichment, burnup, and cooling time of the fuel assembly are in accordance with LCO 3.1.3, or Design Features 4.3.1.1, or Licensee Controlled Specification (LCS) 4.0.100. Rev 2, dated 09/27/07. | Prior to moving a fuel assembly to any spent fuel pool storage location. |

FIGURE 3.1.3-1

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION I RACKS

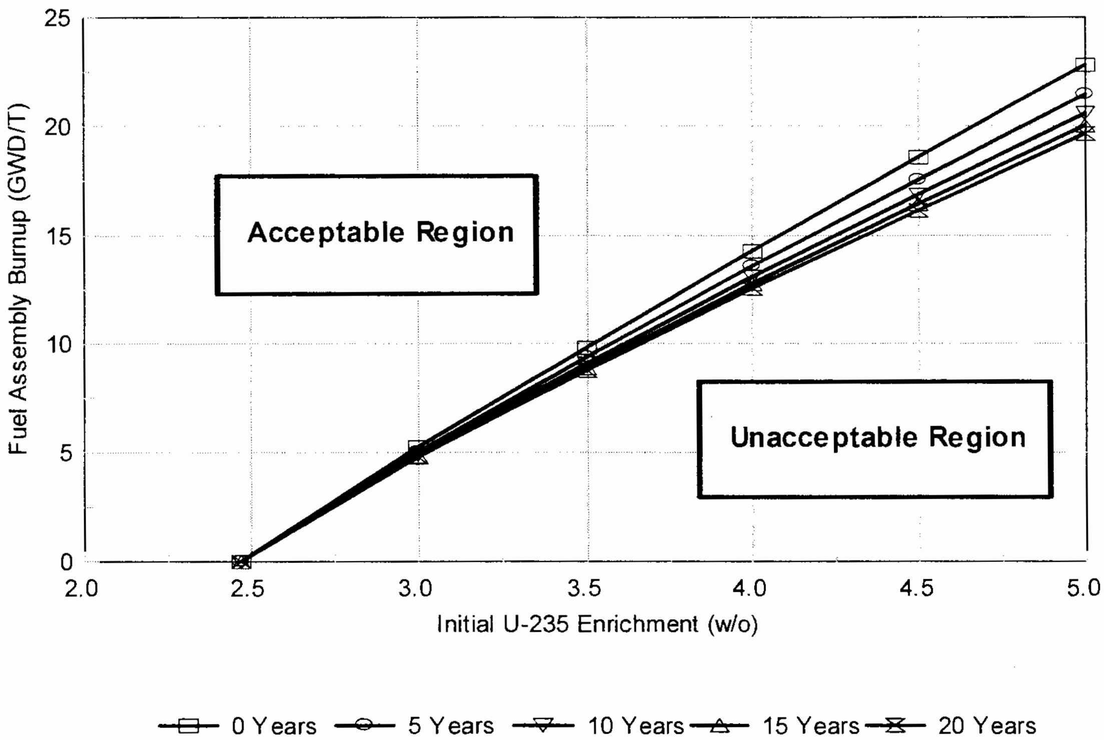


FIGURE 3.1.3-2

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION I RACKS

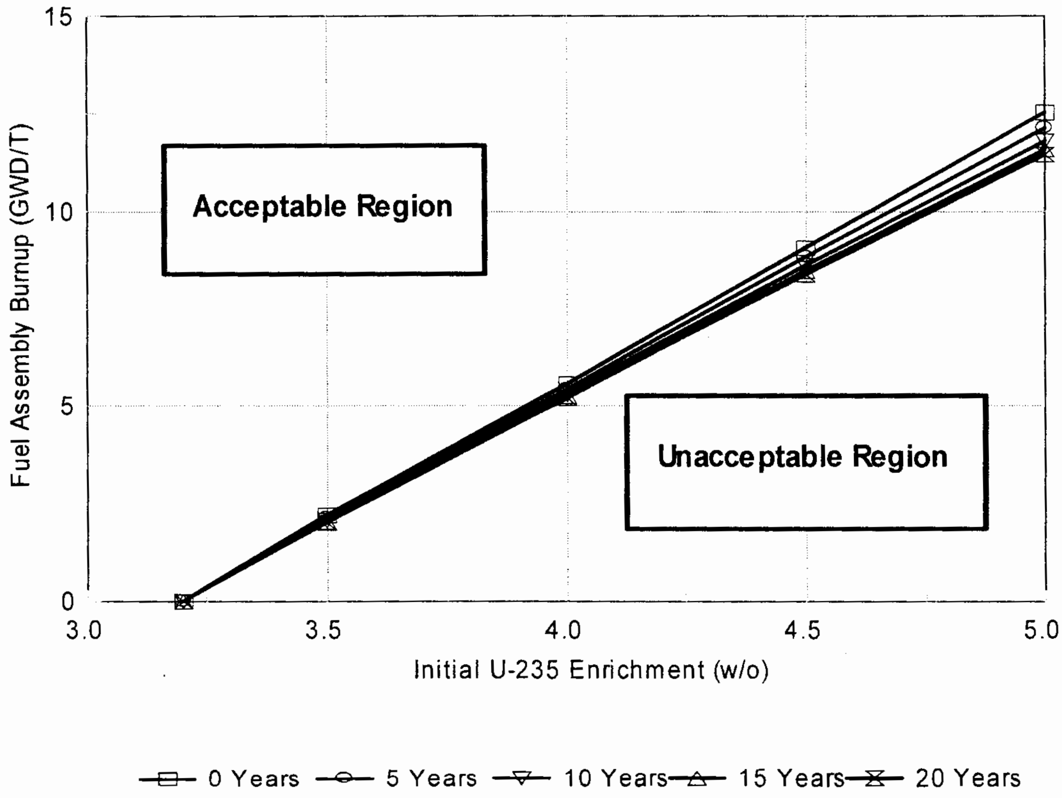


FIGURE 3.1.3-3

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL
IN
REGION II RACKS

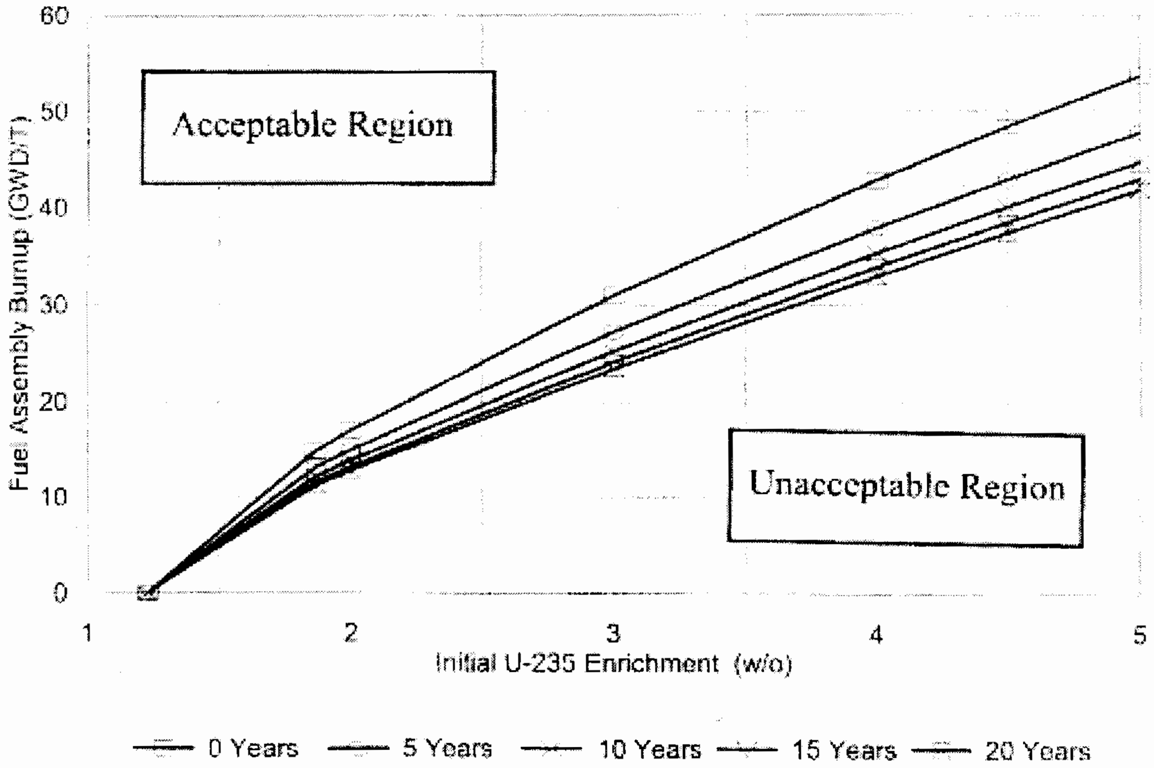
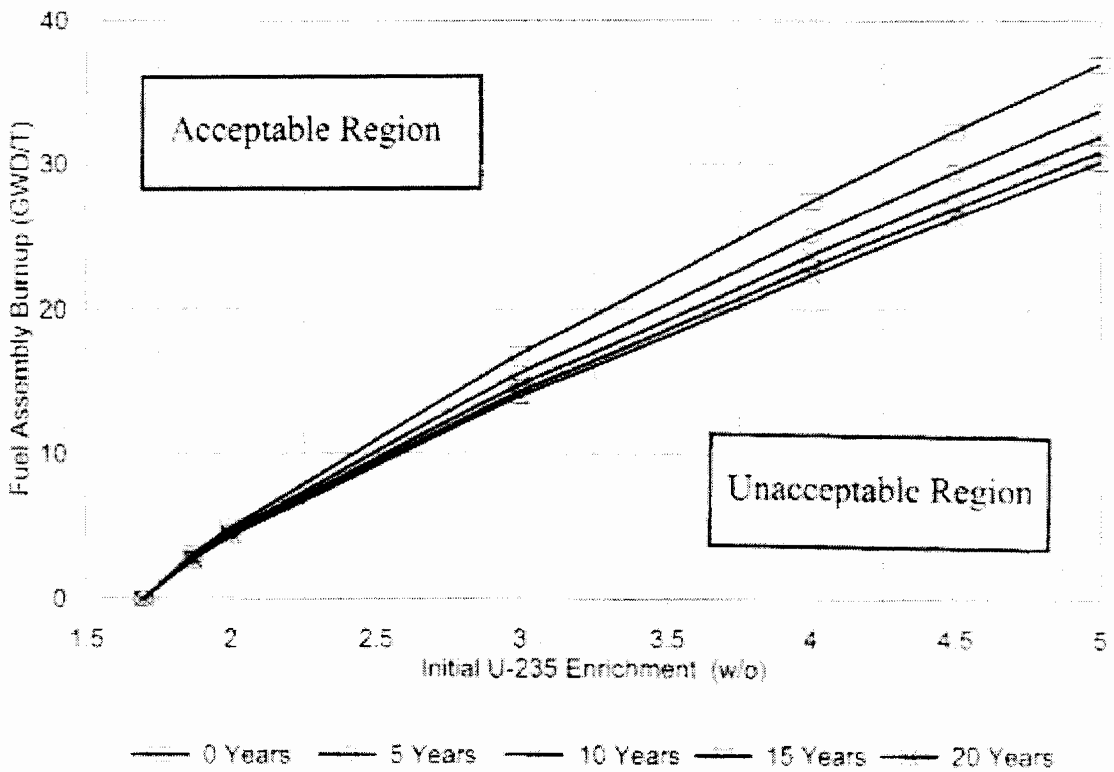


FIGURE 3.1.3-4

MINIMUM BURNUP AND COOLING TIME VS. INITIAL ENRICHMENT
FOR
PLACEMENT OF SONGS 2 AND 3 FUEL IN PERIPHERAL POOL LOCATIONS
IN
REGION II RACKS



# 4.0 DESIGN FEATURES

## 4.1 Site

### 4.1.1 Exclusion Area Boundary

The exclusion area boundary shall be as shown in Figure 4.1-1.

### 4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2.

## 4.2 Deleted.

## 4.0 DESIGN FEATURES (continued)

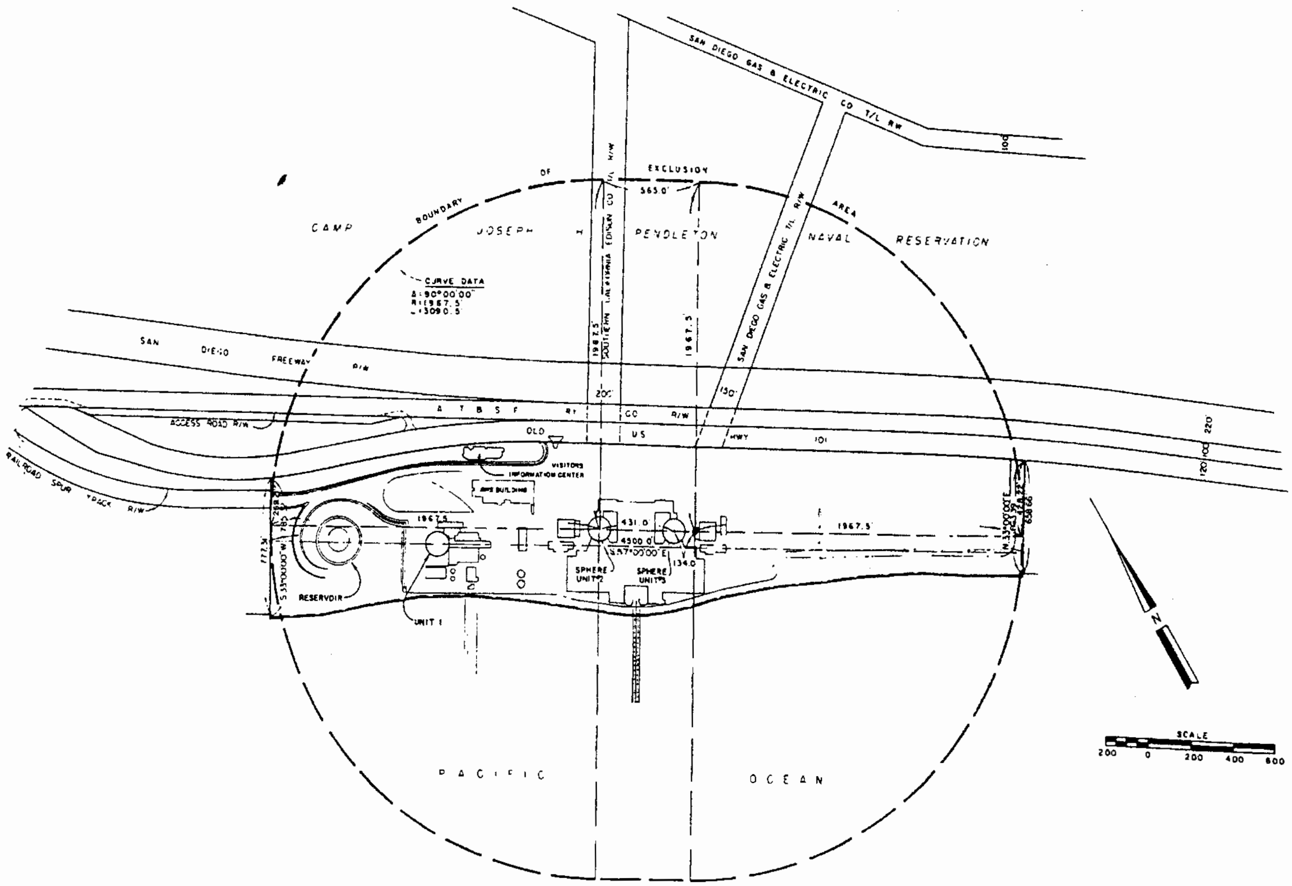



Figure 4.1-1 (page 1 of 1)
Exclusion Area Boundary

4.0 DESIGN FEATURES (continued)

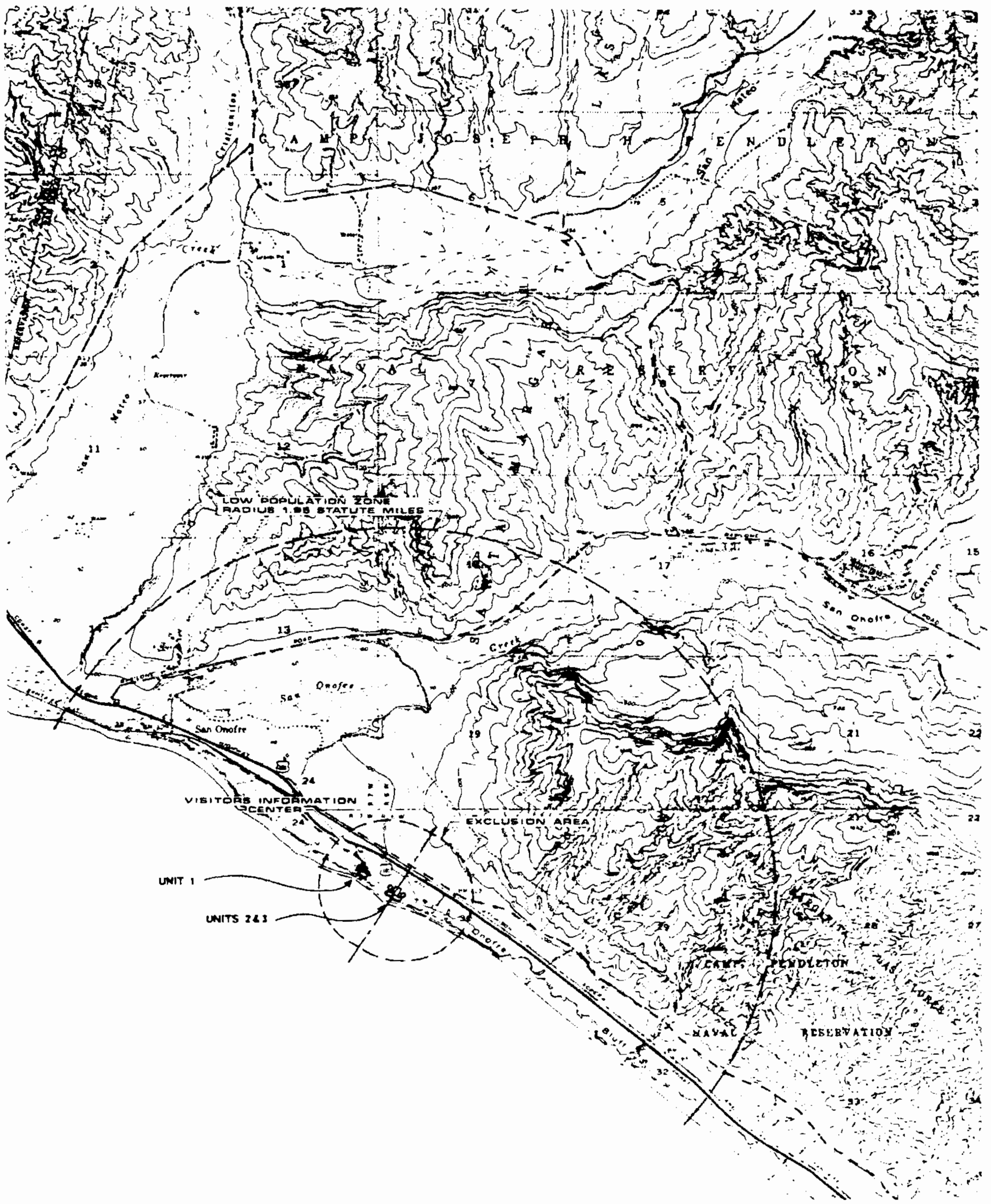



Figure 4.1-2 (page 1 of 1)
Low Population Zone

4.0 DESIGN FEATURES (continued)

## 4.3 Fuel Storage

### 4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.8 weight percent;

b. $K_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to 1700 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. Three or five Borated stainless steel guide tube inserts (GT-Insert) may be used. When three borated stainless steel guide tube inserts are used, they will be installed in an assembly's center guide tube, the guide tube associated with the serial number, and the diagonally opposite guide tube. Fuel containing GT-Inserts may be placed in either Region I or Region II. However, credit for GT-Inserts is only taken for Region II storage.

   A five-finger CEA may be installed in an assembly. Fuel containing a five-finger CEA may be placed in either Region I or Region II. Credit for inserted 5-finger CEAs is taken for both Region I and Region II.

e. A nominal 8.85 inch center to center distance between fuel assemblies placed in Region II;

f. A nominal 10.40 inch center to center distance between fuel assemblies placed in Region I;

g. Prior to using the storage criteria of LCO 3.1.3 and LCS 4.0.100, the following uncertainties will be applied:

   (1) The calculated discharge burnup of San Onofre Units 2 and 3 assemblies will be reduced by 6.6%.

   (2) The calculated discharge burnup of San Onofre Unit 1 fuel assemblies will be reduced by 10.0%.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

h. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-1 are allowed unrestricted storage in Region I;

i. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-2 are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region I;

j. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-3 are allowed unrestricted storage in Region II;

k. Units 2 and 3 fuel assemblies with a burnup in the "acceptable range" of Figure 3.1.3-4 are allowed unrestricted storage in the peripheral pool locations with 1 or 2 faces toward the spent fuel pool walls of Region II;

l. Units 2 and 3 fuel assemblies with a burnup in the "unacceptable range" of Figure 3.1.3-1, Figure 3.1.3-2, Figure 3.1.3-3, and Figure 3.1.3-4 will be stored in compliance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 9/27/07; and

m. Each SONGS 1 uranium dioxide spent fuel assembly stored in Region II shall be stored in accordance with Licensee Controlled Specification 4.0.100 Rev. 2, dated 9/27/07.

4.3.2 <u>Drainage</u>

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Technical Specification 3.1.1 value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

4.3.3 <u>Capacity</u>

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 fuel assemblies.

# 5.0 ADMINISTRATIVE CONTROLS

## 5.1 Responsibility

5.1.1 The corporate officer with direct responsibility for the plant shall be responsible for overall management of the San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the ultimate command decision authority for all unit activities which affect the safety of the plant, site personnel, and/or the general public.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These relationships, including the plant-specific titles of those personnel fulfilling the responsibilities for the positions delineated in these Technical Specifications, are documented in the UFSAR.

b. The corporate officer with direct responsibility for the plant shall be responsible for overall safe handling and storage of nuclear fuel and shall have control over those onsite activities necessary for safe handling and storage of the nuclear fuel.

c. A specified corporate officer (or officers) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure safe management of nuclear fuel.

d. The individuals who train CERTIFIED FUEL HANDLERS, and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 FACILITY STAFF

The facility staff organization shall include the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

b. At least one person qualified as Emergency Coordinator/Emergency Director shall be in the Control Room when nuclear fuel is stored in the spent fuel pools.

5.2 Organization

5.2.2 FACILITY STAFF (continued)

c. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

d. Oversight of fuel handling operations shall be provided by a CERTIFIED FUEL HANDLER.

e. The Shift Manager shall be a CERTIFIED FUEL HANDLER.

f. An individual qualified in radiation protection procedures shall be on site during fuel handling operations or movement of loads over the storage racks containing fuel.

## 5.2 Organization (continued)

Table 5.2.2-1
Minimum Shift Crew Composition

| POSITION | MINIMUM STAFFING |
|---|---|
| CERTIFIED FUEL HANDLER | 1* |
| Certified Operator | 1 |

Note: The Certified Operator position may be filled by a CERTIFIED FUEL HANDLER.
* May be shared between Units 2 and 3.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except: a) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.2 An NRC approved training and retraining program for the CERTIFIED FUEL HANDLERS shall be maintained.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.4 Technical Specifications (TS) Bases Control

5.4.1 Changes to the Bases of the TS shall be made under appropriate administrative controls.

5.4.2 Changes to the Bases may be made without prior NRC approval provided the changes do not require either of the following:

a. A change in the TS incorporated in the license; or

b. A change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

5.4.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

5.4.4 Proposed changes that meet the criteria of (a) or (b) above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC every 24 months.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.5 Procedures, Programs, and Manuals

5.5.1 Procedures

5.5.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory guide 1.33, Revision 2, Appendix A, February 1978;

b. Deleted.

c. Quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1, 1979;

d. Fire Protection Program implementation; and

e. Programs, as specified in Specification 5.5.2.

5.5.2 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.2.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program;

b. The ODCM shall also contain the Radioactive Effluent Controls required by Specification 5.5.2.3 and Radiological Environmental Monitoring programs required by the LCS, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radioactive Effluent Release Report required by Specification 5.7.1.2 and Specification 5.7.1.3.

5.5.2.1.1 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

5.5 Procedures, Programs, and Manuals

5.5.2.1.1 Licensee-initiated changes to the ODCM (continued):

1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

3. Documentation of the fact that the change has been reviewed and found acceptable.

b. Shall become effective upon review and approval by the corporate officer with direct responsibility for the plant or designee.

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Deleted

5.5.2.3 Radioactive Effluent Controls Program

This program conforming to 10 CFR 50.36a provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by operating procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM;

5.5 Procedures, Programs, and Manuals

5.5.2.3 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.2.4 Deleted

5.5.2.5 Deleted

5.5.2.6 Deleted

5.5 Procedures, Programs, and Manuals (continued)

5.5.2.7 Storage Tank Radioactivity Monitoring Program

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include a surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program surveillance frequencies.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.6 Deleted

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

5.7.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4. The reports shall be addressed to the U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., with a copy to the Regional Administrator of the Regional Office of the NRC, unless otherwise noted.

5.7.1.1 Deleted

5.7.1.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the thermoluminescent dosimeter (TLD) results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7 Reporting Requirements (continued)

5.7.1.3 Radiological Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the facility. The report shall also include a summary of the quantities of solid radioactive waste shipped from the facility directly to the disposal site and quantities of solid radioactive waste shipped from the facility's intermediary processor to the disposal site. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

## 5.0 ADMINISTRATIVE CONTROLS

### 5.8 High Radiation Area

5.8.1 Each high radiation area as defined 10 CFR 20 shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area,

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them,

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable REP.

5.8.2 In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift manager on duty or radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved REP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the REP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.8.3 Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 1.0 rem in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area shall be barricaded and conspicuously posted. A flashing light shall be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 223 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 12, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML131640201), Southern California Edison (SCE, the licensee) submitted a certification to the U.S. Nuclear Regulatory Commission (NRC) indicating its intention to permanently cease power operations at San Onofre Nuclear Generating Station (SONGS), Units 2 and 3 as of June 7, 2013, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.82(a)(1)(i). By letters dated June 28, 2013 (ADAMS Accession No. ML13183A391), and July 22, 2013 (ADAMS Accession No. ML13204A304), SCE submitted certifications of permanent removal of fuel from the Unit 3 and Unit 2 reactor vessels as of October 5, 2012, and July 18, 2013, respectively, pursuant to 10 CFR 50.82(a)(1)(ii). Upon docketing of these certifications, and pursuant to 10 CFR 50.82(a)(2), the SONGS Units 2 and 3 facility operating licenses no longer authorize operation of the reactors or emplacement or retention of fuel into the reactor vessels. Spent fuel is currently stored onsite in the spent fuel pools (SFPs) and in the onsite independent spent fuel storage installation (ISFSI).

By letter dated March 21, 2014 (ADAMS Accession No. ML14085A141), as supplemented by letters dated October 1, 2014; and February 23, February 25, and March 18, 2015 (ADAMS Accession Nos. ML14280A264, ML15058A030, ML15058A033, and ML15082A017, respectively), SCE submitted a license amendment request consisting of amendment applications to Facility Operating License Nos. NPF-10 and NPF-15 for SONGS Units 2 and 3, respectively. The proposed amendments would revise the facility operating licenses and revise the associated technical specifications (TSs) to reflect the permanent cessation of operations of SONGS Units 2 and 3.

The supplemental letters dated October 1, 2014; and February 23, February 25, and March 18, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no

significant hazards consideration determination as published in the *Federal Register* on September 16, 2014 (79 FR 55513).

As stated above, pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors or emplacement or retention of fuel in the reactor vessels. SONGS Units 2 and 3 have been shut down since January 2012. At the time of the licensee's submittal, the fission product inventory of all spent fuel that is stored in the SONGS Units 2 and 3 spent fuel pools had decayed more than two years since last irradiated in the reactor core. SONGS Unit 1 was permanently shut down in 1993 and is already in the decommissioning phase where above ground structures have been dismantled and the spent fuel is stored in either the SONGS ISFSI or in the GE-Hitachi Morris facility.

The existing SONGS TSs contain limiting conditions for operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the existing TSs provide an appropriate level of control. However, the majority of the existing TSs are only applicable when the reactor is in an operational MODE. Since the SONGS Units 2 and 3, Part 50 licenses no longer authorize emplacement or retention of fuel in the reactor vessels, the LCOs (and associated surveillance requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The proposed amendments revise the operating licenses and associated TSs to reflect the permanent cessation of reactor operations and the permanently defueled condition of the reactor vessels at SONGS Units 2 and 3. In general, the changes eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of irradiated fuel assemblies. Changes were also proposed to TS definitions, administrative controls, and related to programs and procedures. The proposed amendments also revise the facility operating licenses to clarify or remove certain conditions no longer relevant and add conditions consistent with other permanently shutdown and defueled reactors.

Amendment Nos. 227 and 220 for SONGS Units 2 and 3, respectively, were issued by the NRC on September 30, 2014 (ADAMS Accession No. ML14183B240), to revise certain requirements in the permanently shutdown and defueled facility's TSs, Section 5.0, "Administrative Controls," related to responsibilities, organization, and facility staff qualifications that reflect new staffing and training requirements for operating staff. Issuance of the enclosed amendments, in conjunction with the previously issued TS administrative control amendments, completes the revision to the SONGS permanently shutdown and defueled technical specifications. The TSs being proposed for revision by SCE for incorporation into the SONGS Units 2 and 3 facility operating licenses, referred to by SCE as the Permanently Defueled Technical Specifications (PDTS), have been combined into a single TS that applies to both units. The licensee states that the changes to the facility operating licenses and TSs provide an appropriate level of safety, considering the reduced risk of an offsite radiological release from the remaining postulated design-basis accidents (DBAs) associated with a defueled plant, as described in this safety analysis.

## 2.0 REGULATORY EVALUATION

### 2.1 Technical Specifications

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the application. The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." Pursuant to 10 CFR 50.36, each operating license issued by the Commission includes TSs and includes items in the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) LCOs, (3) SRs, (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

Section 50.36 of 10 CFR states, in part, that "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity... Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions."

Section 50.36 of 10 CFR provides four criteria to define the scope of equipment and parameters to be included in the TS LCOs. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focused on instrumentation to detect degradation of the reactor coolant system (RCS) pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during DBAs or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those structures, systems, and components (SSCs) shown to be significant to public health and safety.

SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCO will be met.

Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs 50.36(c)(1), (2), and (3).

Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility is in a safe manner.

The regulations in 10 CFR 50.36 further state that TSs involving safety limits, limiting safety system settings, and limiting control system settings; LCOs; SRs; design features; and administrative controls for decommissioning facilities will be developed on a case-by-case basis.

A general discussion of the criteria that were used by the NRC staff in its evaluation to ensure that the TS LCOs proposed for deletion are no longer required to be included in the TSs is

provided below. These criteria were also used in the evaluation of the proposed changes to the existing TSs and the proposed new TSs.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for "installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary [RCPB]." Since no fuel is present in the reactor, maintenance of the RCS pressure boundary as a fission product barrier is no longer relevant at the SONGS Units 2 and 3 facility, and therefore, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. The scope of DBAs applicable to a reactor permanently shutdown and defueled is reduced from those postulated for an operating reactor, and most TSs satisfying Criterion 2 are no longer applicable. The scope of applicable DBAs that apply to SONGS Units 2 and 3 are discussed in more detail in Sections 3.1 through 3.6 of this safety evaluation (SE). There are no transients that continue to apply to the permanently shutdown and defueled reactors.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for a SSC "that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The intent of this criterion is to capture into TSs those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. The scope of applicable DBAs that apply to SONGS Units 2 and 3 are discussed in more detail in Sections 3.1 through 3.6 of this SE. There are no transients that continue to apply to the permanently shutdown and defueled reactors.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs "which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs. There are no longer any DBAs at SONGS Units 2 and 3 that can result in a significant offsite radiological risk to public health and safety.

The NRC staff notes that in the course of this evaluation, information contained in DRAFT NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," March 1998 (ADAMS Accession No. ML082330233), was also considered. This draft NUREG provides examples of decommissioning TSs for Westinghouse

pressurized water reactors that the staff has previously found acceptable during TS reviews for permanently shutdown and defueled reactors.

2.2 Radiological Consequences from Design-Basis Accidents

During normal power reactor operations, the forced flow of water through the RCS removes the heat generated by the reactor. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator (SG) tubes to the secondary system. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the RCS and subsequent release of some fission products to the environment. Many of the accident scenarios postulated in the facility safety analysis report involve failures or malfunctions of systems that could affect the reactor core. With the termination of reactor operations and the permanent removal of the fuel from the reactor core, such accidents are no longer possible. Therefore, the postulated accidents involving failure or malfunction of the reactor, RCS, or secondary system are no longer applicable. Postulated accidents that could potentially apply to a permanently shutdown and defueled facility include a fuel handling accident (FHA), an accidental release of waste liquid, an accidental release of waste gas, a spent fuel cask drop accident, and a spent fuel pool boiling event. The potential offsite consequences of these events are affected by the time available for decay of fission products in the fuel and, possibly, the availability of engineered safety features, such as ventilation systems to filter fission products from the accident area atmosphere before they are released outside the facility.

The regulations in 10 CFR 50.67, "Accident source term" state, in part, that:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem) total effective dose equivalent (TEDE), (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), and (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Appendix A to 10 CFR Part 50, "General Design Criteria (GDC)," Criterion 19--Control room, states, in part:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving

> radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ADAMS Accession No. ML003716792), provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67 - Accident source term. Regulatory Guide 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the AST.

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," SRP, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190), provides review guidance to the NRC staff for the review of alternative source term amendment requests. SRP Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. As provided in RG 1.183, the dose acceptance criteria for an FHA are a TEDE of 6.3 rem at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room (CR) for the duration of the accident.

SRP 11.0, Branch Technical Position 11-5, "Postulated Radioactive Release Due to a Waste Gas System Leak or Failure," provides guidance to the NRC staff in assessing the analysis of an accidental release from the waste gas system.

The NRC approved implementation of the AST methodology at SONGS Units 2 and 3, by Amendment Nos. 210 and 202, "San Onofre Nuclear Generating Station, Units 2 and 3 – Issuances of Amendments Re: Full-Scope Implementation of an Alternative Source Term (TAC Nos. MC5495 and MC5496)," dated December 29, 2006 (ADAMS Accession No. ML063400359). These license amendments represent full scope implementation of the AST, as described in RG 1.183.

NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347), discusses experiences with analyzing an accident involving a release from off-gas or waste systems. As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste system. For this type of accident, licensees have proposed acceptance criteria of 500 millirem (mrem) TEDE. The acceptance criterion for this event is that associated with the dose to an individual member of the public, as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed

from 500 mrem whole body to 100 mrem TEDE. Therefore, a licensee who chooses to implement AST for an off-gas or waste gas system release, as did SCE, should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

The U.S. Environmental Protection Agency's (EPA's) "Protective Action Guide (PAG) and Planning Guidance for Radiological Incidents," Draft for Interim Use and Public Comment, issued March 2013 (PAG Manual), provides radiological protection criteria for application to all incidents that would require consideration of protective actions, with the exception of nuclear war. This manual provides recommended numerical PAGs for the principal protective actions available to public officials during a radiological incident.

The Nuclear Energy Institute (NEI) document NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6, dated November 2012 (ADAMS Accession No. ML12326A805), provides guidance for the development of emergency action levels (EALs) for reactors in a permanently defueled condition. NEI 99-01, Revision 6, was endorsed by the NRC in a letter dated March 28, 2013 (ADAMS Accession No. ML12346A463). NEI 99-01 states that the accident analysis necessary to adopt the permanently defueled EAL scheme must confirm that the source terms and release motive forces are not sufficient to warrant classification of a site area emergency (SAE) or General Emergency, resulting in the maximum classification level of an Alert during an accident. An SAE would be declared for any event where exposure levels beyond the EAB are expected to exceed 10 percent of the EPA PAGs, which are a projected dose of 1 to 5 rem TEDE in four days for sheltering or evacuation of the public, and a projected dose of 5 rem child thyroid dose from radioactive iodine for administration of prophylactic drugs (potassium iodide). Correspondingly, NEI 99-01 established the SAE classification threshold as 100 mrem TEDE or 500 mrem thyroid committed dose equivalent.

## 3.0 TECHNICAL EVALUATION

### 3.1 Design-Basis Accident Analysis

SCE described that, with the permanent cessation of reactor operations and permanent removal of fuel from the reactor vessels for SONGS Units 2 and 3, most of the initial conditions, and most of the accident and transient analyses, that were included in Chapter 15 of the SONGS Updated Final Safety Analysis Report (UFSAR) when Units 2 and 3 were authorized to operate, are no longer possible. Therefore, SCE has updated the SONGS UFSAR to reflect that accidents and transients involving the failure or malfunction of fuel within primary containment, the RCS, or the secondary system are no longer applicable. The only DBA scenarios with the potential to result in a radiological release, as described in the UFSAR that are applicable to the permanently shutdown and defueled SONGS Units 2 and 3, are an FHA in the fuel handling building (FHB), a spent fuel cask drop accident, a SFP boiling accident, a liquid radioactive waste system leak or failure, a radioactive release due to liquid tank failures, and an accidental release of waste gas. Because the waste gas decay tanks have been purged of their contents and analyses of liquid tank failures in SONGS UFSAR Section 15.7.3.3.5 describe that no credible liquid release would exceed 10 CFR Part 20 limits, an accidental release of waste gas

and a liquid tank failure are not relevant at SONGS. The licensee determined that the remaining DBAs would be within relevant regulatory limits, assuming the fuel activity calculated as of August 2013 and without credit for dose consequence mitigation by engineered safety feature (ESF) systems. The NRC staff's technical evaluation of the licensee's analysis of the remaining DBAs at SONGS is provided in Sections 3.2 through 3.6, below.

3.2 Fuel Handling Accident Inside Fuel Building

A revision to the FHA accident analysis was incorporated into the SONGS UFSAR, Section 15.7.3.4, under the provisions of 10 CFR 50.59, "Changes, tests, and experiments," to address the permanently defueled condition. The analysis determined a reasonable time, post-cessation of operations, for movement of fuel from the SFP during which, if an FHA occurs, dose consequences would be within 10 CFR 50.67 and RG 1.183 dose limits. The licensee evaluated the maximum 2-hour TEDE to an individual located at the EAB, and the 30-day TEDE to an individual at the outer boundary of the LPZ and in the CR. The resulting doses in SCE's analyses are less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria, the 10 CFR 50.67 limits, and the EPA PAG levels recommended for protection of the public.

The FHA inside the FHB (FHA-FHB) involves the inadvertent dropping of a fuel assembly during fuel handling operations, and the subsequent rupture of fuel pins in the dropped assembly and any stationary assembly impacted by the dropped assembly. A maximum of 472 fuel rods are assumed to fail, as a result of the drop of a fuel assembly onto the fuel assemblies stored in SFP fuel racks. The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously into the SFP. The FHA-FHB dose analysis models 17 months (12,240 hours) of radioactive decay prior to the event. The NRC staff finds that the decay time assumed by the licensee is consistent with RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," which provides that, "For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled."

The SFP water level is controlled by current SONGS Units 2 and 3 TS LCO 3.7.16 (renumbered to TS 3.1.1 in the proposed permanently defueled TSs), which limits the movement of irradiated fuel assemblies in the SFP, unless the water level is at least 23 feet over the top of the irradiated fuel assemblies, seated in the storage racks. As such, the licensee assumes that the SFP water level is at least 23 feet over the top of the irradiated fuel assemblies, seated in the storage racks, at the commencement of an FHA-FHB.

Should an FHA occur, fission products released from the damaged fuel are decontaminated by passage through the pool water, with the degree of decontamination dependent upon their physical and chemical forms. The licensee assumed no decontamination for noble gases, a decontamination factor of 200 for radioiodine, and retention of all aerosol and particulate fission products. This is consistent with RG 1.183, Appendix B, Section 2, "Water Depth," which provides that, "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200..."

The radioactive material that escapes from the SFP to the FHB is assumed to be released to the environment over a 2-hour time period. The FHA-FHB dose analysis does not credit the generation of an engineered safety feature actuation system (ESFAS) fuel handling building isolation signal (FHIS). The FHB normal ventilation exhaust is assumed to remain operational throughout the FHA-FHB event. The FHA-FHB AST dose analysis does not model a reduction in the amount of radioactive material available for release from the FHB by the fuel handling building Post-Accident Cleanup Unit (PACU) filter system. Therefore, the licensee assumes the release to the environment is an unfiltered release via the FHB normal ventilation exhaust system through the main plant vent, or as leakage through FHB penetrations. This is consistent with RG 1.183, Appendix B, Section 4.1, which states, "The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period."

Activity released during the FHA-FHB event is transported by atmospheric dispersion to the CR heating, ventilation and air conditioning (HVAC) intake and to the offsite EAB and LPZ dose receptors. Consistent with RG 1.183, Regulatory Position 5.3, "Meteorology Assumptions," the atmospheric dispersion factor values for the EAB and the LPZ, which were approved by the NRC during initial facility licensing were used by the licensee in performing the AST radiological analyses. The NRC had also approved the use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202, dated December 29, 2006 (for SONGS Units 2 and 3, respectively). Consistent with RG 1.183, Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The CR dose during a design-basis FHA-FHB, following permanent shutdown of SONGS Units 2 and 3, was based on no credit for the Control Room Emergency Air Cleanup System (CREACUS) and Control Room Isolation Signal (CRIS) and no gamma radiation shine from CREACUS charcoal and high-efficiency particulate air (HEPA) filters. Control room doses are evaluated at various CR unfiltered inflow (including in leakage) flow rates. The flow rates were varied from 500 cubic feet per minute (cfm) to 15,000 cfm, but only the bounding CR dose is reported. The SONGS site-specific 95th percentile meteorology atmospheric dispersion factors for the CR were used.

The licensee concluded that the radiological consequences at the EAB and LPZ and in the CR are within the dose criteria for DBAs specified in 10 CFR 50.67 and SRP Section 15.0.1. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee, as well as, NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the FHA-FHB analyses assumptions described above, the NRC staff's confirmatory analyses of the licensee's FHA-FHB yield results for the CR, EAB, and LPZ that are less than the RG 1.183 and SRP 15.0.1 dose acceptance criteria and would not exceed the EPA PAG recommendations at the EAB.

3.3 Spent Fuel Cask Drop Accident

A re-analysis of the spent fuel cask drop accidents specified in the UFSAR, Section 15.7.3.5, was performed with a cask load of up to 32 fuel assemblies and a minimum of 17 months of decay. The spent fuel cask drop event is evaluated based on the potential of the cask drop to cause a release of radioactive material. This includes consideration of the allowed travel paths of the casks, their lift heights, and the items onto which they can be dropped. Even though single-failure-proof cranes are used at SONGS Units 2 and 3 to lift a spent fuel transfer cask out of a cask pool, a drop can be postulated when the cask is placed on the upper shelf (i.e., step) of a cask pool when performing a yoke-lift change-out, prior to the transfer cask being welded closed. The spent fuel cask drop accident considered to bound the radiological consequences of a spent fuel transfer cask drop (due to a seismic event) is from the upper shelf in the cask pool back into the lower portion of the cask pool. During this postulated accident, the transfer cask is not restrained and could fall back into the lower portion of the cask pool if an earthquake occurs. The fuel rods from all 32 fuel assemblies present in a transfer cask are conservatively assumed to rupture on impact with the bottom of the cask pool. All of the radioactive iodine and noble gases present in the gap volumes of the spent fuel rods are assumed to be released from the unwelded transfer cask. As required by the AST Amendment Nos. 210 and 202, dated December 29, 2006 (for SONGS Units 2 and 3, respectively), the new analysis was performed by the licensee using the AST methodology, including TEDE criteria. The NRC staff concludes that the licensee's modelling of decay time is consistent with RG 1.183, Regulatory Position 3.1.

Other than the number of fuel assemblies considered to fail, the radiological consequence analysis model is identical to that of the FHA in the FHA-FHB (see Section 3.2 of this SE). The fission product inventory in the fuel rod gap of the damaged rods is assumed to be released instantaneously into the SFP. The SFP water level is required to be at least 23 feet over the top of the irradiated fuel assemblies seated in the storage racks, as controlled by TSs. Consistent with RG 1.183, Appendix B, Regulatory Position 4.1, the radioactive material that escapes from the SFP to the FHB is released to the environment over a 2-hour time period, ensuring that at least 99.9 percent of the gaseous activity will be released to the environment. Consistent with RG 1.183, Regulatory Position 5.3, the atmospheric dispersion factor values for the EAB and the LPZ that were approved by the NRC during initial facility licensing are used in performing the AST radiological analyses. The NRC had also approved use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202 for SONGS Units 2 and 3, respectively. Consistent with RG 1.183, Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The licensee concluded that the radiological consequences at the EAB and LPZ and in the CR are within the dose criteria for the DBAs, as specified in 10 CFR 50.67. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the analyses assumptions

described above, the NRC staff's confirmatory analyses yielded results for the CR, EAB, and LPZ that were less than the RG 1.183 dose acceptance criteria and would not have exceeded the EPA PAG recommendations at the EAB.

3.4 Spent Fuel Pool Boiling Accident

The postulated loss of all SFP cooling is assumed to result in SFP boiling and release of a portion of the radionuclide inventory contained in the stored spent fuel assemblies and the SFP water. The re-evaluation of the radiological consequences for the SFP boiling event assumes a minimum of 17 months since the shutdown of SONGS Units 2 and 3. The licensee used the AST methodology in performing this evaluation. The NRC staff concludes that the licensee's modelling of decay time is consistent with RG 1.183, Regulatory Position 3.1.

The radiological consequence analysis does not differentiate between the activity release rates before and after the onset of SFP boiling. Noble gas, iodine and tritium activity present in the failed fuel rod gap spaces of fuel rods, stored within the SFP, is released to the SFP water at the noble gas, iodine, and tritium escape rate coefficients, with the added conservatism of an assumed spiking factor of 100. The noble gas and iodine fuel rod gap fractions are consistent with the AST methodology. The tritium fuel rod gap fraction is assumed to be the same as that for the majority of noble gas and iodine isotopes. Tritium activity present in the SFP water prior to the loss of SFP cooling, is assumed to be released at the SFP boiling rate for the duration of the event. Both before and after the onset of SFP boiling spent fuel noble gases, iodine and tritium gas escaping from the failed fuel rod gap spaces are assumed to be instantaneously released with no hold up or iodine partitioning in the SFP water. The SFP boiling rate is a function of the decay heat load and the heat of vaporization of water.

Following a loss of SFP cooling, activity releases from the spent fuel due to evaporation and boiling disperse to the CR, EAB, and LPZ locations. No credit is taken for activity retention within the FHB. No credit is taken for FHIS or filtration by the FHB PACUs. All activity escaping from the SFP is assumed to be instantaneously released to the environment and atmospherically dispersed to the CR and offsite dose receptors. No credit is taken for CRIS or CREACUS.

The SFP boiling accident consequence analysis uses the identical model used for the FHA-FHB (see Section 3.2 of this SE). Consistent with RG 1.183, Appendix B, Section 4.1, the radioactive material that escapes from the SFP to the FHB is released to the environment over a 2-hour time period, ensuring that at least 99.9 percent of the gaseous activity will be released to the environment. For conservatism, the CR dose is calculated for an individual at the CR outside air intake location.

Consistent with RG 1.183, Regulatory Position 5.3, the atmospheric dispersion factor values for the EAB and the LPZ, which were approved by the NRC during initial facility licensing, are used in performing the AST radiological analyses. The NRC staff had also approved the use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202, dated December 29, 2006 (for SONGS Units 2 and 3, respectively). Consistent with RG 1.183,

Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The licensee concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria for DBAs, as specified in 10 CFR 50.67. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee, as well as NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the analyses assumptions described above, the NRC staff's confirmatory analyses yielded results for the CR, EAB, and LPZ that are less than the RG 1.183 dose acceptance criteria and would not exceed the EPA PAG recommendations at the EAB.

3.5 Radioactive Waste System Leak or Failure (Release to Atmosphere) Accident

The radioactive waste system leak or failure (with release to atmosphere) accident analysis (UFSAR Section 15.7.3.2) was revised to calculate the EAB and LPZ doses using the AST methodology. As required by the AST Amendment Nos. 210 and 202 for SONGS Units 2 and 3, respectively, the evaluation includes TEDE dose criteria, and a revised offsite dose acceptance criterion of 100 mrem TEDE, as addressed in NRC RIS 2006-04. The evaluation does not assume any post-shutdown decay time.

Releases from the Liquid Radioactive Waste System considered rupture of: radwaste tanks, refueling water storage tanks, primary ion-exchangers, and the blowdown demineralizer neutralization sump line. The most limiting of these is defined as an unexpected and uncontrolled release of the radioactive liquid stored in a radwaste secondary tank. The radwaste secondary tanks are Seismic Category II, Quality Class III tanks at atmospheric pressure. Rupture of these tanks is considered a limiting fault. A radwaste secondary tank rupture would release the liquid contents into the auxiliary building (radwaste area). It is assumed that all of the radioactive fission gases and iodines are released to the outside atmosphere within 2 hours.

The dose analysis for persons located at the EAB and the LPZ considers the dose consequences of inhalation and submersion in a radioactive cloud, as described in RG 1.183. Activity released during the event is transported by atmospheric dispersion to the offsite EAB and LPZ dose receptors. Consistent with RG 1.183, Regulatory Position 5.3, the atmospheric dispersion factor values for the EAB and the LPZ, which were approved by the NRC during initial facility licensing, are used in performing the AST radiological analyses. The NRC staff had also approved use of these meteorology atmospheric dispersion values by Amendment Nos. 210 and 202 for SONGS Units 2 and 3, respectively. Consistent with RG 1.183, Regulatory Position 4.1.7, no correction is made for depletion of the effluent plume by deposition on the ground.

The licensee concluded that the radiological consequences are less than 100 mrem TEDE offsite dose criterion per RIS 2006-04. The licensee also concluded that the radiological consequences are less than the dose criteria specified in the EPA PAG Manual. The NRC staff reviewed the licensee's evaluation and performed confirmatory calculations. In performing this review, the NRC staff relied upon information provided by the licensee, as well as NRC staff experience in performing similar reviews. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are acceptable because they are consistent with the guidance provided in RG 1.183. Using the analyses assumptions described above, the NRC staff's confirmatory analyses yielded results for the EAB, LPZ, and CR that are less than RG 1.183 dose acceptance criteria and are also less than the offsite dose criteria per RIS 2006-04 and would not exceed the EPA PAG recommendations at the EAB.

### 3.6 Accident Analysis Conclusions

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The NRC staff finds that the licensee's analyses are acceptable because their analysis methods and assumptions are consistent with the guidance of RG 1.183. The NRC staff compared the doses estimated by the licensee to the applicable criteria and to the results of confirmatory analyses by the staff. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses due to postulated DBAs at SONGS will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183, and 10 CFR Part 20, as addressed in NRC RIS 2006-04. The NRC staff finds, with respect to the consequences of the remaining DBAs at SONGS, that no CR dose limits will be exceeded and that any offsite radiological release will not exceed the EPA PAGs at the EAB.

### 3.7 Proposed TS Changes

#### 3.7.1 Section 1.1, Definitions

The licensee proposed deleting the following definitions because they pertain to an operating reactor. Since SONGS Units 2 and 3 are permanently shut down and defueled, the definitions have no relevance and no longer apply:

> AXIAL SHAPE INDEX (ASI) – ASI shall be the power generated in the lower half of the core less the power generated in the upper half of the core, divided by the sum of the power generated in the lower and upper halves of the core.
>
> $$\mathrm{ASI} = \frac{\text{lower}-\text{upper}}{\text{lower}+\text{upper}}$$
>
> AZIMUTHAL POWER TILT ($T_q$) - AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.
>
> CHANNEL CALIBRATION – A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the

necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element.

The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK – A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST - A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display and trip functions;

b. Bistable channels (e.g., pressure switches and switch contacts) - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions; or

c. Digital computer channels - the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION - CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components, excluding control element assemblies (CEAs) withdrawn into the upper guide structure, affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE

ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.7.1.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131 - DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of 1-131, 1-132, 1-133, 1-134, and 1-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP-30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

$\bar{E}$- AVERAGE DISINTEGRATION ENERGY - $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME - The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE – LEAKAGE shall be:

a. Identified LEAKAGE

   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

   2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the

operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE – A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

PHYSICS TESTS – PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, "Initial Test Program of the SONGS Units 2 and 3 UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) - The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.7.1.6.

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

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME - The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power to the CEAs drive mechanism is interrupted. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire

response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM) – SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All full length CEAs (shutdown and regulating) are fully inserted except for the single CEA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CEAs verified fully inserted by two independent means, it is not necessary to account for a stuck CEA in the SDM calculation. With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM, and

b. There is no change in part length CEA position.

STAGGERED TEST BASIS - A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER – THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

In conjunction with deletion of the term "MODE," TS Table 1.1-1, "MODES," is also being deleted.

The NRC staff examined the TS definitions proposed for deletion and concluded that all the terms listed above are only meaningful to a reactor authorized to operate. Since SONGS Units 2 and 3 are permanently shut down and defueled, the NRC staff finds that the licensee's proposed change to delete these definitions from the TSs is acceptable.

In addition, the licensee proposed adding a definition for CERTIFIED FUEL HANDLER. The licensee proposed to define a certified fuel handler as:

CERTIFIED FUEL HANDLER – A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by TS 5.3.2.

TS 5.3.2 states, "An NRC approved training and retraining program for the Certified Fuel Handlers shall be maintained." The NRC staff finds the definition of a Certified Fuel Handler

conforms to the usage contained in the Administrative Controls section of the SONGS Units 2 and 3 permanently defueled TSs and is consistent with the definition in 10 CFR Part 50 and is, therefore, acceptable.

3.7.2 Section 1.2, Logical Connectors

Section 1.2, "Logical Connectors," of the SONGS TSs provides an explanation of the use of logical connectors. Logical connectors are used in TSs to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TSs are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

The licensee proposed deleting Example 1.2-2 in this section of the SONGS Units 2 and 3 defueled TSs. Example 1.2-2 explains the use of multiple logical connectors and nested connectors in the Required Action section. The licensee states that Example 1.2-1 adequately explains the use of the logical connectors that remain in the proposed defueled TS.

The only logical connector retained in the defueled TS is contained in the Required Action section for renumbered LCO 3.1.2. This use of logical connectors is fully illustrated by Example 1.2-1 in Section 1.2. Therefore, the NRC staff finds that the licensee's proposed change to delete Example 1.2-2 from TS Section 1.2 is acceptable.

3.7.3 Section 1.3, Completion Times

Section 1.3, "Completion Times," of the SONGS TSs establishes the completion time convention throughout the TSs and provides guidance for its use. The licensee has proposed to replace each reference to "operation of the unit" and "unit" with the new terminology, "storage of fuel assemblies" and "facility," respectively, since operation of the unit is no longer permitted and safe storage of fuel assemblies is the primary objective of the permanently defueled TSs. In its February 23, 2015, response to the NRC staff's request for additional information (RAI), the licensee stated that the requirements in the defueled TSs are applicable to the storage of any fuel assembly, and are not limited to the safe storage of irradiated fuel assemblies. In addition, the licensee proposed to delete references to "MODE" to be consistent with the removal of these definitions from TSs and because this term is no longer used in the Required Actions of the subsequent remaining LCOs in the proposed SONGS Units 2 and 3 defueled TSs. The licensee also proposed to delete Examples 1.3-2 through 1.3-7 because these examples refer to activities that no longer pertain to a permanently defueled condition.

The proposed change is shown below, with a strikethrough of the current wording and highlighting of the proposed changes:

> BACKGROUND
>
> Limiting Condition [sic] for Operation (LCOs) specify minimum requirements for ensuring safe ~~operation of the unit~~ storage of fuel assemblies. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which

the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the ~~unit~~ facility is in a ~~MODE or~~ specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the ~~unit~~ facility is not within the LCO Applicability. ...

The licensee proposed to delete several paragraphs of the Description Section that discuss entry into multiple Conditions, subsequent discovery of additional inoperable equipment, and the effects on the total Completion Times.

The licensee proposed to modify the Example, as follows:

EXAMPLE~~S~~ The following example~~s~~ illustrates the use of Completion Times ~~with different types of Conditions and changing Conditions~~.

EXAMPLE 1.3-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required Action and associated Completion Time not met. | B.1 ~~Be in MODE 3~~ Verify.... | 6 hours |
| | AND | |
| | B.2 ~~Be in MODE 5~~ Restore.... | 36 hours |

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

> The Required Actions of Condition B are to ~~be in MODE 3~~ perform the verification within 6 hours AND perform the restoration ~~in MODE 5~~ within 36 hours. A total of 6 hours is allowed for performing the verification ~~reaching MODE 3~~ and a total of 36 hours (not 42 hours) is allowed for performing the restoration ~~reaching MODE 5~~ from the time that Condition B was entered. If verification is performed ~~MODE 3 is reached~~ within 3 hours, the time allowed for ~~reaching MODE 5~~ performing the restoration is the next 33 hours because the total time allowed for performing the restoration ~~reaching MODE 5~~ is 36 hours.

The NRC staff has reviewed the proposed wording changes to the TS 1.3 Completion Times guidance and has determined that they are consistent with the transition from an operating reactor to a permanently shutdown and defueled facility with a primary safety focus of storage of fuel assemblies. The proposed changes also remove references to operating MODES that are no longer permitted following the licensee's submittal of certifications in accordance with 10 CFR 50.82(a)(2).

Examples 1.3-2 through 1.3-7, which are proposed to be deleted, provide an explanation of the time requirements for transitioning into a MODE in which the requirements are not applicable associated with entry in TS 3.0.3, or provide an explanation of more complex arrangements of Required Actions and Completion Times beyond those retained in the defueled TSs. TS 3.0.3 is being deleted as discussed in Section 3.7.6 of this safety evaluation. The NRC staff finds that these examples are no longer necessary to understand and properly implement the remaining Required Actions and Completion Times.

For the reasons discussed above, the NRC staff has determined that the revision to Example 1.3-1 and the deletion of the remaining examples are appropriate. Therefore, the NRC staff finds that the licensee's proposed changes to TS Section 1.3 are acceptable.

### 3.7.4 Section 1.4, Frequency

Section 1.4, "Frequency," of the SONGS TSs, defines the proper use and application of Frequency requirements throughout the TSs. In this section, the licensee has proposed to delete the final paragraph in the description section. The final paragraph of the TS 1.4 description section discusses notes that modify the frequency of performance of some surveillances and the applicability of operating MODE entry restrictions of SR 3.0.4. None of the surveillances in the proposed TSs contain notes that modify the frequency of performance or the conditions during which the acceptance criteria must be satisfied. Therefore, this paragraph is not applicable to the proposed TS LCOs or SRs and may be deleted. Specifically, the following is being deleted:

> Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential

SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The licensee proposed replacing the statement introducing the examples from "In these examples, the Applicability of the LCO ...is MODES 1, 2, and 3" with the statement "In these examples, the Applicability of the LCO...occurs whenever any fuel assembly is stored in the fuel storage pool." The reference to MODEs 1, 2, and 3 is no longer meaningful to the permanently shutdown and defueled condition at SONGS Units 2 and 3.

The licensee proposed to modify Examples 1.4-1 and 1.4-2 to be applicable to a facility that is permanently shutdown and defueled and proposed to delete Example 1.4-3, in its entirety.

A summary of the proposed changes is shown below, with a strikethrough of the current wording and highlighting of the proposed changes:

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ~~Perform CHANNEL CHECK~~. Verify. . . | ~~12 hours~~ 7days |

The licensee proposed to modify the discussion of Example 1.4-1 to reflect the 7 day frequency chosen for the example; to replace the word "unit" with the word "facility"; to delete the reference to MODEs; and to delete the reference to Example 1.4-3, which is also being deleted.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ~~Verify flow is within limits.~~ Verify... | Prior to moving a fuel assembly. . . ~~Once within 12 hours after~~ |

| SURVEILLANCE | FREQUENCY |
|---|---|
| | ~~≥ 25% RTP~~<br><br>~~AND~~<br><br>~~24 hours thereafter~~ |

The licensee proposed to revise the discussion of Example 1.4-2, as follows:

> Example 1.4-2 **illustrates** ~~has two Frequencies. The first is~~ a one time performance Frequency. ~~and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.~~
>
> ~~The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND").~~ This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. ~~"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP~~.

The licensee also proposed to delete Example 1.4-3 because it is not needed in a permanently defueled condition. The example used a reference to operating reactor MODES that are no longer permitted at SONGS Units 2 and 3. Specifically, Example 1.4-3 refers to surveillances to be performed after power is greater than or equal to (≥) 25 percent rated thermal power (% RTP) and discusses MODE entry restrictions.

The licensee stated that the remaining Examples 1.4-1 and 1.4-2 (as revised) are sufficient to explain the application of TS frequency requirements for the permanently defueled SONGS Units 2 and 3, TSs.

The NRC staff has reviewed the proposed changes to TS Section 1.4 and has determined that they are appropriate for a permanently shutdown and defueled reactor. The proposed changes remove references to operating MODES or rated thermal power that are no longer permitted, following certification under the provisions of 10 CFR 50.82(a)(2). The deletion of the surveillance note referring to MODE entry restrictions of SR 3.0.4 is also appropriate since none of the surveillances in the proposed remaining defueled TSs contain notes that modify the

frequency of performance or the conditions during which the acceptance criteria must be satisfied. Therefore, the NRC staff finds that the licensee's proposed changes to TS Section 1.4 are acceptable.

3.7.5 Section 2.0, Safety Limits

Section 2.0, "Safety Limits," of the SONGS TSs, establishes safety limits (SLs), which are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operation transients, and anticipated operational occurrences (AOOs). The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere.

TS 2.1, "Safety Limits" (SLs), contains two separate specifications:

- TS 2.1.1, Reactor Core SLs; and
- TS 2.1.2, Reactor Coolant System Pressure SL

TS 2.2, "SL Violations," direct actions to be taken if an SL specified in TS 2.1 is violated.

The restrictions of the SLs defined in TS 2.1.1 prevent overheating of the fuel cladding, and possible cladding perforation, which could result in the release of fission products to the reactor coolant. TS 2.1.1 is applicable in MODES 1 and 2. TS 2.1.2 defines requirements on parameters to protect the integrity of the RCS against overpressure. TS 2.1.2 is applicable in MODES 1, 2, 3, 4, and 5.

The licensee proposed to delete the SLs specified in Section 2.0, because they are not applicable to the permanently shutdown and defueled status of the plant. The licensee stated that the SL TSs limit important process variables that are necessary to reasonably protect the integrity of certain physical barriers required for safe operation of the reactor in MODES 1 through 5. However, 10 CFR 50.82(a)(2) prohibits operation of the reactor or placing fuel in the reactor vessel. Therefore, the SL TSs only address specific process variables that are no longer applicable to SONGS Units 2 and 3.

The NRC staff examined the SLs and their TS Bases. There are three SLs in Section 2.0: a minimum limit on the departure from nucleate boiling ratio (DNBR); a maximum limit on the peak fuel centerline temperature to ensure fuel and cladding integrity; and a maximum RCS pressure to ensure RCS integrity. As stated in the "Bases for DNBR," a limit is placed on the DNBR, such that, no fuel clad damage would occur as a result of normal operation and AOOs. A limit is placed on peak fuel centerline temperature, such that, a hot fuel pellet in the core will not experience centerline fuel melting. The TS Bases for the maximum RCS pressure state that RCS integrity is an important barrier in the prevention of an uncontrolled release of fission products. Because SONGS Units 2 and 3 have permanently shut down and defueled, and the

licensee has submitted certifications under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized, therefore, SLs associated with the RCS are no longer applicable. In this condition, there will be no DNBR or peak fuel centerline temperature to be monitored. Based on these findings, the NRC staff concludes the SLs no longer apply. Since the SLs are no longer applicable, TS 2.2, which specifies the actions to be taken if a SL is violated, is no longer necessary. Therefore, the NRC staff finds the licensee's proposed changes to delete TSs 2.1 and 2.2 is acceptable.

3.7.6 Section 3.0, Limiting Condition for Operation and Surveillance Requirement Applicability

Section 3.0, "Limiting Condition for Operation (LCO) Applicability," and "Surveillance Requirement (SR) Applicability," of the SONGS TSs, contains the general requirements applicable to all LCOs and SRs and applies at all times unless otherwise stated in TSs.

LCO 3.0.1, establishes the applicability statement within each individual TS as the requirement for when the LCO shall be met. The licensee proposed to delete the reference to "MODES" and the reference to LCO 3.0.7. The licensee stated that reference to MODES is no longer relevant since SONGS Units 2 and 3 are permanently shut down and defueled. In addition, the deletion of reference to LCO 3.0.7 conforms to the request to delete this LCO from the SONGS Units 2 and 3 TSs, as discussed below.

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met. The licensee proposed to delete the references to LCO 3.0.5 and 3.0.6 to conform to the request to delete TS LCO 3.0.5 and 3.0.6 from the SONGS Units 2 and 3 TSs, as discussed below.

LCO 3.0.3 establishes the Actions that must be implemented when an LCO is not met and the associated Actions are not met, or an associated Action is not provided. LCO 3.0.3 requires placing the unit in a MODE or other specified condition in which the LCO does not apply. The licensee proposed to delete LCO 3.0.3, in its entirety, since it no longer applies. The regulations prohibit operation of the plant or placing fuel in the reactor vessel and references to operating MODES is no longer relevant.

LCO 3.0.4 establishes limitations on changing MODES or other specified conditions in the applicability when an LCO is not met. The licensee proposed to delete LCO 3.0.4 since SONGS Units 2 and 3 are permanently shut down and defueled and references to operating MODES are no longer relevant.

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with Actions. The licensee proposed to delete LCO 3.0.5, in its entirety. The licensee stated that LCO 3.0.5 is no longer necessary because the proposed defueled TSs do not contain requirements for declaring equipment inoperable or removing equipment from service.

LCO 3.0.6 establishes an exception to LCO 3.0.2 to allow the use of the Safety Function Determination Program. The licensee has proposed to delete that the Safety Function Determination Program, described in TS 5.5.13. The deletion of the Safety Function Determination Program is discussed in Section 3.7.17.4 of this SE. Consequently, the licensee has proposed to delete LCO 3.0.6, in its entirety, to conform the TSs to the proposed deletion of the Safety Function Determination Program.

LCO 3.0.7 pertains to certain reactor physics special tests and operations required to be performed at various times over the life of the unit. The licensee proposed to delete LCO 3.0.7, in its entirety, since reactor physics testing is no longer relevant to a permanently shutdown and defueled facility.

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification ensures that surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. The licensee proposed to revise SR 3.0.1 to delete the reference to "MODES" since the reference to "MODES" is no longer relevant for the permanently shutdown and defueled condition at SONGS Units 2 and 3.

SR 3.0.2 permits a 25 percent extension of the interval specified in the Frequency, and establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval. The licensee proposed to delete the statements that pertain to Frequencies specified as "once" or "once per" and to delete the statement that "[e]xceptions to this Specification are stated in the individual Specifications." The licensee stated that the proposed defueled TSs no longer contain this type of Frequency or Completion Time.

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability. The licensee proposed to delete the reference to "MODE" since SONGS Units 2 and 3 is permanently shut down and defueled and operating MODES are no longer relevant.

The NRC staff has reviewed the proposed changes to TS LCO 3.0.1, SR 3.0.1, and SR 3.0.4 and determined that the changes acceptably removes the reference to "MODE," which is no longer applicable to a permanently shutdown and defueled facility. The staff has also reviewed the change to LCO 3.0.2 and determined that the change acceptably removes the reference to other TS requirements that are being deleted. Therefore, the NRC staff finds that the licensee's proposed changes to these LCOs and SRs are acceptable.

The NRC staff has reviewed the proposed deletion of TS LCOs 3.0.3 and LCO 3.0.4, and has determined that, consistent with the transition to a permanently shutdown and defueled facility, these LCOs are no relevant. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, the references to "MODE," and the discussions about shutting down the unit, are no longer applicable. The NRC staff finds the licensee's

proposed changes to delete these LCOs, reflect the current SONGS Units 2 and 3 plant status, and, therefore, are acceptable.

The NRC staff has reviewed the proposed deletion of TS LCO 3.0.5. The NRC staff agrees that this specification is no longer necessary because the defueled TSs do not contain requirements to declare equipment inoperable or to remove equipment from service. Since LCO 3.0.5 is being deleted, the deletion of reference to LCO 3.0.5 in LCO 3.0.2 is also appropriate. The NRC staff finds this change appropriately reflects the requirements in the defueled TSs.

The NRC staff has reviewed the proposed deletion of TS LCO 3.0.6. The NRC staff agrees that there will not be any systems in the permanently defueled TSs that are interrelated with other systems that have TS LCOs (support and supported systems). As a result, the conditions of LCO 3.0.6 no longer apply. Therefore, the staff finds that it is appropriate to delete LCO 3.0.6. Since LCO 3.0.6 is being deleted, the deletion of reference to LCO 3.0.6 in LCO 3.0.2 is also acceptable.

The NRC staff has reviewed the proposed deletion of TS LCO 3.0.7. The facility operating licenses no longer permit emplacement of fuel in the reactor vessels or operation of the facility, and therefore, no physics testing will be performed in the future. Therefore, the provisions of LCO 3.0.7 are no longer necessary. The NRC staff finds that the deletion of LCO 3.0.7 appropriately reflects the permanently shutdown and defueled status of the facility.

The NRC staff has reviewed the proposed changes to TS SR 3.0.2. The NRC staff agrees that the statements to be deleted are no longer necessary because the defueled TSs do not contain Frequencies and Completion Times of the type described in the statements being deleted. Therefore, the NRC staff finds the proposed changes acceptable.

Based on the above, the NRC staff finds that the licensee's proposed changes to LCO 3.0.1, LCO 3.0.2, LCO 3.0.3, LCO 3.0.4, LCO 3.0.5, LCO 3.0.6, LCO 3.0.7, SR 3.0.1, SR 3.0.2, and SR 3.0.4 are acceptable.

3.7.7 Section 3.1, Reactivity Control Systems

Section 3.1, "Reactivity Control Systems," of SONGS Units 2 and 3 TSs, contain LCOs, Actions, and SRs that provide for appropriate control of process variables, design features, or operating restrictions that are required to protect the integrity of a fission product barrier. The following TSs for SONGS Units 2 and 3 are being proposed for deletion.

TS 3.1.1, "SHUTDOWN MARGIN (SDM) – $T_{avg} > 200$ °F [degrees Fahrenheit]," specifies the requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shut down and AOOs assuming the highest reactivity worth CEA remains fully withdrawn. TS 3.1.1 is applicable in "MODES 3 and 4."

TS 3.1.2, "SHUTDOWN MARGIN (SDM) - $T_{avg} \leq 200$ °F," specifies the requirements to provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for

normal shut down and AOOs assuming the highest reactivity worth CEA remains fully withdrawn. TS 3.1.2 is applicable in "MODE 5."

TS 3.1.3, "Reactivity Balance," specifies the requirements for the comparison of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that DBAs and transient safety analyses remain valid. TS 3.1.3 is applicable in "MODES 1 and 2."

TS 3.1.4, "Moderator Temperature Coefficient (MTC)," specifies the requirements to ensure that core overheating and overcooling accidents will not violate the accident analysis assumptions. TS 3.1.4 is applicable in "MODES 1 and 2 with $k_{eff}$ [effective multiplication factor] $\geq$ 1.0."

TS 3.1.5, "Control Element Assembly (CEA) Alignment," specifies the limits on shutdown and regulating CEA alignments to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. TS 3.1.5 is applicable in "MODES 1 and 2."

TS 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," specifies the limits on shutdown CEA insertion to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required shutdown margin following a reactor trip. TS 3.1.6 is applicable in "MODE 1, MODE 2 with any regulating CEA not fully inserted."

TS 3.1.7, "Regulating CEA Insertion Limits," specifies the limits on regulating CEA sequence and physical insertion for the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion. TS 3.1.7 is applicable in "MODES 1 and 2."

TS 3.1.8, "Part Length Control Element Assembly (CEA) Insertion Limits," specifies the limits on part length CEA insertion for the function of preserving power distribution and ensuring that ejected CEA worth is maintained within limits. TS 3.1.8 is applicable in "MODE 1 > 20% RTP."

TS 3.1.9, "Boration Systems - Operating," establishes the requirements for borated water sources and flow paths to the RCS to ensure that sufficient borated water is available to maintain the reactor subcritical and provide makeup water to account for RCS shrinkage during cooldown to cold shutdown conditions. TS 3.1.9 is applicable in "MODES 1, 2, 3, and 4."

TS 3.1.10, "Boration Systems – Shutdown," establishes the requirements for borated water sources and flow paths to the RCS to ensure that sufficient borated water is available to maintain the reactor subcritical. TS 3.1.10 is applicable in "MODES 5 and 6."

TS 3.1.12, "Special Test Exception (STE) - Low Power Physics Testing," permits the relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.12 is applicable in "MODES 2 and 3 during PHYSICS TESTS."

TS 3.1.13, "Special Test Exception (STE) - At Power Physics Testing," permits relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.13 is applicable in "MODE 1 during PHYSICS TESTS."

TS 3.1.14, "Special Test Exceptions (STE) - Reactivity Coefficient Testing," permits relaxation of existing TS LCOs to allow the performance of PHYSICS TESTS. TS 3.1.14 is applicable in "MODE 1."

The NRC staff has reviewed the licensee's proposed change to delete the reactivity control system TSs for SONGS Units 2 and 3, and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain reactivity parameters of fuel loaded into a reactor vessel within the margins of conditions encountered during normal operations, anticipated occurrences, and for DBAs. The reactivity control systems TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, there is no reactor core and the reactivity control systems are no longer relevant.

The NRC staff has also reviewed Section 3.1, the reactivity control systems TSs proposed for deletion (TS 3.1.1, TS 3.1.2, TS 3.1.3, TS 3.1.4, TS 3.1.5, TS 3.1.6, TS 3.1.7, TS 3.1.8, TS 3.1.9, TS 3.1.10, TS 3.1.12, TS 3.1.13, and TS 3.1.14), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. These TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.1, Reactivity Control Systems, is acceptable.

3.7.8 Section 3.2, Power Distribution Limits

Section 3.2, "Power Distribution Limits," of the SONGS Units 2 and 3 TSs, contains LCOs, Required Actions, and SRs that provide for appropriate control of process variables, design features, or operating restrictions that are required to control power distribution in the reactor, and in turn, protect the integrity of a fission product barrier. The following TSs are being proposed for deletion.

TS 3.2.1, "Linear Heat Rate (LHR)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitation on LHR ensures that in the event of a loss-of-coolant accident (LOCA) the peak

temperature of the fuel cladding does not exceed 2200 °F. TS 3.2.1 is applicable in MODE 1 with THERMAL POWER > [greater than] 20% RTP.

TS 3.2.2, "Planar Radial Peaking Factors ($F_{xy}$)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. Limiting of the calculated Planar Radial Peaking Factors to values equal to or greater than the measured Planar Radial Peaking Factors ensures that the calculated limits remain valid. TS 3.2.2 is applicable in "MODE 1 with THERMAL POWER > 20 % RTP."

TS 3.2.3, "AZIMUTHAL POWER TILT (Tq)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitations on the Tq are provided to ensure that design operating margins are maintained. TS 3.2.3 is applicable in "MODE 1 with THERMAL POWER > 20% RTP."

TS 3.2.4, "Departure from Nucleate Boiling Ratio (DNBR)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. Operation of the core with a DNBR at, or above, this limit ensures that an acceptable minimum DNBR is maintained in the event of a loss of flow transient. TS 3.2.4 is applicable in "MODE 1 with THERMAL POWER > 20% RTP."

TS 3.2.5, "AXIAL SHAPE INDEX (ASI)," specifies the limits based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analysis. The ASI limits ensure that with $T_q$ at its maximum upper limit, the DNBR does not drop below the DNBR safety limit for AOOs. TS 3.2.5 is applicable in "MODE 1 with THERMAL POWER > 20% RTP."

The NRC staff has reviewed the proposed deletion of the power distribution limits TSs and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain reactor power and heat generation within the margins of conditions encountered during normal operation, anticipated occurrences, and for DBAs. The power distribution limits TSs are only important for a reactor authorized to operate. Because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, there is no reactor core generating power and the power distribution limits are no longer relevant.

The NRC staff has also reviewed Section 3.2, power distribution limits TSs proposed for deletion (TS 3.2.1, TS 3.2.2, TS 3.2.3, TS 3.2.4, and TS 3.2.5), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. These TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Unit 2 and Unit 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the

reactors or placing fuel in the reactor vessels and SONGS Unit 2 and Unit 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.2, Power Distribution Limits, is acceptable.

3.7.9 Section 3.3, Instrumentation

Section 3.3, "Instrumentation," of the SONGS Units 2 and 3 TSs, contains the LCOs, Actions, and SRs that provide for appropriate functional capability of sensing and control instrumentation required for the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility. The following TSs are being proposed for deletion.

TS 3.3.1, "Reactor Protective System (RPS) Instrumentation-Operating," specifies the requirements for the RPS instrumentation system to maintain the SLs during AOOs, and mitigates the consequences of DBAs in MODE 1 and MODE 2 (startup).

TS 3.3.2, "Reactor Protective System (RPS) Instrumentation – Shutdown," specifies the requirements for the RPS instrumentation system to maintain the SLs during all AOOs, and mitigates the consequences of DBAs in MODE 3 (hot standby), MODE 4 (hot shutdown), and MODE 5, when the reactor trip circuit breakers (RTCBs) are closed and the CEA drive system is capable of CEA withdrawal.

TS 3.3.3, Control Element Assembly Calculators (CEACs)," specifies the requirements to ensure the core protection calculators (CPCs) are either informed of individual CEA position within each subgroup, using one or other CEACs, or that appropriate conservatism is included in the CPC calculations to account for anticipated CEA deviations in MODES 1 and 2.

TS 3.3.4, "Reactor Protective System (RPS) Logic and Trip Initiation," specifies the requirements for RPS matrix logic, RPS initiation logic, RTBCs, and manual trip channels to effect automatic trip signals received from RPS instruments and to provide a means to manually trip the reactor in MODES 1 and 2, and in MODES 3, 4, and 5, when the RTCBs are closed and the CEA drive system is capable of CEA withdrawal.

TS 3.3.5, "Engineered Safety Features Actuation System (EFSAS) Instrumentation," specifies the requirements for the ESFAS Instrumentation to ensure ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCPB during AOOs and ensures acceptable consequences during accidents.

TS 3.3.6, "Engineered Safety Features Actuation System (EFSAS) Logic and Manual Trip," specifies the requirements for ESFAS Matrix Logic, ESFAS Initiation Logic, and Manual Trip channels to effect automatic ESFAS initiation received from ESFAS instruments and to provide a means to manually actuate an ESF system in MODES 1, 2, 3, and 4.

TS 3.3.11, "Post Accident Monitoring Instrumentation (PAMI)," is applicable in MODES 1, 2, and 3, and provides the operability requirements for accident monitoring instruments, which provides information required by the control room operators. The operability of the PAMI ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

TS 3.3.12, "Remote Shutdown System," is applicable in MODES 1, 2, and 3, and provides the operability requirements for instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room.

TS 3.3.13, "Source Range Monitoring Channels," is applicable in MODES 3, 4, and 5, and provides the operability requirements for source range monitoring instrumentation for indication, alarms, and reactor trips.

The NRC staff has reviewed the proposed deletion of the instrumentation TSs above. TS 3.3.1, TS 3.3.2, TS 3.3.3 and TS 3.3.4, are only necessary to maintain the ability of the RPS to automatically initiate a reactor scram to preserve the integrity of the fuel cladding, preserve the integrity of the primary system barrier, and minimize the energy which must be absorbed, and prevent criticality following a LOCA. TS 3.3.5 and TS 3.3.6 only concern instrumentation designed to mitigate accidents related to reactor operation. TS 3.3.11 and TS 3.3.12 only concern instrumentation to ensure there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident, and allow operators to take manual actions specified in the emergency operating procedures or to remotely shut down the reactor from a location other than the control room. TS 3.3.13 requires monitoring source range count rate level and detects a loss of SDM caused by a boron dilution event (detected as an increase in neutron flux). None of the instrumentation addressed by these TSs is needed by a reactor that has permanently shut down and defueled in accordance with 10 CFR 50.82(a)(2).

The NRC staff also has reviewed the above Section 3.3 instrumentation TSs that are proposed for deletion (TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, TS 3.3.12, and TS 3.3.13), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. These TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Unit 2 and Unit 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete Section 3.3, instrumentation TS 3.3.1, TS 3.3.2, TS 3.3.3, TS 3.3.4, TS 3.3.5, TS 3.3.6, TS 3.3.11, TS 3.3.12, and TS 3.3.13 is acceptable.

TS 3.3.7, "Diesel Generator (DG) – Undervoltage Start," specifies the loss of voltage start (LOVS) instrumentation functions be operable in MODES 1, 2, 3, 4, and when the associated DG is required to be operable by LCO 3.8.2, AC Sources – Shutdown. TS 3.3.7 is proposed for deletion by the licensee.

TS 3.3.7 provides the LCO and SRs to ensure availability of backup safety-related alternating current (AC) power to the SSCs used to prevent or mitigate postulated accidents resulting in an uncontrolled release of radioactivity of DBAs as analyzed in the SONGS Unit 2 and 3 UFSARs. This TS LCO ensures the operability of instrumentation designed to detect an undervoltage on the safety-related AC electrical busses upon a loss of offsite power and start the emergency diesel generators (EDGs) to supply backup power to the AC busses. Since SONGS Units 2 and 3 are permanently shut down and defueled, SCE has analyzed the remaining DBAs at SONGS Units 2 and 3, and given the significant fuel decay period, found that the radiological consequences will not exceed the EPA's PAGs at the site boundary. The licensee's analysis also demonstrates that the dose consequences, within the CR, of any DBAs are acceptable without relying on SSCs remaining functional for accident mitigation except the passive fuel storage pool structure which will be maintained as a TS for SONGS.) SCE calculated the DBA radiological consequences assuming no credit for control room isolation or recirculation filtration and no credit for any accident mitigation by the auxiliary building ventilation system. Calculated doses at the EAB, LPZ, and CR are within 10 CFR 50.67 and RG 1.183 dose limits. Since the bounding accident analysis for the permanently defueled condition assumed no credit for control room post-accident recirculation system emergency ventilation or filtration for mitigation of radiological releases, the CREACUS is not required (see evaluation of TS 3.7.11 in Section 3.7.13 of this SE). Because CREACUS is not required to function to mitigate a DBA, backup electrical power required to support operation of CREACUS upon a loss of offsite power is unnecessary. Therefore the EDG loss of offsite power start instrumentation is also unnecessary for all postulated DBAs.

The NRC staff determined in Section 3.1 through 3.6 of this SE that with SONGS Units 2 and 3 permanently shut down and defueled and the irradiated fuel having decayed for a significant period, CREACUS is no longer needed or credited in the primary success path of a safety sequence analysis related the remaining DBAs at SONGS Units 2 and 3. Consequently, neither primary nor backup power to support operation of CREACUS is needed. Therefore, actuation instrumentation to start the backup power EDGs is no longer required to satisfy TS Criterion 3 for inclusion in TSs as a support or actuation system that is necessary for items in the primary success path to successfully function. The NRC staff has confirmed that there are no other DBAs analyzed in the SONGS Units 2 and 3 UFSAR that rely on this instrumentation system. Based on the above, the NRC staff finds that the licensee's proposed change to delete TS 3.3.7 is acceptable.

TS 3.3.8, "Containment Purge Isolation Signal (CPIS)," specifies the requirements for instrumentation designed to close the containment purge isolation valves upon a detection of high gaseous radiation in containment. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. TS 3.3.8 is applicable in MODES 1, 2, 3, and 4, during core alterations, and during movement of fuel assemblies within containment. TS 3.3.8 is proposed for deletion by the licensee.

This TS indicates MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Unit 2 and Unit 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply. The staff also reviewed the non-MODE dependent applicability during core alterations, and during movement of fuel assemblies in containment. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the prohibition on placing fuel in the reactor vessel, it also precludes core alterations and the movement of fuel assemblies within containment.

The NRC staff also evaluated the proposed deletion of TS 3.3.8, to ensure that the LCO no longer satisfies the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff has determined that TS 3.3.8 only addresses specific plant systems, control of process variables, design features, or operating restrictions associated with the containment and are no longer needed or credited in the primary success path of a safety sequence analysis related the remaining DBAs at SONGS Units 2 and 3. Based on the above, the NRC staff finds that the licensee's proposed change to delete TS 3.3.8 is acceptable.

TS 3.3.9, "Control Room Isolation Signal (CRIS)," specifies the requirements to ensure instrumentation (actuation logic, manual trip, and gaseous radiation monitors) necessary to initiate CREACUS is operable. The CRIS terminates the normal supply of outside air to the CR and initiates actuation of the CREACUS to minimize operator radiation exposure. The radiation monitor actuation of the CREACUS in MODES 5 and 6 and during movement of fuel assemblies is the primary means to ensure control room habitability in the event of an FHA. TS 3.3.9 is applicable in MODES 1, 2, 3, 4, 5, 6, and during movement of fuel assemblies within containment and in the fuel storage pool. TS 3.3.9 is proposed for deletion by the licensee.

This TS indicates MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

The NRC staff evaluated the proposed deletion of TS 3.3.9 to ensure that the LCO no longer satisfies the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The NRC staff also reviewed the non-MODE dependent applicability during movement of fuel assemblies within containment and in the fuel storage pool. As detailed in Sections 3.1 through 3.6 of this SE, the remaining accident analyses applicable to the permanently shutdown and defueled reactors of SONGS Units 2 and 3 show that the dose

consequences within the control room are acceptable. Furthermore, the analyses do not rely on the functioning of any SSCs to mitigate the postulated DBAs with the exception of the passive fuel storage pool structure.

The NRC staff evaluated the remaining DBAs that credited the CREACUS, and its support systems, that were previously relied upon to mitigate the CR, EAB or LPZ dose consequences during reactor operation. This includes the FHIS, the FHB PACU filtration system, and the CRIS. As discussed in the basis for deleting CREACUS TS 3.7.11, (see Section 3.7.13 of this SE), the CRIS is no longer required for providing airborne radiological protection for the control room operators in the event of a DBA. Since TS 3.3.9 exists solely to support CREACUS Operability, the elimination of the need for the CREACUS also obviates the need for its support systems. The deletion of the CREACUS TS 3.7.11 eliminates the need for the CRIS TS 3.3.9. Therefore, the NRC staff finds the CRIS isolation signal is no longer required and that the licensee's proposed change to delete TS 3.3.9 is acceptable.

3.7.10 Section 3.4, Reactor Coolant System

The RCS TSs of Section 3.4, "Reactor Coolant System (RCS)," for SONGS Units 2 and 3 contain the LCOs, Actions, and SRs that provides for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility. The following TSs are being proposed for deletion.

TS 3.4.1, "RCS DNB [Departure from Nucleate Boiling] (Pressure, Temperature, and Flow) Limits," specifies the process variables requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum DNBR will be met for each of the analyzed transients. TS 3.4.1 is applicable in MODE 1.

TS 3.4.2, "RCS Minimum Temperature for Criticality," specifies the requirements for RCS loop cold leg temperature ($T_c$) before the reactor can be made critical and while the reactor is critical. Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} > 1.0$) outside a temperature operating range of 522 °F to 558 °F, and to prevent operation in an unanalyzed condition. TS 3.4.2 is applicable in "MODE 1, THERMAL POWER ≤ 30% RTP and $T_c < 535$ °F, and in MODE 2, $K_{eff} \geq 1.0$ and $T_c < 535$ °F."

TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," specifies that the RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits as specified in the Pressure - Temperature Limits Report (PTLR). The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB. TS 3.4.3 is applicable at all times. The purpose for TS LCO 3.4.3 during normal operation of the RCS is to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. The RCS P/T limits in LCO 3.4.3 provide a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR Part 50, Appendix G. Although the P/T limits were developed to provide guidance for operation during heatup, or

cooldown, or inservice leak and hydrostatic testing, the applicability of these limits is at all times in keeping with the concern for nonductile failure.

TS 3.4.3.1, “Pressurizer Heatup and Cooldown Limits,” requires that the pressurizer heatup and cooldown rates shall be maintained within the specified limits. The pressurizer is designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. Therefore, TS 3.4.3.1 is applicable at all times.

TS 3.4.4, “RCS Loops - MODES 1 and 2,” specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODES 1 and 2. The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service. TS 3.4.4 is applicable in MODES 1 and 2.

TS 3.4.5, “RCS Loops - MODE 3,” specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 3. In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the SGs, to the secondary plant fluid. TS 3.4.5 is applicable in MODE 3.

TS 3.4.6, “RCS Loops - MODE 4,” specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 4. In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to the SGs or shutdown cooling (SDC) heat exchangers. TS 3.4.6 is applicable in MODE 4.

TS 3.4.7, “RCS Loops - MODE 5, Loops Filled,” specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops filled with coolant. In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the SGs or SDC heat exchangers. While the principal means for decay heat removal is via the SDC heat exchangers, the SGs are specified as a backup means for redundancy. TS 3.4.7 is applicable in MODE 5 with the RCS loops filled.

TS 3.4.8, “RCS Loops - MODE 5, Loops Not Filled,” specifies the requirements to ensure heat removal capability of the RCS loops with the reactor in MODE 5 with the RCS loops not filled with reactor coolant. In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the SDC heat exchangers. The SGs are not available as a heat sink when the loops are not filled. TS 3.4.8 is applicable in MODE 5 with the RCS loops not filled.

TS 3.4.9, “Pressurizer,” specifies the OPERABILITY requirements for the RCS pressurizer. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium

under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. TS 3.4.9 is applicable in MODES 1, 2, and 3.

TS 3.4.10, "Pressurizer Safety Valves," specifies the OPERABILITY and lift setpoint parameters for the pressurizer safety valves. The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The pressurizer safety valves are designed to prevent the RCS from exceeding the system safety limit of 2750 pounds per square inch absolute (psia) in MODES 1, 2, and 3. In MODES 4, 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP) System. TS 3.4.10 is applicable in MODES 1, 2, and 3.

TS 3.4.12.1, "Low Temperature Overpressure Protection (LTOP) System, RCS Temperature ≤ PTLR Limit," specifies the requirements for controlling RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G. TS LCO 3.4.12.1 provides RCS overpressure protection by minimizing coolant input capability and having adequate pressure relief capacity. In MODES 1, 2, and 3, the pressurizer safety valves will prevent RCS pressure from exceeding limits. In MODE 4 when the temperature of any RCS cold leg is less than or equal to the enable temperature specified in the PTLR, MODE 5, and MODE 6 when the reactor vessel head is on and the RCS is not vented, overpressure prevention falls to the OPERABLE SDC system relief valve or to a depressurized RCS and a sufficient sized RCS vent. When the reactor vessel head is off, overpressurization cannot occur.

TS 3.4.12.2, "Low Temperature Overpressure Protection (LTOP) System, RCS Temperature ≥ PTLR Limit," specifies requirements for controlling RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G. TS LCO 3.4.12.2 provides RCS overpressure protection by having adequate pressure relief capacity. In MODES 1, 2, and 3 the pressurizer safety valves will prevent RCS pressure from exceeding limits. In MODE 4 when the temperature of all RCS cold legs are greater than the enable temperature specified in the PTLR, overpressure prevention falls to the OPERABLE SDC system relief valve or to an OPERABLE pressurizer code safety valve.

TS 3.4.13, "RCS Operational LEAKAGE," specifies the process variable limits and operating restrictions for RCS pressure boundary leakage, unidentified RCS leakage, identified RCS leakage, and primary to secondary leakage. RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event. The primary to secondary leakage limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures (SGTRs). TS 3.4.13 is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, leakage limits are not required because the

reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for leakage.

TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," specifies the process variable limits and operating restrictions for RCS PIV leakage. The regulations in 10 CFR 50.2, 10 CFR 50.55a(c), and 10 CFR Part 50, Appendix A, GDC 55, discuss RCPB valves, which are normally closed valves in series within the RCPB that separate the high pressure RCS from an attached low pressure system. Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems (intersystem LOCA). PIVs are provided to isolate the RCS from the following typically connected systems: SDC system; safety injection system; and the chemical and volume control system. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. TS 3.4.14 is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for PIV leakage outside the containment.

TS 3.4.15, "RCS Leakage Detection Instrumentation," specifies the OPERABILITY requirements for RCS leakage detection instrumentation. Leakage detection systems are provided to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, they provide an early indication or warning signal to permit proper evaluation of RCS leakage into the containment area. TS LCO 3.4.15 requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS leakage indicates possible RCPB degradation. TS 3.4.15 is applicable in MODES 1, 2, 3, and 4.

TS 3.4.16, "RCS Specific Activity," specifies the process variable limits and operating restrictions for Dose Equivalent 1-131 and gross specific activity. The TS LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite doses meet the appropriate RG 1.183 acceptance criteria following a SGTR accident. TS 3.4.16 is applicable in MODES 1, 2, and MODE 3 with RCS average temperature ≥ 500 degrees F.

TS 3.4.17, "Steam Generator (SG) Tube Integrity," specifies the requirements to ensure the RCPB integrity function of the SG. The SGTR accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. TS 3.4.17 is applicable in MODES 1, 2, 3, and 4.

The licensee proposed to delete all of Section 3.4 of the SONGS Units 2 and 3, RCS TSs, since all except TS 3.4.3, are only applicable to operating reactor MODES and do not apply to a permanently shutdown and defueled reactor. The NRC staff has reviewed these proposed changes and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain functionality and integrity of the RCS pressure boundary. These TSs contain requirements for various RCS parameters such as: thermal limitations for heatup and cooldown rates during plant operation in order to operate within the analyzed requirements for stress intensity and fatigue limits for the reactor vessel; pressurization, which established and maintained an equilibrium under saturated conditions for pressure control to prevent bulk boiling

in the remainder of the RCS; coolant chemistry, which included limits on RCS activity to limit potential offsite doses due to postulated events and limits on RCS conductivity, chlorides, and pH to prevent stress-corrosion cracking; coolant leakage, which established primary system leakage limits to allow prompt identification and isolation of leaks before the integrity of the RCS pressure boundary was impaired; safety and relief valves, which specifies operability requirements for the safety and relief valves designed to prevent overpressurization of, and damage to, the primary system boundary; and structural integrity, which addresses the inservice inspection requirements of the primary system boundary components. All of these TSs are related to assuring the integrity of the RCS pressure boundary. The RCS TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, the RCS is no longer functional or used in any capacity.

Regarding the applicability of TS 3.4.3 and TS 3.4.3.1 at all times, the NRC staff notes that the RCS and pressurizer are drained and vented, to the extent possible, and consequently there is no longer any concern about exceeding the RCS and pressurizer P/T or cyclic limits. The requirements of 10 CFR Part 50, Appendix G, no longer apply to a permanently shutdown and defueled reactor because the RCPB will no longer be used as a fission product barrier. Therefore, TS 3.4.3 is no longer needed and may be deleted. Similarly, operating the unit within the fatigue analysis performed in accordance with the ASME Code Section III requirements no longer applies. Therefore, TS 3.4.3.1 is no longer needed and may also be deleted.

The NRC staff has also reviewed the RCS TSs proposed for deletion (TS 3.4.1, TS 3.4.2, TS 3.4.3, TS 3.4.3.1, TS 3.4.4, TS 3.4.5, TS 3.4.6, TS 3.4.7, TS 3.4.8, TS 3.4.9, TS 3.4.10, TS 3.4.12.1, TS 3.4.12.2, TS 3.4.13, TS 3.4.14, TS 3.4.15, TS 3.4.16, and TS 3.4.17), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this SE. The staff notes that these TSs indicate MODES for which these TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2) for SONGS Units 2 and 3, it is prohibited from operating the reactors or placing fuel in the reactor vessels and, therefore, SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply. Furthermore, because irradiated fuel has been permanently removed from the reactor pressure vessels, the RCS is no longer relevant as a fission product barrier.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.4, Reactor Coolant System, is acceptable.

### 3.7.11 Section 3.5, Emergency Core Cooling Systems (ECCS)

Section 3.5 of the SONGS Units 2 and 3 TSs, "Emergency Core Cooling Systems (ECCS)," contains LCOs that provide for appropriate functional capability of ECCS equipment required for

mitigation of DBAs or transients so as to protect the integrity of a fission product barrier. The following TSs are being proposed for deletion.

TS 3.5.1, "Safety Injection Tanks (SITs)," specifies the requirements for the SITs to ensure they are capable of supplying water to the reactor vessel during the blowdown phase of a LOCA, to provide inventory to help accomplish the refill phase that follows thereafter, and to provide RCS makeup for a small-break LOCA. TS 3.5.1 is applicable in MODES 1 and 2, and in MODE 3 with pressurizer pressure ≥ 715 psia.

TS 3.5.2, "ECCS - Operating," specifies the requirements for the ECCS trains so as to provide core cooling and negative reactivity to ensure that the reactor core is protected after a LOCA, CEA ejection accident, loss of secondary coolant accident (including uncontrolled steam release), and SGTR. The ECCS consists of the high pressure safety injection (HPSI) and the low pressure safety injection (LPSI) subsystems. TS 3.5.2 is applicable in MODES 1 and 2, and MODE 3 with pressurizer pressure ≥ 400 psia.

TS 3.5.3, "ECCS - Shutdown," specifies the requirements for ECCS with the reactor in MODE 3 with pressurizer pressure < 400 psia, and in MODE 4. In these MODES, an ECCS train is composed of a single HPSI subsystem. One OPERABLE ECCS train is acceptable without a single failure consideration, based on the stable reactivity condition of the reactor and the limited core cooling requirements.

TS 3.5.4, "Refueling Water Storage Tank (RWST)," specifies the requirements for RWST OPERABILITY. During accident conditions, the RWST provides a source of borated water to the ECCS and containment spray system pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. TS 3.5.4 is applicable in MODES 1, 2, 3, and 4 because RWST OPERABILITY requirements are dictated by ECCS and containment spray system OPERABILITY requirements. Since both the ECCS and the containment spray system must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation.

TS 3.5.5, "Trisodium Phosphate (TSP) Dodecahydrate," specifies the requirements for TSP crystals to be placed in baskets on the floor of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a LOCA, remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident. TS 3.5.5 is applicable in MODES 1, 2, and 3, when the RCS is at elevated temperature and pressure, providing an energy potential for a LOCA.

The NRC staff has reviewed the proposed changes to the ECCS TSs and has determined that these TSs are only needed to provide the LCOs and SRs necessary to maintain functionality of the systems that provide emergency cooling to the reactor core and assure the appropriate functional capability ECCS required for mitigation of DBAs when the reactor is in MODES 1 through 4. These TSs includes multiple LCOs addressing the SITs, HPSI and LPSI subsystems; part of the ECCS, designed to provide adequate emergency cooling capability to

the reactor in the event of a LOCA; the RWST, designed to supply borated water to the ECCS during accident conditions; and the TSP baskets to help retain iodine in solution. All of these TSs are related to provide cooling for a reactor vessel core. Since SONGS Units 2 and 3 are permanently shut down and defueled, there are no accidents of any kind that would require emergency core cooling and the accidents that these systems and components were designed to mitigate are no longer possible.

The NRC staff also reviewed the ECCS TSs proposed for deletion (TS 3.5.1, TS 3.5.2, TS 3.5.3, TS 3.5.4, and TS 3.5.5), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels, and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.5, Emergency Core Cooling Systems, is acceptable.

3.7.12 Section 3.6, Containment Systems

Section 3.6 of SONGS Units 2 and 3 TSs, "Containment Systems," contains the LCOs, Actions, and SRs that provide for appropriate control of process variables, design features, or operating restrictions required to protect the integrity of the containment as a fission product barrier; and appropriate functional capability of ESF equipment required for mitigation of DBAs or transients so as to protect the integrity of containment. The following TSs are being proposed for deletion from this section.

TS 3.6.1, "Containment," specifies the requirements for the containment to ensure it is capable of withstanding the pressures and temperatures of the limiting DBA without exceeding the design leakage rate. The containment steel liner and its penetrations establish the leakage limiting boundary of the containment. This TS provides the operating restrictions required to protect the integrity of the containment as a fission product barrier and limits the leakage of fission product radioactivity from the containment to the environment. TS 3.6.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.2, "Containment Air Locks," specifies the requirements for the structural integrity and leak tightness of the containment air locks. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each containment air lock's structural integrity and leak tightness is essential to the successful mitigation of such an event. TS 3.6.2 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.3, “Containment Isolation Valves,” specifies the requirements for the isolation capability of the containment via the containment isolation valves. Containment isolation valves form a part of the containment boundary and their OPERABILITY supports leak tightness of the containment. TS 3.6.3 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.4, “Containment Pressure,” specifies the limitations on internal containment pressure. Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the maximum allowed containment internal pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the accidental actuation of the Containment Spray System. TS 3.6.4 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.5, “Containment Air Temperature,” specifies the limitations on containment average air temperature. Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature is maintained below its design temperature and that required safety-related equipment will continue to perform its function. TS 3.6.5 is applicable in MODES 1, 2, 3, and 4.

TS 3.6.6.1, “Containment Spray and Cooling Systems,” specifies the operability requirements for containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. The containment spray system consists of two separate trains. Each train includes a containment spray pump, spray headers, valves and piping. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump. Two trains of containment cooling, each of sufficient capacity to supply 50 percent of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from the component cooling water system. All four fans are required to furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the SG compartments and pressurizer compartment. TS 3.6.6.1 is applicable in MODES 1, 2, and 3.

TS 3.6.6.2, “Containment Cooling Systems,” specifies the operability requirements for containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. Two trains of containment cooling, each of sufficient capacity to supply 50 percent of the design cooling requirement, are provided. Two trains with two fan units each are supplied with cooling water from the component cooling water system. All four fans are required to

furnish the design cooling capacity. Air is drawn into the coolers through the fans and discharged to the SG compartments and pressurizer compartment. TS 3.6.6.2 is applicable in MODE 4, when a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the containment cooling trains.

TS 3.6.8, "Containment Dome Air Circulators," specifies the requirements for the containment dome air circulators to reduce the potential for breach of the containment due to a hydrogen oxygen reaction. The dome air circulators accelerate the air mixing process between the upper dome space of the containment atmosphere during LOCA operations. They also prevent any hot spot air pockets during the containment cooling mode and avoid any hydrogen concentration in pocket areas. Two dome air circulator trains are required to be operable. Each train consists of two fans with their own motors and controls and is automatically initiated by a containment cooling actuation signal (CCAS). While each train has two fans, only one operable fan is required for the train to be operable, since each fan can provide the necessary flow rate to adequately mix the containment atmosphere. TS 3.6.8 is applicable in MODES 1 and 2.

The NRC staff has reviewed the licensee's proposed changes for Section 3.6 of the SONGS Units 2 and 3 TSs and has determined that the TSs are only needed to provide the LCOs and SRs necessary to maintain functionality of the containment. These TSs include multiple LCOs addressing containment integrity, which includes: containment pressure, containment air temperature, and containment air locks, which forms part of the containment pressure boundary; and containment isolation valves, designed to isolate the containment in the event of a LOCA to prevent the release of fission products to the atmosphere; containment spray and cooling, which limit post-accident pressure and temperature in containment; and dome air circulators that help reduce the potential hydrogen concentration pockets in containment following a design basis accident. All of these TSs are related to assuring the integrity of the containment as a fission product boundary. The containment TSs are only important for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3, the containment SSCs are no longer functional or used in any capacity and the associated TSs are no longer meaningful.

The NRC staff also reviewed the containment TSs proposed for deletion (TS 3.6.1, TS 3.6.2, TS 3.6.3, TS 3.6.4, TS 3.6.5, TS 3.6.6.1, TS 3.6.6.2, and TS 3.6.8), to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels, and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 3.6, Containment Systems, is acceptable.

3.7.13 Section 3.7, Plant Systems

Section 3.7 of the SONGS Units 2 and 3 TSs, "Plant Systems," contains the LCOs, Actions, and SRs that provide for appropriate functional capability of balance-of-plant equipment required for safe operation of the facility. This section contains operability requirements related to the steam generators, feedwater system, cooling water, ventilation, and spent fuel storage.

The licensee proposed deletion of the following LCOs in Section 3.7 of the SONGS Units 2 and 3 TSs:

TS 3.7.1, "Main Steam Safety Valves (MSSVs)," specifies the requirements for the MSSVs to ensure they are capable of providing overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the RCPB by providing a heat sink for the removal of energy from the RCS, if the preferred heat sink provided by the condenser and circulating water system, is not available. TS 3.7.1 is applicable in MODES 1, 2, and 3.

TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," specifies the requirements for the MSIVs to ensure that they are capable of isolating steam flow from the secondary side of the SGs following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator. One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the MSSVs, atmospheric dump valves, and auxiliary feedwater pump turbine steam supplies to prevent them from being isolated from the SGs by MSIV closure. Closing the MSIVs isolates each SG from the other, and isolates the turbine, steam bypass system, and other auxiliary steam supplies from the steam generators. TS 3.7.2 is applicable in MODES 1, 2, and 3 except when all MSIVs are closed and deactivated.

TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)," specifies the requirements for MFIVs. The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a HELB. Closure of the MFIVs terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs. TS 3.7.3 is applicable in MODES 1, 2, and 3 except when MFIV is closed and deactivated.

TS 3.7.4, "Atmospheric Dump Valves (ADVs)," specifies the requirements for providing a method for cooling the unit to shutdown cooling system entry conditions, should the preferred heat sink via the steam bypass system to the condenser not be available. This is done in conjunction with the auxiliary feedwater (AFW) system providing cooling water from the condensate storage tank (CST). TS 3.7.4 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.5, “Auxiliary Feedwater (AFW) System,” specifies the requirements to ensure that the AFW system automatically supplies feedwater to the steam generators to remove decay heat from the RCS upon the loss of normal feedwater supply. TS 3.7.5 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.6, “Condensate Storage Tank (CST T-121 and T-120),” specifies the requirements to ensure a safety grade source of water to the SGs for removing decay and sensible heat from the RCS. The CSTs provide a passive flow of water, by gravity, to the AFW System. TS 3.7.6 is applicable in MODES 1, 2, and 3, and in MODE 4 when steam generator is relied upon for heat removal.

TS 3.7.7, “Component Cooling Water (CCW) System,” specifies the requirements to ensure that the CCW system provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, the CCW system also provides this function for various nonessential components. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the salt water cooling system, and thus to the environment. TS 3.7.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.7.1, “Component Cooling Water (CCW) Safety Related Makeup System,” specifies the requirements to ensure a safety-related CCW makeup system is available to maintain the water inventory in the CCW trains during a 7-day post-accident period. The safety-related makeup system is designed to supply water to the CCW trains following loss of normal CCW makeup from the nuclear service water system. For this purpose, sufficient water inventory is contained in the single primary plant makeup (PPMU) storage tank for both CCW trains. From the PPMU tank, water is transferred to the CCW return heads by two safety-related pumps. TS 3.7.7.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.8, “Salt Water Cooling (SWC) System,” specifies the requirements to ensure that the SWC system provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, and a normal shutdown, the SWC system also provides this function for various safety-related and nonsafety-related components. The safety-related function is covered by TS 3.7.8. TS 3.7.8 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.10, “Emergency Chilled Water (ECW),” specifies the requirements to ensure that the ECW system provides a heat sink for the removal of process and operating heat from selected safety-related air handling systems during a DBA or transient. The design basis of the ECW system is to remove the post-accident heat load from ESF spaces following a DBA coincident with a loss of offsite power. Each train provides chilled water to the HVAC units at the design temperature and flow rate. TS 3.7.10 is applicable in MODES 1, 2, 3, and 4.

TS 3.7.19, “Secondary Specific Activity,” specifies the limit on secondary coolant specific activity during power operation to minimize releases to the environment because of normal operation, AOOs, and accidents. The accident analysis of the main steam line break (MSLB) assumes an

initial secondary coolant specific activity used for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed the TEDE limit. TS 3.7.19 is applicable in MODES 1, 2, 3, and 4.

The NRC staff has reviewed the proposed changes to TSs 3.7.1 through TS 3.7.10, and TS 3.7.19 and has determined that these TSs are only necessary to assure the operability of certain plant systems during reactor operation. These TSs involve: MSSVs, which provide overpressure protection for the secondary system; MSIVs, which isolate steam flow from the secondary side of the steam generator following a MSLB; MFIVs, which isolate main feedwater flow to the secondary side of the steam generators following a HELB; ADVs, which provide a method for cooling the unit should the condenser not be available; AFW system, which supplies feedwater to the steam generators upon the loss of the normal feedwater supply; CSTs, which provide the preferred source of water to the steam generators for removing decay and sensible heat from the RCS; CCW system, CCW safety-related makeup system, and the SWC system, which provide a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient to the ultimate heat sink; the ECW system that removes heat from ESF spaces through safety-related air handling systems; and secondary specific activity, which specifies a limit on secondary coolant specific activity during power operation.

The above TSs were intended to protect the fuel in the reactor from potential operational transients and accidents. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel. Consequently, there are no longer any transient or accident conditions that these systems and components protect against or mitigate. Therefore, the NRC staff finds the deletion of 3.7.1 through 3.7.10, as detailed above, is acceptable. TS 3.7.19 provides the operational limits on secondary coolant specific activity limiting the potential radiological consequences of an accident that could release pressurized steam from the SGs. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel, there is no source of heat available to pressurize the SGs and no source of activity. Therefore, the NRC staff finds the deletion of TS 3.7.19 is acceptable.

The NRC staff has also reviewed Section 3.7 of the SONGS Units 2 and 3, TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10, and TS 3.7.19 proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TSs are applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Therefore, based on the evaluation above, the NRC staff finds that the licensee's proposed change to delete Plant Systems TS 3.7.1, TS 3.7.2, TS 3.7.3, TS 3.7.4, TS 3.7.5, TS 3.7.6, TS 3.7.7, TS 3.7.7.1, TS 3.7.8, TS 3.7.10 and 3.7.19, is acceptable.

TS 3.7.11, "Control Room Emergency Air Cleanup System (CREACUS)," specifies the requirements to ensure that the CREACUS provides a protected environment from which operators can control SONGS Units 2 and 3, following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The CREACUS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREACUS train consists of an emergency air conditioning unit, emergency ventilation air supply unit, emergency isolation dampers, and cooling coils and two cabinet coolers. Each emergency air conditioning unit includes a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. A second bank of HEPA filters follows the adsorber section to collect carbon fines. Ductwork, motor-operated dampers, doors, barriers, and instrumentation also form part of the system. Upon receipt of the actuating signal, normal air supply to the CRE is isolated and the stream of ventilation air is recirculated through the system's filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

There are two CREACUS operational modes. Emergency mode is an operational mode when the control room is isolated to protect operational personnel from radioactive exposure through the duration of a DBA. Isolation mode is an operational mode when the CRE is isolated to protect operational personnel from toxic gases and smoke. Actuation of the CREACUS places the system into either of two separate states of operation, depending on the initiation signal. Actuation of the system to either the emergency mode or isolation mode of CREACUS operation closes the unfiltered-outside-air intake and unfiltered exhaust dampers and aligns the system for recirculation of air within the CRE through the redundant trains of HEPA and charcoal filters. The emergency mode also initiates pressurization of the CRE. Outside air is added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. The CRE supply and the outside air supply of the normal control room HVAC are monitored by radiation and toxic-gas detectors, respectively. One detector output above the setpoint will cause actuation of the emergency mode or isolation mode as required. The actions of the isolation mode are more restrictive, and will override the actions of the emergency mode of operation. TS 3.7.11 is applicable in MODES 1, 2, 3, 4, 5, and 6 and during movement of fuel assemblies in the containment or fuel storage pool.

When SONGS Units 2 and 3 were authorized to operate, the CREACUS provided a protected environment from which operators could control the units following postulated accidents involving an uncontrolled release of radioactivity, including an FHA. Prior to SONGS Units 2 and 3 permanently shutting down and defueling, the TSs for the CREACUS provided the LCOs and SRs necessary to maintain the control room environment following an accident. Specifically, during irradiated fuel movement, the CREACUS provided a protected environment from which operators can control the unit following a postulated uncontrolled release of radioactivity from an FHA. CRE will remain habitable during and following a DBA. In MODES 5

and 6, the CREACUS is required to cope with the release from a rupture of a waste gas tank. During movement of fuel assemblies, the CREACUS must be operable to cope with the release from an FHA.

The licensee provided information on the toxic gases isolation of CREACUS in Section 3.2.10.2.3 of the licensee's permanently defueled technical specification amendment request. Specifically, per the NRC's SE associated with the issuance of SONGS License Amendment Nos. 127 and 116 for Units 2 and 3, respectively dated February 9, 1996 (ADAMS Accession No. ML021990684), the toxic gas isolation of CREACUS is not relied on to prevent or mitigate a design basis accident or transient because the plant design includes other means to safely shut down the plant if the control room becomes uninhabitable. As such, the toxic gas isolation instrumentation was relocated from the TS and placed in the Licensee Controlled Specifications with an applicability of MODES 1, 2, 3, 4, 5, and 6. Since an NRC SE has already accepted the removal of toxic gas isolation of CREACUS from the TSs, a new NRC staff determination is not required. The staff concludes that automatic toxic gas isolation of CREACUS is not required during movement of fuel assemblies in the fuel storage pool at the permanently defueled SONGS Units 2 and 3.

With the termination of reactor operations at SONGS Units 2 and 3 and the permanent removal of the fuel from the reactor core in each unit, the postulated accidents involving failure or malfunction of the reactor, RCS, or secondary system are no longer applicable. While there are no transients that continue to apply to SONGS Units 2 and 3, there are still postulated DBAs. As discussed in Sections 3.1 through 3.6 of this SE, the remaining DBAs applicable to the defueled reactors of SONGS Units 2 and 3 show that the dose consequences are acceptable without relying on SSCs remaining functional for accident mitigation during and following the event, with the exception of the SFP structure.

The NRC staff evaluated these accident analyses and confirmed that no ESF system is used to mitigate the CR, EAB, or LPZ dose consequences. This includes no credit for the FHIS, the PACU filtration system, the CRIS and the CREACUS. Since SONGS Units 2 and 3 are permanently shutdown and defueled, and greater than 17 months of decay time has elapsed since permanent shut down, the remaining DBAs applicable to the facility demonstrate that the dose consequences within the CR are acceptable without relying on SSCs remaining functional for accident mitigation, including an FHA in the FHB. (The one exception to this is the continued function of the passive fuel storage pool structure, which will be maintained as a TS for SONGS Units 2 and 3.)

In summary, the radiological consequences of the remaining DBAs for SONGS Units 2 and 3 assume no credit for CR isolation or recirculation filtration and no credit for any accident mitigation by the FHB ventilation system. Calculated doses at the EAB, LPZ, and CR are within 10 CFR Part 50.67 limits, and RG 1.183 dose limits. Since the DBA accident analysis for SONGS Units 2 and 3 assumed no credit for control room post-accident recirculation system emergency ventilation or filtration, the CREACUS is no longer required. Therefore, isolation of the CRE via the CRIS and CREACUS is not necessary for any of the postulated DBAs. As noted before, the intent of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) is to capture into TSs those SSCs that are part of the primary success path of a safety sequence analysis. With SONGS

Units 2 and 3 permanently shutdown and defueled, and the irradiated fuel having decayed for a minimum period of 17 months, the CREACUS is no longer needed or credited in the primary success path of a safety sequence analysis related to an accident. Since the radiological consequences of the accident analyses are within the appropriate acceptance criteria without credit for the CREACUS, the NRC staff finds that the licensee's proposed change to delete TS 3.7.11, is acceptable.

The licensee intends to retain TS 3.7.16, TS 3.7.17, and TS 3.7.18 but revise these TSs to delete the REQUIRED ACTIONS note that states that LCO 3.0.3 is not applicable. LCO 3.0.3 is being deleted from the SONGS Units 2 and 3 TSs and removal of a reference to TS 3.0.3 is a conforming change. In addition, with the deletion of all TSs in Section 3.1 through 3.6, the licensee also proposes to renumber these TSs to 3.1.1 to 3.1.3, respectively

TS 3.7.16, "Fuel Storage Pool Water Level," specifies the requirements to ensure that the minimum water level in the SFP meets the assumptions of iodine decontamination factors following an FHA. The water also provides shielding during the movement of spent fuel. This TS is applicable during movement of irradiated fuel assemblies in the SFP. The licensee has proposed to retain this TS in the permanently defueled TSs essentially unchanged. The Note in Required Action A.1 (LCO 3.0.3 is not applicable), is being deleted to conform to the deletion of TS LCO 3.0.3. The licensee has also proposed to renumber this TS as 3.1.1, based on the proposed deletion of all the preceding TSs.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA. TS 3.7.16, "Fuel Storage Pool Water Level," specifies the TS required LCOs and SRs that ensure the minimum water level in the SFP meets the assumptions of iodine decontamination factors following an FHA or cask drop accident.

SCE's analysis of the postulated FHA or cask drop accident assumes that there is at least 23 feet of water between the top of the damaged fuel assemblies and the fuel pool surface. The gap activity in the damaged rods is assumed to be instantaneously released into the SFP. Radionuclides in the gap release are assumed to be filtered by the 23 feet of water before emerging from the SFP. The activity exhaust rate from the auxiliary building is established to complete the release in 2 hours, as required by RG 1.183, but does not credit the auxiliary building ventilation for any mitigation of the release.

Since the 23-foot water level of the SFP is an initial condition of the FHA and the cask drop DBA, it satisfies Criterion 2 for inclusion in TSs and is being retained for SONGS Units 2 and 3 in their permanently shutdown and defueled condition. The amendment request by SCE does not involve any change to the technical language in the TS. The discussion in this evaluation of the SFP water level TS is provided only for completeness since the SFP water level is an important initial condition in the FHA and cask drop accident analysis and will continue to be part of the SONGS TSs.

TS 3.7.17, "Fuel Storage Pool Boron Concentration," specifies the requirements to ensure that the SFP boron concentration is > 2000 parts per million (ppm). The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in the criticality analysis of record, which is that a minimum of 2000 ppm of boron is needed to ensure that criticality does not occur during the worst case fuel loading accident. This concentration of dissolved boron is the minimum required for fuel assembly storage and movement within the spent fuel pool. This TS is applicable whenever fuel assemblies are stored in the spent fuel pool. This TS is being retained in the permanently defueled TS essentially unchanged. The Note in Required Action A.1 (LCO 3.0.3 is not applicable), is being deleted to conform to the deletion of TS LCO 3.0.3. The licensee has also proposed to renumber this TS as 3.1.2, based on the proposed deletion of all the preceding TSs.

TS 3.7.18, "Spent Fuel Assembly Storage," specifies the restrictions on the placement of fuel assemblies within the SFP, in accordance with Figure 3.1.3-1 through Figure 3.1.3-4 in the accompanying LCO, to ensure the keff of the SFP will always remain < 0.95, assuming the pool to be flooded with unborated water. This TS applies whenever any fuel assembly is stored in the spent fuel pool. TS 3.7.18 is being retained in the permanently defueled TS essentially unchanged. The Note in Required Action A.1 (LCO 3.0.3 is not applicable), is being deleted to conform to the deletion of TS LCO 3.0.3. The licensee has also proposed to renumber this TS as 3.1.3, based on the proposed deletion of all the preceding TSs.

The NRC staff reviewed the proposed deletion of the reference to LCO 3.0.3 in the Required Actions Note in TS 3.7.16, TS 3.7.17, and TS 3.7.18. The staff finds that deletion of the Note, "LCO 3.0.3 is not applicable," in each of these TSs is appropriate and the conforming change to the deletion of TS LCO 3.0.3, as discussed in Section 3.7.6 of this SE. Therefore, the staff finds that the licensee's proposed change to delete the reference to LCO 3.0.3 in TS 3.7.16, TS 3.7.17, and TS 3.7.18 (renumber as TS 3.1.1, TS 3.1.2, and TS 3.1.3, respectively – see below), is acceptable.

The NRC staff also reviewed the proposed change to renumber TS 3.7.16, 3.7.17, and TS 3.7.18, to TS 3.1.1, TS 3.1.2, and TS 3.1.3, respectively, and found the change to be editorial and conforming to the overall changes to the TSs. Therefore, the NRC staff finds that the licensee's proposed renumbering of the TSs is acceptable.

3.7.14 Section 3.8, Electrical Power Systems

The licensee proposed to delete SONGS Units 2 and 3 electrical power systems TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9, since these TSs are MODE dependent and only applicable to an operating reactor. Therefore, these TSs do not apply to the permanently shutdown and defueled condition of SONGS Units 2 and 3.

TS 3.8.1, "AC [Alternating Current] Sources - Operating," specifies the requirements to ensure that the offsite power sources (normal preferred and alternate preferred power sources), and the standby power sources (Train A and Train B DGs), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that

the fuel, RCS, and containment design limits are not exceeded. TS 3.8.1 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.4, "DC [Direct Current] Sources - Operating," specifies the requirements to ensure that the DC electrical power subsystems (with each subsystem consisting of one battery, the required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem) are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or postulated DBA. TS 3.8.4 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.7, "Inverters - Operating," specifies the requirements to ensure that required inverters are OPERABLE such that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. These requirements include the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.7 is applicable in MODES 1, 2, 3, and 4.

TS 3.8.9, "Distribution Systems - Operating," specifies the requirements to ensure availability of AC, DC, and AC instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. The AC, DC, and AC vital electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.9 is applicable in MODES 1, 2, 3, and 4.

The NRC staff has reviewed the SONGS Units 2 and 3 electrical power systems LCOs in TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9, which have been proposed for deletion, to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for the permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels. Therefore, SONGS Units 2 and 3 are no longer in a configuration or a condition under which these TS MODES apply. Based on the above, the staff finds the deletion of TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9 from TS Section 3.8, Electrical Systems, is acceptable.

The licensee has also proposed to delete SONGS Units 2 and 3 electrical power systems TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10, based on the MODE dependent applicably of these TSs. However, these TSs are also directly applicable during the movement of irradiated fuel assemblies or are support systems for TSs required during the movement of irradiated fuel assemblies. The following evaluations of these electrical power systems TSs assess the licensee's justification as to why these TSs do not apply to the

permanently shutdown and defueled condition of SONGS Units 2 and 3 during movement of irradiated fuel assemblies.

TS 3.8.2, "AC Sources - Shutdown," specifies the requirements to ensure that the offsite power sources (normal preferred and alternate preferred power sources), and the standby power sources (Train A and Train B DGs), provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded. TS 3.8.2 is applicable during MODES 5 and 6 and during movement of fuel assemblies in containment or in the fuel storage pool.

TS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," provides for proper operation of the DGs, by specifying the parameters and ensuring there will be sufficient quantity and proper quality of the fuel oil, lube oil, and starting air systems. Stored diesel fuel oil is required to have sufficient supply for 7 days of rated load operation for each DG. It is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate each DG at rated load for 7 days. Lastly, each DG is equipped with two air start systems, which have adequate capacity for five successive start attempts on the DG without recharging the air start receivers. TS 3.8.3 is applicable whenever the DGs are required to be operable.

TS 3.8.5, "DC Sources Shutdown," specifies the requirements to ensure availability of the DC electrical power system and subsystems (with each subsystem consisting of one battery, required battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the subsystem), in order to provide normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching. TS 3.8.5 is applicable during MODES 5 and 6 and during movement of fuel assemblies in containment or in the fuel storage pool.

TS 3.8.6, "Battery Parameters," specifies the requirements to ensure the limits on battery float current as well as electrolyte temperature, level, and float voltage for the DC power subsystem batteries. Battery parameters are required solely for the support of the associated DC electrical power subsystems (per TS 3.8.4 and TS 3.8.5). Therefore, battery parameter limits are only required (and TS 3.8.6 is only applicable) when the DC electrical power source is required to be operable.

TS 3.8.8, "Inverters – Shutdown," specifies the requirements to ensure stability and reliability of the preferred source of power for the 120 Volt AC vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The inverter provides an uninterruptible power source for the safety-related instrumentation and controls. TS 3.8.8 is applicable during MODES 5 and 6 and during movement of fuel assemblies within containment or in the fuel storage pool.

TS 3.8.10, "Distribution Systems - Shutdown," specifies the requirements for the onsite AC, DC, and AC instrument bus electrical power distribution systems. The TS specifies sufficient capacity, capability, redundancy, and reliability of the distribution system to ensure the availability of necessary power to ESF systems so that the fuel, RCS, and containment design

limits are not exceeded. TS 3.8.10 is applicable during MODES 5 and 6 and during movement of fuel assemblies within containment or in the fuel storage pool.

The SONGS Unit 2 and 3 TS Basis documents indicates that the shutdown electrical power systems TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10 provide assurance that:

a. The units can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the units status; and

c Adequate AC electrical power is provided to mitigate events postulated during shut down, such as an FHA.

The NRC staff has reviewed the SONGS Units 2 and 3 shutdown electrical power systems TSs (TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10), which have been proposed for deletion, to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The staff notes that these TSs indicate MODES for which the TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for the permanently shutdown and defueled reactors, such as SONGS Units 2 and 3, has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactors or placing fuel in the reactor vessels and SONGS Units 2 and 3 is no longer in a configuration or a condition under which these TS MODES apply. Based on the above, the NRC staff finds that the licensee's proposed change to delete TS 3.8.2, TS 3.8.3, TS 3.8.5, TS 3.8.6, TS 3.8.8, and TS 3.8.10, from TS Section 3.8, Electrical Systems, for MODES 5 and 6, is acceptable.

The NRC staff also reviewed the non-MODE dependent applicability during movement of fuel assemblies within containment and in the fuel storage pool. Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, the prohibition on placing fuel in the reactor vessel, it also precludes core alterations and the movement of fuel assemblies within containment. Therefore, these TSs are no longer needed during movement of fuel assemblies within containment.

As detailed in Sections 3.1 through 3.6 of this SE, the remaining accident analyses applicable to the permanently shutdown and defueled reactors of SONGS Units 2 and 3 show that the dose consequences within the control room are acceptable without relying on SSCs remaining functional for accident mitigation during any of the remaining DBAs, including FHAs. (The one exception to this is the continued function of the passive fuel storage pool structure, which will be maintained as a TS for SONGS).

For TS 3.8.2, AC Sources – Shutdown, the FHA is the applicable DBA related to the TS requirement for functional capability of AC sources (offsite power and DGs) during the TS

specified condition of during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3, do not rely on normal or emergency power for accident mitigation (including any need for providing airborne radiological protection), the AC sources are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events, with the exception of the SFP support structure. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for AC sources is no longer necessary because there are no design-basis events that rely on AC sources for mitigation. Consequently, AC sources no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.2, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air, during movement of fuel assemblies in the fuel storage pool. Since TS 3.8.3 exists solely to support the DG requirements of TS 3.8.1 and TS 3.8.2, the deletion of these TSs is consistent with the elimination of the need for DGs and also eliminates the need for the DG support systems. The NRC staff has determined that the requirement for DGs and associated supporting TSs are no longer necessary because the remaining DBAs for SONGS Units 2 and 3 do not rely on the DGs for mitigation. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.3, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.5, DC Sources – Shutdown, during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3, do not rely on safety-related DC sources of electrical power for accident mitigation (including any need for providing airborne radiological protection), the DC sources are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for DC sources is no longer necessary because there are no design-basis events that rely on DC sources for mitigation. Consequently, DC sources no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.5, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.6, Battery Parameters, during movement of fuel assemblies in the fuel storage pool. Since TS 3.8.6 exists solely to support the DC source requirements of TS 3.8.4 and TS 3.8.5, the deletion of these TSs is consistent with the elimination of the need for DC sources and also obviates the need for the battery support

systems. The staff has determined that the requirement for DC sources and associated supporting TSs are no longer necessary because the remaining DBAs for SONGS Units 2 and 3 do not rely on the DC sources for mitigation. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.6, battery parameters, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.8, Inverters – Shutdown, during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3 do not rely on inverters or the safety-related 120 Volt AC electrical power for accident mitigation (including any need for providing airborne radiological protection), the inverters are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events, with the exception of the SFP support structure. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for inverters is no longer necessary because there are no design-basis events that rely on inverters for mitigation. Consequently, inverters no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.8, during movement of fuel assemblies in the fuel storage pool, is acceptable.

The NRC staff has reviewed the need for TS 3.8.10, Distribution System – Shutdown, during movement of fuel assemblies in the fuel storage pool. Because the FHA analysis, and the other DBAs identified for SONGS Units 2 and 3, do not rely on the safety-related AC, DC and AC instrument bus electrical distribution systems for accident mitigation (including any need for providing airborne radiological protection), these safety-related distributions systems are not required during movement of fuel assemblies in the fuel storage pool for mitigation of a potential FHA or any of the other DBAs. Specifically, the accident analyses show that the dose consequences are acceptable without relying on any SSCs to remain functional during and following the postulated events, with the exception of the SFP support structure. Therefore, during movement of fuel assemblies in the fuel storage pool, there are no systems that function or actuate and are credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the DBA. As such, the requirement for safety-related AC, DC and AC instrument bus electrical distribution systems is no longer necessary because there are no design-basis events that rely on safety-related AC, DC and AC instrument bus electrical distribution systems for mitigation. Consequently, AC, DC and AC instrument bus electrical distribution systems no longer meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) and can be removed from TSs. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 3.8.10, during movement of fuel assemblies in the fuel storage pool, is acceptable.

3.7.15 Section 3.9, Refueling Operations

Section 3.9 of the SONGS Units 2 and 3 TSs, "Refueling Operations," contains the LCOs, Actions, and SRs related to refueling operations. This section contains the following LCOs:

TS 3.9.1, "Boron Concentration," places limits on the boron concentrations of the RCS and the refueling canal to ensure that the reactor remains subcritical during refueling. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes, which have direct access to the reactor core during refueling. The boron concentration limits required by TS LCO 3.9.1 are specified in the COLR. The boron concentration limit specified in the COLR will maintain a $k_{eff}$ of < 0.95 during fuel handling operations with CEAs and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures. TS 3.9.1 is applicable in MODE 6.

TS 3.9.2, "Nuclear Instrumentation," requires that two source range monitors (SRMs) to be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. The SRMs are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution event or an improperly loaded fuel assembly. TS 3.9.2 is applicable in MODE 6.

TS 3.9.3, "Containment Penetrations," specifies the requirements for containment closure during the conduct of CORE ALTERATIONS and movement of fuel assemblies within containment. The containment penetrations included within TS 3.9.3 are the equipment hatch, personnel airlock doors, and penetrations that provide direct access from the containment atmosphere to the outside atmosphere. TS 3.9.3 limits the consequences of an FHA involving handling fuel within containment by limiting the potential escape paths for fission product radioactivity released within containment. TS 3.9.3 is applicable during CORE ALTERATIONS and during the movement of fuel assemblies within containment.

TS 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," specifies requirements for the SDC system in MODE 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. One loop of the SDC system is required to be OPERABLE and in operation in MODE 6, with the water level > 20 feet above the top of the reactor vessel flange. Only one SDC loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. TS 3.9.4 is applicable in MODE 6, with the water level > 20 feet above the top of the reactor vessel flange.

TS 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level," also specifies requirements for the SDC system in MODE 6 to remove decay heat and sensible heat from the RCS, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. However, with the water level < 20 feet above the top of the reactor vessel flange, both SDC loops must be OPERABLE. Additionally, one loop of SDC must be in operation. TS 3.9.5 is applicable in MODE 6 with the water level < 20 feet above the top of the reactor vessel flange.

TS 3.9.6, "Refueling Water Level," specifies a minimum water level of 23 feet above the top of the reactor vessel flange during movement of fuel assemblies or CEAs within the reactor pressure vessel, and during movement of fuel assemblies within containment. A minimum refueling cavity water level of 23 feet above the top of the reactor vessel flange is required to ensure that the radiological consequences of a postulated FHA inside containment are within acceptable limits. The requirements of TS LCO 3.9.6, in conjunction with a minimum decay time of 72 hours prior to fuel movement, ensures that the release of fission product radioactivity, subsequent to an FHA, results in doses that are well within the guideline values specified in Regulatory Guide 1.183. TS 3.9.6 is only applicable during movement of fuel assemblies or CEAs within the reactor pressure vessel, and during movement of fuel assemblies within containment.

The licensee proposed to delete Section 3.9 of the SONGS Units 2 and 3 TSs LCOs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of SONGS Units 2 and 3.

The NRC staff has reviewed the proposed changes and has determined that Section 3.9 TSs are only needed to provide the LCOs and SRs necessary to maintain functionality of plant systems required for refueling operations. These TSs involve: boron concentration, which places limits on the boron concentrations of the RCS and the fuel transfer canal during refueling; nuclear instrumentation, which monitors the core reactivity condition during refueling operations; containment penetrations, which specifies requirements for containment closure during the conduct of refueling operations; residual heat removal and coolant circulation – high and low water level, which removes decay heat and sensible heat from the RCS, provides mixing of borated coolant, and prevents boron stratification; and refueling cavity water level, which specifies a minimum water level of 23 feet above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel. Therefore, refueling operations are no longer permitted at SONGS Units 2 and 3, and the LCOs in Section 3.9 TSs are no longer relevant.

The NRC staff has also reviewed the refueling operations TSs proposed for deletion to ensure that these LCOs were no longer required to satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.1 of this evaluation. The NRC staff notes that these TSs indicate MODES for which each TS is applicable. MODES, as defined in TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to MODES for a permanently shutdown and defueled reactor has no meaning and is not relevant. Because SCE has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and SONGS Units 2 and 3 are no longer in a configuration or a condition under which the TS MODES apply.

Based on the above, the NRC staff finds that the licensee's proposed change to delete TS Section 3.9, Refueling Operations, is acceptable.

3.7.16 Section 4.0, Design Features

TS 4.1, "Site" provides a description regarding the location of SONGS. The licensee has proposed to retain this TS section in the permanently defueled SONGS Units 2 and 3 TSs with no changes.

TS 4.2, "Reactor Core," provides a general description of the number of and design material requirements for the fuel and control element assemblies used in the reactor core. The licensee has proposed to delete the design feature descriptions for fuel and control element assemblies, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of SONGS Units 2 and 3.

The NRC staff has reviewed the proposed changes to delete the reactor core fuel and control element assemblies design features from SONGS Units 2 and 3 TSs. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels, the design features related to the reactor core fuel assemblies and control rods are no longer relevant at SONGS Units 2 and 3. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 4.2, reactor core design features, is acceptable.

TS 4.3, "Fuel Storage," provides a description and the requirements regarding prevention of criticality of spent fuel, prevention of SFP drainage and spent fuel capacity limitations. This TS section is being retained in the permanently defueled TSs, with the exception of TS 4.3.1.2, which is the design and maintenance of the new fuel storage racks as discussed below. The licensee has also made editorial changes to the TS references in this section to conform to the proposed renumbering of certain retained TSs.

TS 4.3.1.2 has been proposed to be deleted because new fuel is no longer stored onsite and License Condition 2.B.(3) is being revised to no longer allow receipt of new fuel. The NRC staff has reviewed the proposed changes to remove the new fuel storage rack design features from the TSs. Since the licensee currently has no new fuel stored onsite and since the facility license will no longer allow new fuel to be stored onsite, the requirements for new fuel storage racks are no longer applicable.

Based on the above, the NRC staff finds the proposed changes to delete the new fuel storage rack design features from SONGS Units 2 and 3 TS 4.3.1.2 to be acceptable. The staff also reviewed the proposed renumbering of references in TS 4.3.1, Criticality, and determined that the changes to be conforming and editorial in nature. Therefore, the NRC staff finds that the licensee's proposed changes to TS 4.3, Fuel Storage, is acceptable.

3.7.17 Section 5.2, Organization and Section 5.3, Facility Staff Qualifications

SONGS Units 2 and 3 permanently defueled TS 5.1, "Responsibility"; TS 5.2, "Organization"; and TS 5.3, "Facility Staff Qualifications," were previously approved by the NRC staff in License Amendment Nos. 227 and 220 for SONGS Units 2 and 3, respectively, dated September 30, 2014 (ADAMS Accession No. ML14183B240).

The licensee has proposed several additional changes to TS 5.2, Organization, and TS 5.3, Facility Staff Qualification that were not included in Amendment Nos. 227 and 220. The first change is to capitalize the position of CERTIFIED FUEL HANDLER, consistent with its use as a defined term in TS 1.0, Definitions.

The NRC staff reviewed the proposed change to capitalize the position of CERTIFIED FUEL HANDLER where it is used in TS 5.2 and TS 5.3 and concludes the change is editorial in nature such that the current intent of the affected TS requirements is unchanged. Therefore, the staff finds that the licensee's proposed change to capitalize CERTIFIED FUEL HANDLER in TS 5.2 and TS 5.3, is acceptable.

The licensee has also proposed a change to Facility Staff TS 5.2.2.c (note that this was originally TS 5.2.2.b but was renumbered to TS 5.2.2.c by Amendment Nos. 227 and 220), to clarify that during unexpected absences of on-duty shift crew members, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted. Specifically;

Facility Staff TS 5.2.2.c currently states:

> c. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

Revised Facility Staff TS 5.2.2.c would state:

> c. Shift crew composition may be less than the minimum requirement of Table 5.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. During such absences, no fuel movement or movement of heavy loads over storage racks containing fuel is permitted.

The NRC staff has reviewed the proposed revision to TS 5.2.2.c restricting fuel movement or movement of heavy loads over storage racks containing fuel when an unexpected absence of the on-duty shift crew results in a minimum crew composition less than specified in TSs. The staff finds that additional restriction on fuel movement and heavy loads prudent considering the reduced staffing levels at a permanently shutdown and defueled reactor facility. Therefore, the NRC staff finds that the proposed licensee change to TS 5.2.2.c, is acceptable.

The licensee has also proposed a change to Facility Staff Qualifications TS 5.3.1 to delete the qualification requirements for multi-discipline supervisors. SCE states that it will no longer be utilizing the position of multi-discipline supervisor. Specifically;

Facility Staff Qualifications TS 5.3.1 currently states:

> 5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and b) multi-discipline supervisors who shall meet or exceed the qualifications listed below.
>
> a. Education: Minimum of a high school diploma or equivalent.
>
> b. Experience: Minimum of four years of related technical experience which shall include three years power plant experience of which one year is at a nuclear plant.
>
> c. Training: Complete the multi-discipline supervisor training program.

Revised Facility Staff Qualifications TS 5.3.1 would state:

> 5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except a) the radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

The NRC Staff has reviewed the proposed deletion of the qualifications for multi-discipline supervisors from the SONGS Units 2 and 3 TSs and concluded that the qualifications are not necessary since the licensee no longer utilizes multi-discipline supervisors. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.3.1, is acceptable.

3.7.17.1 Section 5.4, Technical Specification (TS) Bases Control

SONGS Units 2 and 3, TS 5.4, "Technical Specifications (TS) Bases Control," is a program that provides the requirements for changing the TS Bases without prior NRC approval. TS 5.4 will remain applicable with the reactor permanently shutdown and defueled. As such, it is being retained and revised, as follows, to reflect a permanently defueled condition.

Currently, the licensee is required to submit changes to the TS Bases to the NRC, which have been implemented without prior NRC approval, within 6 months following every Unit 3 refueling, not to exceed 24 months. The licensee has proposed to revise TS 5.4.4 to be consistent with the submittal of UFSAR updates for the permanently shutdown and defueled status of SONGS Units 2 and 3. The TS Bases changes (that do not require NRC approval) will be submitted to the NRC for information and/or review every 24 months consistent with the UFSAR updates.

The NRC staff has reviewed the proposed change to TS 5.4.4 that aligns the submittal of changes to the TS Bases to every 24 months consistent with the submittal of the UFSAR changes. The NRC staff has determined that the proposed revision to the frequency of submitting the TS Bases Control changes to NRC is administrative in nature. The revised TS

5.4.4 continues to meet the minimum frequency of the original TS. In addition, the change is consistent with the requirements of 10 CFR 50.71(e) for providing UFSAR updates to the NRC for a permanently shutdown and defueled reactor (i.e., every 24 months). Therefore, the NRC staff finds that the licensee's proposed change to TS 5.4.4, is acceptable.

3.7.17.2 Section 5.5.1, Procedures

TS 5.5.1, "Procedures," addresses procedures, programs and manuals required by the SONGS Units 2 and 3 TSs. The licensee proposes to delete that following procedures from the permanently defueled technical specifications:

TS 5.5.1.1, "Scope," requires that written procedures be established, implemented, and maintained covering certain activities. The licensee has proposed to delete TS 5.5.1.1, paragraphs b and f.

TS 5.5.1.1, paragraph b., currently states:

> The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 (ADAMS Accession No. ML051400209), and NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," January 1983 (ADAMS Accession No. ML102560009), as stated in Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Emergency Response Capabilities," dated December 17, 1982 (ADAMS Accession No. ML031080548), incorporated into one document all Three Mile Island (TMI)-related items approved for implementation by the Commission at that time. This included the use of human factored, function oriented, emergency operating procedures to improve human reliability and the ability to mitigate the consequences of a broad range of initiating events for operating reactors, and subsequent multiple failures or operator errors, without the need to diagnose specific events.

The licensee has proposed to delete the requirement of TS 5.5.1.1.b. because the emergency operating procedures discussed therein only pertain to accidents and events resulting from reactor operation. The licensee stated that the referenced procedures are no longer required for a permanently shutdown and defueled reactor.

The NRC staff reviewed the proposed deletion of TS 5.5.1.1.b. and determined that NUREG-0737, as supplemented, implemented programmatic changes to the way reactor operators are trained, instrumentation information is presented, and procedures are structured, using human factors and a function oriented approach to address operating events and accidents. These accidents, and the associated emergency operating procedures to detect, respond to, and mitigate such accidents, concerned malfunctions of the reactor and its supporting systems are not relevant to a permanently shutdown and defueled reactor, which is

no longer authorized to operate or place fuel in the reactor vessel. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.1.1.b., is acceptable.

TS 5.5.1.1.f., concerns the modification of the core protection calculator (CPC) addressable constants. Software modifications to constants, algorithms, or fuel cycle specific data shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," (CEN-39(A)-P). The licensee has proposed to delete TS 5.5.1.1.f. because the CPC is no longer required. The CPCs are one of two systems that monitor core power distribution online and derive the LHR and DNBR parameters and associated RPS trips. The TSs that rely on the CPC are TS 3.3.1 RPS Instrumentation - Operating and TS 3.3.3 Control Element Assembly Calculators, and are only applicable in MODES 1 and 2.

The NRC staff has determined that the instrumentation-related TS 3.3.1 and TS 3.3.3 that reference the CPC, as discussed in the Section 3.7.9 of this SE, are no longer required based on the permanent shutdown and defueled condition of SONGS Units 2 and 3. The CPC is part of the RPS to protect the reactor core from damage. Since SONGS Units 2 and 3 are not authorized to operate or emplace fuel in the reactor vessel, protection of the reactor core is no longer relevant, and a control procedure for the modification of the CPC, as required in TS 5.5.1.1.f., is unnecessary. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.1.1.f., is acceptable.

3.7.17.3 Section 5.5.2, Programs and Manuals

TS 5.5.2.4, "Component Cyclic or Transient Limit Program," controls to track cyclic and transient occurrences to ensure that RCS components are monitored for fatigue evaluation based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients and accident conditions. The licensee proposes to delete this program since the RCS components monitored by this program are no longer used at SONGS Units 2 and 3 considering its permanently shutdown and defueled status.

The NRC staff has determined that deletion of the Component Cyclic or Transient Limit Program from TSs is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, the RCS and reactor support systems are no longer in use. Consequently, the component cyclic or transient limit program is not relevant at SONGS Units 2 and 3 since the components monitored by the program are permanently out of service. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.4, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.5, "Reactor Coolant Pump Flywheel Inspection Program," provides for the inspection of the reactor coolant pump flywheels. The licensee proposed to delete this program since the reactor coolant pump flywheel is a component only used in support of reactor operation. Inspection of the reactor coolant pump flywheel is not relevant to SONGS Units 2 and 3 since the licensee is no longer authorized to operate the reactor or emplace fuel in the reactor vessel.

The NRC staff has determined that deletion of the Reactor Coolant Pump Flywheel Inspection Program from TSs is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, reactor coolant pumps are no longer used in support of any function at the facility. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.5, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.6, "Secondary Water Chemistry Program," provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The licensee proposed to delete this program because the components that the program was established to protect, using water chemistry control, are associated with reactor operation. With the licensee's decision to cease reactor operations, these components are no longer in operation and do not need protection from degradation or stress corrosion cracking.

The NRC staff has determined that the deletion of the Secondary Water Chemistry Program is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, the SGs and turbine are no longer used in support of any function at the facility. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.6, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.7, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," provides controls for potentially explosive gas mixtures in the gaseous radwaste system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The licensee has proposed to revise the explosive gas and storage tank radioactivity monitoring program to be consistent with the permanently shutdown and defueled condition of the SONGS Units 2 and 3 facility. Paragraphs a. and b. of the program are being deleted because these portions of the explosive gas and storage tank radioactivity monitoring program pertain only to reactor support systems that are no longer needed due to SONGS permanently shutdown and defueled condition. Specifically, there will no longer be any source of explosive or radioactive gases generated from reactor operation. In addition, the licensee states that the gaseous radwaste system and the waste gas decay tank have been vented and removed from service. As such, references to potentially explosive gas mixtures and methods for determining gaseous radioactivity have been deleted. The licensee has proposed to retain the storage tank radioactivity monitoring program as modified below:

TS 5.5.2.7 Storage Tank Radioactivity Monitoring Program

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The liquid radwaste quantities shall be determined in accordance with the

> Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."
>
> The program shall include a surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and do not have tank overflows and surrounding area drains connected to the Liquid Waste Management System is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.
>
> The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program surveillance frequencies.

The NRC staff has reviewed the proposed revision to the Storage Tank Radioactivity Monitoring Program. The staff finds the proposed changes prudent given the uncertainty in how future radwaste generated by flushing and cutting of radioactive systems will be stored and processed. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.5.2.7, Storage Tank Radioactivity Monitoring Program, is acceptable.

TS 5.5.2.8, "Primary Coolant Sources Outside Containment Program," was established to minimize leakage from portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The licensee proposed to delete this program since primary coolant systems have been drained at SONGS Units 2 and 3 and there are no longer any transient or accident conditions associated with primary coolant sources given the permanently shutdown and defueled condition of the plant.

The NRC staff has determined that deletion of TS 5.5.2.8, "Primary Coolant Sources Outside Containment Program," is consistent with the transition to a permanently shutdown and defueled facility. Since the licensee has certified its permanent cessation of operations and defueling in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the reactors or placing fuel in the reactor vessels. Consequently, there are no DBAs involving reactor operation or refueling and there can no longer be any transients or accidents involving primary coolant outside of containment. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.8, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.9, "Pre-Stressed Concrete Containment Tendon Surveillance Program," provides controls for monitoring any tendon degradation in the pre-stressed concrete containment. The licensee has proposed to delete this program because the status of the containment is not relevant to the permanently shutdown and defueled reactors at SONGS Units 2 and 3.

The NRC staff considers that TS 5.5.2.9, "Pre-Stressed Concrete Containment Tendon Surveillance Program," is only applicable to a reactor authorized to operate or retain irradiated fuel in the reactor vessel. Pursuant to the licensee's certifications under 10 CFR 50.82(a)(2), the license is prohibited from operating the reactors or placing fuel in the reactor vessels at SONGS Units 2 and 3. Consequently, there are no DBAs involving reactor operation or refueling and no reliance on the containment to mitigate operating reactor DBAs. Thus, the staff has determined that containment tendon surveillance program TS is no longer applicable. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.9, is acceptable.

TS 5.5.2.10, "Inservice Inspection and Testing Program," establishes the controls for periodic inspection and testing of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves in accordance with the ASME Operation and Maintenance Code. These code classes protect equipment relied upon to prevent and mitigate DBAs. The licensee proposed to delete this program since there is no longer any ASME Code Class 1, 2 or 3 pumps and valves, or Code Class CC or MC components in the SONGS Units 2 and 3 inservice inspection and testing program that continue to operate and perform a specific function in mitigating the consequences of a reactor accident due to the permanently shutdown and defueled status of the plants.

Because the licensee is prohibited from operating the plant or placing fuel in the reactor vessel, in accordance with 10 CFR 50.82(a)(2), there are no longer any ASME Code class pumps and valves that remain in operation and are to be relied upon to mitigate a DBA. As such, the inservice inspection and testing program is no longer relevant to SONGS Units 2 and 3, given the permanently shutdown and defueled status of these facilities. The NRC staff also notes that the licensee shall continue to monitor the performance and condition of all SSCs associated with the storage, control, or maintenance of spent fuel in in a safe condition and with reasonable assurance that these SSCs are capable of fulfilling their intended functions, pursuant to 10 CFR 50.65, ""Requirements for monitoring the effectiveness of maintenance at nuclear power plants." Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.10, Inservice Inspection and Testing Program, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.11, "Steam Generator (SG) Program," ensures that the SG tube integrity is maintained. The licensee proposed to delete this program since SONGS Units 2 and 3 are permanently defueled and not authorized to operate; therefore, the SGs are no longer functional and the SG tubes will not be subjected to the temperature and pressure effects that the SG program was put in place to protect against.

The NRC staff has determined that the SG program is only relevant to an operating reactor where the SGs are used for removing heat associated with reactor operation. Since the licensee has certified its permanent cessation of operations and defueling in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the reactors or placing fuel in the reactor vessels. Consequently, the SGs are no longer used in support of any function at the facility. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.11, appropriately reflects the change in plant status, and is acceptable.

TS 5.5.2.12, "Ventilation Filter Testing Program (VFTP)," establishes the required testing and frequency of the CREACUS high efficiency particulate filters and charcoal adsorbers utilized by the system.

The VFTP is being deleted because it pertains only to reactor support systems that does not apply in a permanently defueled condition. As noted, in part, by the licensee in its license amendment request, dated March 21, 2014, "[t]he accident analysis applicable to the permanently defueled condition does not rely on ventilation filters for accident mitigation."

The NRC staff has determined that reference to the VFTP only appears in SONGS Units 2 and 3 TSs in three places: TS 5.5.2.12; TS 3.7.11 "Control Room Emergency Air Cleanup System (CREACUS)" (SR 3.7.11.2 and SR 3.7.11.4); and TS 5.5.2.16.d of the "Control Room Envelope Habitability Program." The VFTP is used to confirm the function and operability of the CREACUS. The NRC staff has evaluated CREACUS in Section 3.7.13 (TS 3.7.11) and found that CREACUS is no longer required in the SONGS TSs per Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C). Since TS 5.5.2.12 "Ventilation Filter Testing Program" only exists to support the SRs of TS 3.7.11 (i.e. SR 3.7.11.2 and SR 3.7.11.4, respectively) and since the NRC approves that deletion of TS 3.7.11, the NRC staff finds the licensee's proposed change to delete TS 5.5.2.12, is acceptable.

TS 5.5.2.13, "Diesel Fuel Oil Testing Program," pertains to the testing of both new and stored fuel oil used to supply the EDGs. The accident analyses applicable to the permanently shutdown and defueled condition at SONGS no longer rely on EDGs for accident mitigation. The requirement for EDGs, which are supported by the fuel oil being tested per this program, has been proposed for deletion from the TSs.

The NRC staff has reviewed the proposed changes against the requirements in 10 CFR 50.36 and Chapter 15 of the SONGS UFSAR and concluded that the EDG fuel oil and lube oil system are not required. These support systems to the EDGs are not required because there are no active systems or associated support systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the FHA DBA. The staff confirmed that there are no other DBAs that rely on EDGs or the EDG support systems. In addition, the NRC staff approves the deletion of TS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," in Section 3.7.14 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.13, the diesel fuel oil testing program, is acceptable.

TS 5.5.2.15, "Containment Leakage Rate Testing Program," is being proposed for deletion because the containment leakage rate testing program pertains only to verifying the operability of the containment systems. The need for containment or the associated required TSs does not apply in a permanently shutdown and defueled condition. The requirements for containment systems (i.e. TS 3.6.1, TS 3.6.2, TS 3.6.3, TS 3.6.4, TS 3.6.5, TS 3.6.6.1, TS 3.6.6.2, TS 3.6.8 and TS 3.9.3) are being deleted, as described in Section 3.7.12 of this SE.

Primary containment integrity and isolation are only required for post-accident conditions from power operations. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant

or placing fuel in the reactor vessel. Therefore TS 3.6.1, TS 3.6.2, TS 3.6.3, TS 3.6.4, TS 3.6.5, TS 3.6.6.1, TS 3.6.6.2, TS 3.6.8 and TS 3.9.3, which address primary containment integrity and isolation during power operations and refueling operations, are no longer applicable. The program specified TS 5.5.2.15 requires the implementation of containment leakage rate testing in accordance with 10 CFR Part 50 Appendix J, Option B, "Performance-Based Requirements." The TS 5.5.2.15 program is no longer needed since 10 CFR 50.54(o) excludes permanently defueled units from the requirements of 10 CFR Part 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.15, Containment Leakage Rate Testing Program, is acceptable.

TS 5.5.2.16, "Control Room Envelope Habitability Program," ensures that adequate radiation protection is provided to permit access and occupancy of the CRE under DBA conditions without personnel receiving radiations exposures above limits. The licensee has proposed this program for deletion because the CRE is not required for providing airborne radiological protection for the control room operators for the remaining DBAs at SONGS Units 2 and 3 based on the permanently shutdown and defueled status of the facility.

The NRC staff evaluated the remaining accident analyses at SONGS Units 2 and 3 and confirmed that no ESF system is credited in the mitigation of the CR, EAB, or LPZ dose consequences, as detailed in Sections 3.2 through 3.6 of this SE. This includes no credit for the FHIS, the fuel handling building PACU filtration system, the CRIS and the CREACUS. The evaluation of the DBAs applicable to the permanently shutdown and defueled facility demonstrate that the dose consequences within the CRE are acceptable without relying on SSCs remaining functional for accident mitigation, including FHAs. (The one exception to this is the continued function of the passive fuel storage pool structure, which will be maintained as a TS for SONGS.)

Reference to the "Control Room Envelope Habitability Program" only appears in the current SONGS Units 2 and 3 TSs in two places: TS 5.5.2.16, "Control Room Envelope Habitability Program" and TS 3.7.11, "Control Room Emergency Air Cleanup System (CREACUS)" (SR 3.7.11.4).

The NRC staff previously determined in its evaluation of TS 3.7.11, "CREACUS," Section 3.7.13 of this SE, that CREACUS no longer satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C). Consequently, the NRC staff has approved the deletion of TS 3.7.11 for SONGS Units 2 and 3. Since the Control Room Envelope Habitability Program only exists to support a surveillance requirement of TS 3.7.11 (i.e. SR 3.7.11.4) and since TS 3.7.11 will be deleted, the NRC staff finds that the licensee's proposed change to delete TS 5.5.2.16, Control Room Envelope Habitability Program, is acceptable.

TS 5.5.2.17, "Battery Monitoring and Maintenance Program," provides controls for safety-related battery maintenance. The licensee proposes deletion of this program consistent with the deletion of the corresponding TS for DC electrical systems and associated batteries. The licensee states that the SONGS accident analyses do not rely on batteries for any accident mitigation.

The NRC staff has reviewed the proposed changes against the requirements in 10 CFR 50.36 and Chapter 15 of the SONGS UFSAR and concluded that the DC electrical distribution system batteries are not required. The support systems to the DC electrical distribution system, including the batteries, are not required because there are no active systems or associated support systems credited as part of the initial conditions of an analysis or as part of the primary success path for mitigation of the remaining DBAs at SONGS Units 2 and 3. In addition, the NRC staff has approved the deletion of TS 3.8.6, "Battery Parameters," in Section 3.7.14 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete the TS 5.5.2.17, the battery monitoring and maintenance program, is acceptable.

3.7.17.4 Section 5.6, Safety Function Determination Program (SFDP)

The SONGS Units 2 and 3, "Safety Function Determination Program (SFDP)," as detailed in TS 5.6.1, TS 5.6.2, TS 5.6.3 and TS 5.6.4, ensures that a loss of safety function is detected and appropriate actions taken. Upon failure to meet two or more LCOs at the same time, an evaluation shall be made to determine if loss of safety function exists. The program implements the requirements of LCO 3.0.6. LCO 3.0.6 directs an evaluation in accordance with the SFDP to determine if a loss of safety function exists based on the status of redundant TS safety systems and associated support systems (systems that support the functionality of the safety system) to ensure the appropriate required actions are taken to maintain overall reactor safety. There are no active SSCs at SONGS Units 2 and 3 that are required for accident mitigation with the permanent cessation of reactor operations and the permanent removal of the fuel from the reactor vessels, as discussed in the evaluation of the remaining DBAs in Sections 3.2 through 3.6 of this SE. Therefore, the requirements of the SFDP, which directs cross-train checks of multiple and redundant safety systems, no longer apply.

Based on the permanently shutdown and defueled status of SONGS Units 2 and 3, all specifications for the active systems from the defueled TSs have been proposed for deletion by this licensing action. Consequently, the SFDP is no longer meaningful. In addition, the SFDP is invoked by LCO 3.0.6, which is being deleted in its entirety, as discussed in Section 3.7.6 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete TS Section 5.6, Safety Function Determination Program, is acceptable.

3.7.17.5 Section 5.7, Reporting Requirements

TS 5.7.1.1, "Annual Reports," requires a Reactor Coolant System Specific Activity Report in accordance with TS 5.7.1.1.b. The report gathered data on reactor conditions when the I-131 or gross specific activity of the reactor coolant exceeded limits specified in TS 3.4.16. The licensee has proposed to delete SONGS Units 2 and 3, TS 5.7.1.1.b, "Reactor Coolant System Specific Activity Report," since it is not applicable to a permanently shutdown and defueled reactor.

The NRC staff has reviewed the proposed deletion of TS 5.7.1.1 concerning the Reactor Coolant Specific Activity Report. The facility RCSs have been drained and the activity of the RCS is no longer relevant to the SONGS Units 2 and 3 in their permanently shutdown and

defueled status. In addition, as noted above, TS 5.7.1.1.b only exists to analyze data related to the exceedance of limits specified in TS 3.4.16. Since RCS activity is not meaningful for SONGS and TS 3.4.16 will be deleted, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.1, is acceptable.

TS 5.7.1.2, "Annual Radiological Environmental Operating Report," covers summaries, interpretations, and analyses of trends related to the radiological environmental monitoring program, for each unit, during the previous calendar year.

The licensee has proposed to revise the TS description by replacing applicability of the report to the "facility" rather than to each 'unit." In addition, the licensee is deleting a Note indicating "a single submittal may be made for a multiple unit station." This note is no longer necessary since the SONGS facility is no longer treated as a multiunit site for the purposes of the annual radiological environmental operating report.

The NRC staff reviewed the proposed revision to TS 5.7.1.2 and concludes that changing the word "unit" to "facility" and the deletion of the multiple unit station note is a clarifying change that is editorial in nature such that the current intent of the requirement is unchanged. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.7.1.2, Annual Radiological Environmental Operating Report, is acceptable.

TS 5.7.1.3, "Radioactive Effluent Release Report," covers "...the operation of the unit during the previous calendar year...." In addition, the report shall summarize the "...effluents released from the unit," and "... radioactive waste shipped from the unit directly..." and "... radioactive waste shipped from the unit's intermediary processor..."

The licensee proposed to revise the TS description by replacing "unit" with "facility" such that the description will state "... the operation of the facility during the previous calendar year ..." and, effluents "... released from the facility" and "... radioactive waste shipped from the facility directly ..." and "... radioactive waste shipped from the facility's intermediary processor...." In addition, the licensee is deleting a Note indicating "a single submittal may be made for a multiple unit station." This note is no longer necessary since the SONGS facility is no longer treated as a multiunit site for the purposes of the radioactive effluent release report.

The NRC staff reviewed the proposed revision to TS 5.7.1.3 and concludes that changing the word "unit" to facility" and the deletion of the multiple unit station note is a clarifying change that is editorial in nature such that the current intent of the requirement is unchanged. Therefore, the NRC staff finds that the licensee's proposed change to TS 5.7.1.3, Radioactive Effluent Release Report, is acceptable.

TS 5.7.1.5, "Core Operating Limits Report (COLR)," establishes the core operating limits prior to each reload cycle. The licensee proposed to delete this program since it is prohibited from reloading fuel into the SONGS Units 2 and 3 reactor core and the safety limits established by this report no longer apply.

The NRC staff has determined that the proposed deletion of the COLR would appropriately reflect the permanently shutdown and defueled condition of the facility. The COLR only applies to reactors authorized to operate. Since the licensee is prohibited from operating the SONGS reactors or placing fuel in the reactor vessels, the COLR is no longer necessary. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.5, is acceptable.

TS 5.7.1.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," documents the pressure and temperature limits for heatup, cooldown, heatup and cooldown rates, low temperature operation, criticality, and hydrostatic testing as referenced in the following TSs:

- TS 3.4.3 RCS Pressure and Temperature (P/T) Limits
- TS 3.4.6 RCS Loops – Mode 4
- TS 3.4.7 RCS Loops – Mode 5, Loops Filled
- TS 3.4.12.1 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature≤PTLR Limit
- TS 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System, RCS Temperature>PTLR Limit

The licensee proposes to delete this program since the RCS is no longer used at SONGS Units 2 and 3 in its permanently shutdown and defueled status.

The NRC staff has determined that deletion of the Reactor Coolant System Pressure and Temperature Limits Report from TSs is consistent with the transition to a permanently shutdown and defueled facility. Since, in accordance with 10 CFR 50.82(a)(2), the licensee is prohibited from operating the reactors or placing fuel in the reactor vessels, the RCS and reactor support systems are no longer in use. Consequently, the RCS PTLR is not relevant at SONGS Units 2 and 3 since the RCS is no longer functional. The staff notes that the change is consistent with the deletion of the Section 3.4 RCS TSs that reference the PTLR as discussed in Section 3.7.10 of this SE. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.6, appropriately reflects the change in the SONGS plant status, and is acceptable.

TS 5.7.1.7, "Hazardous Cargo Traffic Report," requires that SCE monitors the hazardous cargo traffic on Interstate Highway 5 and the railroad line near SONGS and submits the results to the NRC Regional Administrator once every 3 years. This reporting requirement addressed potential changes in use characteristics of these transportation routes over the life of the facility. In the enclosure to the license amendment request dated March 21, 2014, SCE proposed to delete this reporting requirement from the TSs. In the supplement dated February 23, 2015, SCE stated that it would continue to perform the hazardous traffic report in accordance with a licensee-controlled documents.

The requirements of 10 CFR 50.36(c)(5) state that Administrative Controls TSs should include reporting necessary to assure operation of the facility in a safe manner. The reporting requirements included in Section 5.6 of NUREG-1432, "Standard Technical Specifications – Combustion Engineering Plants," Volume 1 (ADAMS Accession No. ML12102A165), include only those reports specified in the LCOs and those required by regulation. The Hazardous

Cargo Traffic Report does not directly relate to operation of the facility in a safe manner. Rather, it helps identify changes in the site environs that should be periodically assessed to ensure that the scope of events considered in the design-basis remains adequate. Consequently, the report does not significantly contribute to assuring operation in a safe manner. Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.1.7, Hazardous Cargo Traffic Report, and implement a similar reporting requirement in a licensee controlled document, is acceptable.

TS 5.7.2, "Special Reports," provides a description and requirements regarding reports related to inspections, tests, and maintenance activities as directed in other SONGS TSs. The listed Special Reports pertain to 1) a pre-planned alternate method of monitoring post-accident instrumentation functions, 2) abnormal degradation of the containment structure detected during tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program, and 3) a report, following entry into MODE 4, concerning inspections performed in accordance with the SG program. The licensee states that these reports are being deleted because they do not apply in a permanently defueled condition.

The NRC staff concludes that the TS required special report information on inspections, tests, and maintenance activities for safety-related instrumentation, containment, and SGs, apply to SSCs that are no longer relevant at a permanently shutdown and defueled SONGS reactors. In addition, the NRC has approved the deletion of the associated TSs for the SSC that are subject to these special reports from the SONGS Unit 2 and 3 permanently defueled TSs. Specifically:

(1) the special report for a pre-planned alternate method of monitoring post-accident instrumentation functions is no longer necessary since the post-accident monitoring instrumentation in TS 3.3.11 is being deleted from the SONGS defueled TSs, as discussed in Section 3.7.9 of this SE.

(2) the special report on abnormal degradation of the containment structure detected during tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program is no longer necessary since the tendon surveillance program in TS 5.5.2.9 is being deleted from the SONGS defueled TSs, as discussed in Section 3.7.17.3 of this SE.

(3) the special report, following entry into MODE 4, concerning inspections performed in accordance with the SG program is no longer necessary since the SG program in TS 5.5.2.11 is being deleted from the SONGS defueled TSs, as discussed in Section 3.7.17.3 of this SE.

Therefore, the NRC staff finds that the licensee's proposed change to delete TS 5.7.2, Special Reports, is acceptable.

3.8 Changes to Facility Operating License

In SCE's March 21, 2014, license amendment request, as supplemented by a letters dated February 25, 2015, and March 18, 2015, the licensee proposed to remove, modify, and add, several facility operating license conditions, based on the permanently shutdown and defueled status of SONGS Units 2 and 3.

3.8.1 Changes to License Condition 2.B.(2)

Currently License Condition 2.B.(2), for SONGS Units 2 and 3, reads:

> (2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license.

The licensee is proposing to strike reference in the license condition to "...operate..." the facility.

The revised License Condition 2.B.(2) will read, as follows:

> (2) Southern California Edison Company (SCE), pursuant to Section 103 of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess and use the facility at the designated location in San Diego County, California, in accordance with the procedures and limitations set forth in this license.

Pursuant to 10 CFR 50.82(a)(2), as a result of the 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii) certifications submitted by the licensee, the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors. As such, reference to operation of the facility in License Condition 2.B.(2) is inconsistent with the limitation imposed on the licensee by 10 CFR 50.82(a)(2). Therefore, the NRC staff finds the licensee's proposed change to License Condition 2.B.(2) provides consistency with 10 CFR 50.82(a)(2) and, is acceptable.

3.8.2 Changes to License Condition 2.B.(3)

Currently License Condition 2.B.(3), for SONGS Units 2 and 3, reads:

> (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;

The licensee is proposing to revise this license condition to read, as follows:

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

The licensee states the proposed revision to this license condition is consistent with the restrictions of 10 CFR 50.82(a)(2) that no longer authorizes operation or emplacement of fuel in the reactor vessels at SONGS Units 2 and 3.

The proposed change removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel and eliminates the reference to use of the SNM for reactor operations. The proposed change also limits the possession of SNM pursuant to the license condition as being "that was used" as reactor fuel. Pursuant to 10 CFR 50.82(a)(2) the 10 CFR Part 50 licenses for SONGS Units 2 and 3 no longer authorize operation of the reactors. As such, the licensee has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as the licensee currently possesses the reactor fuel that was used for the past operations of the reactor. Based on the above, the NRC staff finds the licensee's proposed change to License Condition 2.B.(3) is consistent with the permanently shutdown status of SONGS Units 2 and 3 and is, therefore, acceptable.

3.8.3 Changes to License Condition 2.B.(4)

Currently License Condition 2.B.(4), for SONGS Units 2 and 3, reads:

> (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;

The licensee is proposing to revise this license condition to read, as follows:

> (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 sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required; and possess any byproduct, source and special material as sealed neutron sources that was used for reactor startup;

The licensee states the proposed revision to this license condition is consistent with the restrictions of 10 CFR 50.82(a)(2) that no longer authorizes operation or emplacement of fuel in the reactor vessels at SONGS Units 2 and 3. The proposed changes remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup but retains authorization to possess such sources previously used for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that SONGS Units 2 and 3 are no longer authorized to operate and the continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. As such, the NRC staff finds that the licensee's proposed change to License Condition 2.B.(4), is consistent with the permanently shutdown status of the facilities and is, therefore, acceptable.

3.8.4 Changes to License Condition 2.C.(1)

Current License Condition 2.C.(1), for SONGS Units 2 and 3, reads:

Maximum Power Level

(1) 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).

The licensee is proposing to delete this license condition, which will read:

(1) Deleted

The licensee states that this license condition can be deleted because SONGS Units 2 and 3 are permanently shut down and defueled in accordance with 10 CFR 50.82(a)(2) and therefore power operation is no longer authorized.

The NRC staff has reviewed the proposed deletion of License Condition 2.C.(1) and determined that power operation is no longer authorized at SONGS Units 2 and 3 based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The licensee is not authorized to operate the SONGS Units 2 and 3 at any power. Therefore, the NRC staff finds the licensee's proposed change to delete License Condition 2.C.(1) is appropriate and, is acceptable.

3.8.5 Changes to License Condition 2.C.(14) [Unit 2] and License Condition 2.C.(12) [Unit 3]

Current License Condition 2.C.(14) for SONGS Units 2 and License Condition 2.C.(12) for SONGS Unit 3, read:

Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5; SE dated November 15, 1982; Revision 1 to Updated Fire Hazards Analysis Evaluation dated June 29, 1988)

SCE shall implement and maintain in effect all provisions of the approved fire protection program. This program shall be (1) as described in the Updated Fire Hazards Analysis through Revision 3 as revised by letters to the NRC dated May 31, July 22. and November 20, 1987 and January 21, February 22, and April 21, 1988; and (2) as approved in the NRC staff's Safety Evaluation Report (SER) (NUREG-0712) dated February 1981; Supplements 4 and 5 to the SER, dated January 1982 and February 1982, respectively; and the safety evaluation dated November 15, 1982; as supplemented and amended by the Updated Fire Hazards Analysis Evaluation for San Onofre 2 and 3, Revision 1 dated June 29, 1988. SCE may make changes to the approved fire protection program without prior approval of the Commission

only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The licensee is proposing to delete License Condition 2.C.(14) for SONGS Unit 2 and delete License Condition 2.C.(12) for SONGS Unit 3, which will read:

Unit 2

(14) Deleted

Unit 3

(12) Deleted

The licensee states that this license condition is based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, "Fire protection," with the ability to achieve and maintain safe shutdown of the reactor in the event of a fire and is no longer applicable at SONGS Units 2 and 3. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during plant decommissioning. During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard. However, the regulation is applicable regardless of whether a requirement for a fire protection program is included in the facility license. Therefore, a license condition requiring such a program for a permanently shutdown and defueled plant is not needed.

The NRC staff finds that License Conditions 2.C.(14) and 2.C.(12), "Fire Protection," for SONGS Units 2 and 3, respectively, are based on maintaining fire protection programs that provides reasonable assurance that the ability to achieve and maintain safe shutdown in the event of a fire in accordance with 10 CFR 50.48. Achieving and maintaining safe shutdown in the event of a fire is no longer applicable to the decommissioned fire protection programs at SONGS Units 2 and 3, since units are permanently shutdown and the fuel has been removed from the reactors. However, elements of the fire protection program continue during decommissioning to address fire events that could result in radiological hazards. The regulation in 10 CFR 50.48(f) requires SONGS Units 2 and 3 to address the potential for fires, which could result in a radiological hazard. The licensee has proposed that the rule is sufficient to ensure that a program is maintained and therefore having a license condition that also requires fire protection programs for the permanently shutdown and defueled units is redundant. Basis on the evaluation above, the NRC staff concludes that reliance on 10 CFR 50.48(f) is appropriate and the fire protection license condition is no longer necessary. Therefore, the NRC staff finds that the licensee's proposed change to delete License Condition 2.C.(14) for SONGS Units 2, and License Condition 2.C.(12) for SONGS Unit 3, is acceptable.

3.8.6 Changes to License Condition 2.C.(27) [Unit 2] and License Condition 2.C.(28) [Unit 3]

Current License Condition 2.C.(27) for SONGS Unit 2, and License Condition 2.C.(28) for SONGS Unit 3, read:

> Upon implementation of Amendment No. 214 [Unit 2 and Amendment No. 206, Unit 3] adopting TSTF 448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.11.4 in accordance with TS 5.5.2.16.c(i), the assessment of CRE habitability as required by Specification 5.5.2.16.c(ii), and the measurement of CRE pressure as required by Specification 5.5.2.16.d, shall be considered met. Following implementation:
>
> (a) The first performance of SR 3.7.11.4, in accordance with Specification 5.5.2.16.c(i) shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.
>
> (b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.2.16.c(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from May 18, 2004, the date of the most recent successful tracer gas test, as stated in the September 17, 2004, letter response to Generic Letter 2003-01, or within the next 9 month if the time period since the most recent successful tracer gas is greater than 3 years.
>
> (c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.2.16.d, shall be within 6 months.

The licensee is proposing to delete License Condition 2.C.(27) for SONGS Unit 2 and License Condition 2.C.(28) for SONGS Unit 3, which will read:

Unit 2

(27) Deleted

Unit 3

(28) Deleted

The NRC staff evaluated the remaining accident analyses at SONGS Units 2 and 3 and confirmed that no ESF system is used to mitigate the CR, EAB, or LPZ dose consequences, as

detailed in Sections 3.2 through 3.6 of this SE. This includes no credit for the FHIS, the fuel handling building PACU filtration system, the CRIS and the CREACUS. Since SONGS Units 2 and 3 are permanently shut down and defueled, and greater than 17 months of decay time has elapsed since permanent shut down, the remaining DBAs applicable to the facility demonstrate that the dose consequences within the CRE are acceptable without relying on SSCs remaining functional for accident mitigation, with the exception of the passive fuel storage pool structure. In addition, the staff has determined that related CREACUS TS 3.7.11, and the Control Room Envelope Habitability Program TS 5.5.2.16, are no longer needed, as discussed in Sections 3.7.13 and 3.7.17.3, respectively, of this SE. Based on the discussion above, the NRC staff finds that the licensee's proposed change to delete of SONGS Unit 2 License Condition 2.C.(27), and SONGS Unit 3 License Condition 2.C.(28), is acceptable.

3.8.7 New License Condition 2.C.(28) [Unit 2] and License Condition 2.C.(29) [Unit 3]

By letter dated February 25, 2015 (ADAMS Accession No. ML15058A033), the licensee responded to an RAI from the NRC staff regarding the actions that will be taken by SCE to provide reasonable assurance that the passive, long-lived structures and components in the SFP, the fire protection system, and the radiation protection system, will be maintained in a safe condition beyond the normal licensed operating period of 40 years, pursuant to the provisions of 10 CFR 50.51(b). The NRC staff asked the licensee to identify and list the long-live, passive structures and components. In addition, the staff requested a summary of actions that will be taken to monitor and maintain the long-lived, passive structures and components. One of the staff's concerns involved the aging of neutron absorbing materials used for criticality control in SFPs.

SCE responded to the specific concern on the use of neutron absorbing materials in the SFP racks at SONGS. SCE noted that the SONGS SFP racks do contain Boraflex, a neutron-absorbing material. However, no credit is taken in SONGS accident analyses or licensing basis for the existence of the Boraflex. In addition, the NRC previously evaluated and approved borated stainless steel rods that may be placed in fuel assembly guide tubes (GTs) for reactivity control. This feature has not been implemented. If implemented in the future, SONGS will institute a surveillance program where, at 5-year intervals, 1 percent of the GT-Inserts will be inspected for any material degradation. The allowance for GT-Inserts and the commitment to the associated inspection program are described in Section 2.3.3.1.2.4.2 of the SE for Amendment Nos. 213 and 205 for SONGS Units 2 and 3, respectively (ADAMS Accession No. ML072550175).

The licensee stated that its current plans are to have all the spent fuel currently stored in the SFPs transferred to the dry cask storage ISFSI before the operating license for either SONGS Units 2 or 3 expires. However, SCE stated it will develop a list of long-lived, passive structures and components if unforeseen circumstances threaten to extend the period of fuel storage in the SFP beyond the current licensed period. SCE will develop the list and an associated aging-management program for those components if all of the spent fuel has not been removed from the SFP by February 16, 2021.

The expiration date of the Unit 2 operating license (that is, the end of the initial 40-year period of operation) is February 16, 2022. The expiration date of the Unit 3 operating license is November 15, 2022. All spent fuel onsite is expected to be moved to the ISFSI approximately 3 years prior to the expiration of the initial 40-year period of operation for both Units 2 and 3. Therefore, for the Units 2 and 3 SFPs, there is no anticipated need for long-lived, passive structures and components beyond the 40-year period of operation for Units 2 and 3, nor is there an anticipated need to monitor or maintain such structures and components beyond the licensed 40-year period of operation. Should the transition of fuel to the ISFSI be delayed by unforeseen events, it is possible that spent fuel could remain in the SFPs beyond the expiration of the 40-year operating period. Therefore, SCE proposed new license conditions for SONGS Units 2 and 3.

New License Condition 2.C.(28) for SONGS Unit 2, and License Condition 2.C.(29) for SONGS Unit 3, will read:

Unit 2

(28) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 2 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 2 until such time that all spent fuel has been removed from the Unit 2 spent fuel pool.

Unit 3

(29) Prior to February 16, 2021, if all spent fuel has not been removed from the Unit 3 spent fuel pool, an aging-management program shall be submitted for NRC approval. The scope of the program shall include those long-lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the spent fuel pool. Once approved, the program shall be described in the Updated Final Safety Analysis Report and shall remain in effect for Unit 3 until such time that all spent fuel has been removed from the Unit 3 spent fuel pool.

The NRC staff has evaluated the licensee's proposed response to the maintenance of long-lived passive structures and components considering the following applicable NRC regulations:

The regulation in 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 62, "Prevention of criticality in fuel storage and handling," requires the prevention of criticality by physical systems or processes, preferably by use of geometrically safe configurations.

The regulations in 10 CFR 50.51(b) require licensees that have provided certifications for permanent cessation of power operations and permanent removal of fuel in accordance with 10 CFR 50.82(a)(1)(i) and 10 CFR 50.82(a)(1)(ii) to take actions necessary to decommission and decontaminate the facility and continue to maintain the facility in a safe condition.

The regulations in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," require licensees to monitor performance or condition of SSCs to ensure they are capable of fulfilling their intended function. The scope of the monitoring specified in 10 CFR 50.65(a)(1) applies to safety-related SSCs as stated in 10 CFR 50.65(b)(1) and to nonsafety-related SSCs whose failure could prevent safety-related SSCs from fulfilling their intended function as stated in 10 CFR 50.65(b)(2)(ii).

The regulations in 10 CFR 50.68 specify requirements for the prevention of criticality accidents and mitigating the radiological consequences of a criticality accident.

The licensee has proposed aging management related license conditions for both SONGS Units 2 and 3, contingent that all remaining fuel will be removed from the SFP by February 16, 2021. If by this time the fuel is not removed from the SFP, the license condition will require that the licensee submit an aging management program for NRC approval. The scope of the program shall include those long lived, passive structures and components that are needed to provide reasonable assurance of the safe condition of the spent fuel in the SFP. Once approved, the program shall be described in the UFSAR and shall remain in effect until such time that all spent fuel has been removed from the SFP. The NRC staff notes that the proposed changes do not affect the design or use of the existing fuel racks, and therefore no criticality analysis was made in association with the changes. The proposed changes also keep intact the systems for the SFP needed to keep the fuel in a subcritical condition. The staff has reviewed the licensee's response to the staff's aging-management concerns and the proposed license conditions to address the concerns. Given that the licensee expects to have all fuel removed from the SONGS SFPs prior to the expiration of the original operating license, the NRC staff has concluded that the proposed new License Condition 2.C.(28) for SONGS Unit 2 and License Condition 2.C.(29) for SONGS Unit 3, adequately address the staff's concerns regarding the maintenance of passive, long-lived structures and components in a safe condition beyond the normal licensed operating period of 40 years, and therefore, finds that the new license conditions are acceptable.

3.8.8 Deletion of License Condition 2.J and Proposed New License Condition 3

Current License Condition 2.J, reads:

Unit 2

J. This license is effective as of the date of issuance and shall expire at midnight on February 16, 2022.

Unit 3

J. This license is effective as of the date of issuance and shall expire at midnight on November 15, 2022.

Revised License Condition 2.J would state for SONGS Units 2 and 3, will read:

J. Deleted

SCE stated that this license condition can be deleted because SONGS Units 2 and 3 have permanently ceased operation. 10 CFR 50.82(a)(2) prohibits operation of the SONGS Units 2 and 3 reactor since the certifications described therein have been docketed. SCE has proposed that this license condition be replaced by new License Condition 3, which conforms to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of SONGS Units 2 and 3 until the Commission notifies the licensee in writing that the license is terminated. The proposed new license condition for SONGS Units 2 and 3, to be used in place of License Condition 2.J., will be License Condition 3.

New License Condition 3 for SONGS Unit 2, will read:

3 On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 2 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On July 22, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 2 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

New License Condition 3 for SONGS Units 3, will read:

3. On June 12, 2013, Southern California Edison (SCE) certified that operations at San Onofre Nuclear Generating Station Unit 3 would permanently cease in accordance with 10 CFR 50.82(a)(1)(i). On June 28, 2013, SCE certified that the fuel had been permanently removed from the reactor vessel in accordance with 10 CFR 50.82(a)(1)(ii). As a result, the 10 CFR 50 license no longer authorizes operation of the reactor, or the emplacement or retention of fuel in the reactor vessel.

This license is effective as of the date of issuance and authorizes ownership and possession of San Onofre Nuclear Generating Station Unit 3 until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

A. Take actions necessary to decommission the plant and continue to maintain the facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

B. Conduct activities in accordance with all other restrictions applicable to the facility in accordance with the NRC regulations and the applicable provisions of the 10 CFR 50 facility license as defined in Section 2 of this license.

The NRC staff has reviewed the proposed deletion of Licensee Condition 2.J and the proposed new License Condition 3 and determined that License Condition 2.J, which documented the date of the expiration of the license, is no longer meaningful for the permanently shutdown condition of the plant in the process of decommissioning. The proposed new License Condition 3 documents the current condition of the plant and summarizes the actions and requirements applicable to the facility by regulation. The proposed License Condition 3 is consistent with the regulatory requirements applicable to the facility in the permanently shutdown and defueled condition, and consistent with a previously issued license conditions for the permanently shutdown and defueled Millstone Unit 1 and the Kewaunee Power Station. Based on the above, the NRC staff finds that the proposed license condition changes are acceptable.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified on May 28, 2015, of the proposed issuance of the amendments. 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 and changes surveillance requirements. 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 as published in the *Federal Register* on September 16, 2014 (79 FR 55513). The amendments also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 6.0 CONCLUSION

The NRC staff 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) there is reasonable assurance that 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.

Principal Contributors: M. Chernoff
D. Nold
S. Jones
D. Spaulding-Yeoman
T. Martinez-Navedo
D. Cunanan
E. Dickson
J. Hickman

Date: July 17, 2015

T. Palmisano

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

Sincerely,

***/RA/***

Thomas J. Wengert, Senior Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 230 to NPF-10
2. Amendment No. 223 to NPF-15
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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**ADAMS Accession No.: ML15139A390** * see previous **concurrence via memo

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| DATE | 5/21/15 | 7/13/15 | 02/13/15 | 4/2/15 | 9/30/14 |
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