July 13, 2006

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

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS RE: REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (TAC NOS. MC5773 AND MC5774)

Dear Mr. Rosenblum:

The U.S. Nuclear Regulatory Commission (the NRC staff or the Commission) has issued the enclosed Amendment No. 203 to Facility Operating License No. NPF-10 and Amendment No. 195 to Facility Operating License No. NPF-15 for San Onofre Nuclear Generating Station, Units 2 and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 28, 2005, as supplemented by letter dated January 12, 2006.

By letter dated January 28, 2005, you requested revision of TSs 1.1, “Definitions,” 3.4, “Reactor Coolant System,” and 5.7, “Reporting Requirements,” and to relocate the RCS pressure temperature (P-T) curves and limits from the TSs to a licensee-controlled document identified as the Pressure-Temperature Limit Report. In the supplement dated January 12, 2006, in addition to providing the responses to a request for additional information from the NRC staff, you also requested the use of an alternate methodology for calculating the stress intensity factor K_{IM} due to internal pressure loading. Consistent with by the safety evaluation (SE) on Combustion Engineering (CE) Topical Report NPSD-683-A, Revision 6, dated March 16, 2001, you included a request for an exemption from the requirements of 10 CFR Part 50, Appendix G, for P-T limits since the alternate methodology applies the CE Nuclear Steam Supply System method for calculating K_{IM} stress intensity values.

The NRC staff authorized the exemption by letter dated June 5, 2006, and it was published in the *Federal Register* on July 6, 2006 (71 FR 38430).

We are providing the SE to you at this time, for review, to assure that any information that Westinghouse Electric Company claims to be proprietary would not be inadvertently released to the general public. We will delay placing the SE in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the SE will be made publicly available. Our conclusions concerning the proprietary nature of your January 12, 2006, submittal, are contained in a letter to Mr. B. F. Maurer, Westinghouse Electric Company dated February 15, 2006.

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

Sincerely,

/RA/

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

Docket Nos. 50-361 and 50-362

Enclosures:
1. Amendment No. 203 to NPF-10
2. Amendment No. 195 to NPF-15
3. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

/RA/

N. Kalyanam, Project Manager
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cc w/encls: See next page

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*Editorial Changes to staff provided SE.

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DSS/SPWB*	DCI/CVIB*	OGC nlo w/comment	NRR/LPL4/BC
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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 203
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 January 28, 2005, as supplemented by letter dated January 12, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

/RA/

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

Attachment: Changes to the Technical Specifications and Page 3 of the Facility Operating License No. NPF-10

Date of Issuance: July 13, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the 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.

REMOVE	INSERT
iii	iii
1.1-5	1.1-5
----	1.1-5a
3.4-5	3.4-5
3.4-6	3.4-6
3.4-7	3.4-7
3.4-8	3.4-8
3.4-9	3.4-9
3.4-10	3.4-10
3.4-11	3.4-11
3.4-12	3.4-12
3.4-18	3.4-18
3.4-22	3.4-22
3.4-30	3.4-30
3.4-31	3.4-31
3.4-33	3.4-33
3.4-35	3.4-35
5.0-27	5.0-27
--------	5.0-27a

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 195
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 January 28, 2005, as supplemented by letter dated January 12, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

/RA/

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

Attachment: Changes to the Technical Specifications and Page 3 of the Facility Operating License No. NPF-15

Date of Issuance: July 13, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the 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.

REMOVE	INSERT
iii	iii
1.1-5	1.1-5
----	1.1-5a
3.4-5	3.4-5
3.4-6	3.4-6
3.4-7	3.4-7
3.4-8	3.4-8
3.4-9	3.4-9
3.4-10	3.4-10
3.4-11	3.4-11
3.4-12	3.4-12
3.4-18	3.4-18
3.4-22	3.4-22
3.4-30	3.4-30
3.4-31	3.4-31
3.4-33	3.4-33
3.4-35	3.4-35
5.0-27	5.0-27
--------	5.0-27a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By application dated January 28, 2005 (Reference 1), as supplemented by letter dated January 18, 2006 (Reference 2), Southern California Edison Company (SCE, the licensee) requested changes to the Technical Specifications (TSs) for San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3). Reference 2 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 1, 2005 (70 FR 9996).

The license amendment request proposes to relocate the pressure-temperature (P-T) limits and low temperature overpressure protection (LTOP) system limits from the TS limiting conditions for operation (LCOs) for SONGS 2 and 3 into unit-specific pressure temperature limit reports (PTLRs) that will be administratively controlled by TS Section 5.7.1.6 for each unit. The PTLRs for SONGS 2 and 3 are documented in Topical Report (TR) WCAP-16005-NP, Revision (Rev.) 3, “San Onofre Nuclear Generating Station Unit 2 RCS [Reactor Coolant System] Pressure and Temperature Limits Report,” and WCAP-16167-NP, Rev. 0, “San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report,” respectively. These reports were provided as enclosures 4 and 5 in Reference 1.

The licensee provided supplemental information to support its license amendment request by Reference 2, including a request for an exemption from the requirements of Appendix G to Part 50 of Title 10, *Code of Federal Regulations* (10 CFR Part 50, Appendix G) in order to utilize certain provisions of the PTLR methodology, approved by the Nuclear Regulatory Commission (NRC), of Combustion Engineering Owners Group (CEOG) TR CE NPSD-683-A,

Revision 6, “Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications,” (Reference 3). This proposed change will allow Southern California Edison to make periodic updates to the RCS P-T curves, Heatup and Cooldown curves, and LTOP enable temperatures more efficiently, without the requirement for a TS change amendment request to be submitted to the NRC for approval.

2.0 REGULATORY EVALUATION

2.1 Part 50 of 10 CFR Requirements for Generating P-T Limits and LTOP System Limits for Pressurized-Water Reactors (PWRs)

The NRC established requirements in 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Part 50 of 10 CFR, Appendix G, requires that the P-T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated using the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). For conditions with the core critical, P-T limits must be more conservative than the ASME Code, Section XI, Appendix G limits. Table 1 of 10 CFR Part 50, Appendix G, provides a summary of the requirements for P-T limits relative to the ASME Code, Section XI, Appendix G criteria, as well as the minimum temperature requirements for bolting up the reactor vessel (RV) during normal and pressure testing operations. Part 50 of 10 CFR, Appendix G, also requires that applicable surveillance data from RV material surveillance programs be incorporated into the calculations of plant-specific P-T limits and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the RCPB. The rule also establishes conservative requirements for determining the temperature and pressure setpoints for LTOP systems. P-T limits and LTOP system limits are subject to General Design Criteria (GDC) 14, “Reactor coolant pressure boundary"; GDC 15, “Reactor coolant system design"; GDC 30, “Quality of reactor coolant pressure boundary"; and GDC 31, “Fracture prevention of reactor coolant pressure boundary,” in 10 CFR Part 50, Appendix A.

Part 50 of 10 CFR, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” provides the staff’s criteria for the design and implementation of RV material surveillance programs for operating light-water reactors. The staff’s requirements for protecting the RVs of PWRs against pressurized thermal shock (PTS) events are given in 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events.”

Staff regulatory guidance related to determining the change in RV material parameters and P-T limit curves due to the effects of radiation embrittlement is found in Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials.” Staff guidance related to the review of P-T limit curves and PWR PTS criteria is found in Standard Review Plan (SRP) Section 5.3.2, Pressure-Temperature Limits and Pressurized Thermal shock.” Staff guidance related to the review of LTOP system limits is found in SRP Section 5.2.2, “Overpressure Protection.”

The regulatory requirements for RV fluence calculations are specified in GDC 14, 30, and 31 of 10 CFR Part 50, Appendix A. In March 2001, the staff issued RG 1.190, “Calculational and

Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Fluence calculations are acceptable if they are done with approved methodologies or with methods that are shown to conform to the guidance in RG 1.190.

2.2 Technical Specification Requirements for P-T Limits and LTOP System Limits

Section 182a of the Atomic Energy Act of 1954 (Title 42 USC Section 2232) requires applicants for nuclear power plant operating licenses to include TS as part of the operating license. The Commission's regulatory requirements related to the content of the TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions of operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

Paragraph 50.36(c)(2)(ii) of 10 CFR requires that LCOs be established for the P-T limits and LTOP system limits because the parameters fall within the scope of Criterion 2 identified in the rule:

> Criterion 2: 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 P-T limits and LTOP system limits for PWR-designed light-water reactors fall within the scope of Criterion 2 stated above, and are, therefore, ordinarily required to be included within the TS LCOs for a plant-specific facility operating license (FOL).

On January 31, 1996, the staff issued Generic Letter (GL) 96-03, "Relocation of the Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," to inform licensees that they may request a license amendment to relocate the actual P-T limit curves and/or LTOP system limit values from the TS LCOs into a PTLR or other licensee-controlled document that would be administratively controlled through the administrative controls section of the TS. In order to permit relocation of the P-T limits and LTOP system limits, GL 96-03 indicated that licensees seeking to relocate P-T limits and LTOP system limits for their reactors would need to generate their P-T limits and LTOP system limits in accordance with an NRC-approved methodology and that the methodology used to generate the P-T limits and LTOP system limits would need to comply with the requirements of 10 CFR Part 50, Appendices G and H. Furthermore, the methodology used to generate the P-T limits and LTOP system limits would need to be incorporated by reference in the administrative controls section of the TS. The GL also mandated that the TS administrative controls section for the PTLR would need to reference the staff's safety evaluation (SE) issued on the PTLR methodology and that the PTLR be defined in Section 1.0 of the TS. Attachment 1 to GL 96-03 provided a list of the criteria that the approved PTLR methodology and plant-specific PTLR license amendment application would be required to meet.

Technical Specification Task Force (TSTF) Traveler No. TSTF-419 amended the Standard Technical Specifications (STS) (NUREGs-1430, -1431, -1432, -1433, and -1434) to: (1) delete references to the TS LCO specifications for the P-T limits and LTOP system limits in the TS definition of the PTLR, and (2) revised STS 5.6.6 to identify, by number and title,

NRC-approved TRs that document PTLR methodologies, or the NRC SE for a plant-specific methodology by NRC letter and date. A requirement was added to the reviewers note to specify the complete citation of the PTLR methodology in the plant-specific PTLR, including the report number, title, revision, date, and any supplements. Only the figures, values, and parameters associated with the P-T limits and LTOP system limits are relocated to the PTLR. The methodology for their development must be reviewed and approved by the NRC. TSTF-419 did not change the requirements associated with the review and approval of the methodology or the requirement to operate within the limits specified in the PTLR. Any changes to a methodology that had not been approved by the staff would continue to require staff review and approval pursuant to the license amendment request provisions and requirements of 10 CFR 50.90, "Application for amendment of license or construction permit."

3.0 TECHNICAL EVALUATION

Part 50 of 10 CFR, Appendix G, requires licensees to establish limits on the allowable pressure and temperature in order to protect the RCPB against brittle failure. These limits are defined by P-T limit curves for normal operations (including reactor heatup and cooldown operations, operations with the reactor critical, and transient operating conditions) and during pressure testing conditions (i.e., either inservice leak rate testing and/or hydrostatic testing conditions). For PWRs, the LTOP system limits ensure that the pressure remains below the applicable P-T limits.

3.1 Evaluation of the Proposed Changes to the SONGS 2 and 3 TSs

In Reference 1, the licensee proposed TS requirements related to the implementation of the PTLRs for SONGS 2 and 3. The proposed TS requirements discussed herein address the proposed TS definition of the PTLRs, the proposed TS LCOs, related applicability criteria, and the proposed administrative controls governing the PTLR content and reporting requirements. Proposed LCO action statements and surveillance requirements are not discussed here. All of the proposed TS pages related to the implementation of the PTLRs for SONGS 2 and 3, including proposed LCO action statements and surveillance requirements, can be found in Reference 1. The proposed TS requirements related to the implementation of the PTLRs for SONGS 2 and 3, including the definition of the PTLR, TS LCOs, related applicability criteria, and administrative controls, are as follows:

A. The TS definition for the PTLR, as given in Section 1.1 of the TSs for SONGS 2 and 3:

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.

B. The LCOs for the SONGS 2 and 3 P-T limits and LTOP system limits, as given in Sections 3.4.3, 3.4.6, and 3.4.12, respectively, of the TSs for SONGS 2 and 3:

LCO 3.4.3 The combination of RCS [Reactor Coolant System] pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits as specified in the RCS PRESSURE-TEMPERATURE LIMITS REPORT (PTLR).

APPLICABILITY: At all times

LCO 3.4.6 Two loops or trains consisting of any combination of RCS loops and shutdown cooling (SDC) trains shall be OPERABLE and at least one loop or train shall be in operation.

--NOTES--

1. All reactor coolant pumps (RCPs) and SDC pumps may be de-energized for ≤ 1 hour per 8 hour period, provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM [shutdown margin] of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless:

   a. Pressurizer water volume is < 900 ft^3 [cubic feet], or

   b. Secondary side water temperature in each steam generator (SG) is < 100 °F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

LCO 3.4.7 [Similar to LCO 3.4.6, the actual LCO remains unchanged, and a PTLR reference is added to NOTE 4.]

--NOTES--

[Notes 1, 2, 3, 5, and 6 remain unchanged]

4. No reactor coolant pump (RCP) shall be started with one or more of the RCS cold leg temperatures ≤ the temperature in the PTLR unless:

   a. The pressurizer water volume is < 900 ft^3 or

b. The secondary side water temperature in each steam generator (SG) is < 100°F above each of the RCS cold leg temperatures.

3.4.12.1 Low Temperature Overpressure Protection (LTOP) System
RCS Temperature ≤ PTLR Limit

LCO 3.4.12.1 No more than two high pressure safety injection pumps shall be OPERABLE, the safety injection tanks shall be isolated or depressurized to less than the limit specified in the PTLR and at least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

1) A lift setting of 406 ± 10 psig [pounds per square inch],

2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377, and 2HV9378 open,

or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to the enable temperatures specified in the PTLR,

MODE 5, and

MODE 6 when the head is on the reactor vessel and the RCS is not vented.

------------------------------------NOTE------------------------------------
SIT [safety injection tank] isolation or depressurization to less than the limits in the PTLR is only required when SIT pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the PTLR.

3.4.12.2 Low Temperature Overpressure Protection (LTOP) System
RCS Temperature > PTLR Limit

LCO 3.4.12.2 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

1) A lift setting of 406 ± 10 psig,

2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377, and 2HV9378 open,

or,

b. A minimum of one pressurizer code safety valve with a lift setting of 2500 psia ± 1%.

APPLICABILITY: MODE 4 when the temperature of all RCS cold legs are greater than the enable temperatures specified in the PTLR.

----------------------------------NOTES---

1. The lift setting pressure of the pressurizer code safety valve shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

2. The SDCS [shutdown cooling system] Relief Valve lift setting assumes valve temperatures less than or equal to 130 °F.

C. The administrative control section of the SONGS 2 and 3 TSs for the PTLRs, as given in Section 5.7.1.6 of the TSs:

5.7.1.6 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Technical Specification 3.4.3 RCS Pressure and Temperature (P/T) Limits,

Technical Specification 3.4.6 RCS Loops - MODE 4,

Technical Specification 3.4.7 RCS Loops - MODE 5, Loops Filled,

Technical Specification 3.4.12.1 Low Temperature Overpressure Protection (LTOP) System RCS Temperature ≤ PTLR Limit, and

Technical Specification 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System RCS Temperature > PTLR Limit.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and

approved by the NRC, specifically those described in the following document:

CE NPSD-683-A, The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

The licensee's proposed definition of the PTLR, as set forth in Section 1.1 of the SONGS 2 and 3 TSs, identifies the specification, TS 5.7.1.6, which controls the PTLR content. TS 5.7.1.6 directly references the individual specifications for which RCS P-T limits and LTOP system limits are established in the PTLRs. Each of these specifications references the PTLR, as appropriate, in the specification LCO. The proposed PTLR definition and controlling TSs meet the technical criteria of GL 96-03 and are consistent with NUREG-1432, Revision 3, "Standard Technical Specifications, Combustion Engineering Plants." Therefore, the staff concludes that the proposed definition of the PTLR, as given in TS 1.1, and the proposed changes to TS LCOs 3.4.3, 3.4.6, 3.4.12, and TS 5.7.1.6 are acceptable.

The adoption of TSTF-419 allows NRC-approved TRs to be identified by number and title in TS 5.7.1.6. This allows the licensee to use current, approved TRs to support the calculation of parameters in the PTLRs without having to submit an amendment to the FOL every time the TR is revised. The licensee's PTLR submittals provided the specific information (i.e., report number, title, revision, and date) identifying the particular approved TR which documents the methodology used to determine the P-T limits and LTOP system limits. This provides assurance that only the approved version of the referenced TRs is used for the determination of the P-T limits and LTOP system limits since the complete citation is provided in the PTLR, and the PTLR methodology documented in the TR was approved by the NRC.

3.2 Evaluation of the Proposed Methodology for the PTLRs Against the Criteria for Approved Methodologies in Attachment 1 of GL 96-03

References 1 and 2 indicate that future changes to the PTLRs for SONGS 2 and 3 will be carried out in accordance with an approved version of Reference 3, which (also referred to as the Combustion Engineering Nuclear Steam Supply System [CE NSSS] methodology) is specifically referenced in Section 5.7.1.6 of the proposed revised TS as the controlling document governing future changes to the SONGS 2 and 3 PTLRs. The licensee's PTLRs indicate that P-T limits and LTOP system limits are established in accordance with Reference 3.

The staff evaluated the methodology in Reference 3 for establishing the P-T limits and LTOP system limits for CE plants in a PTLR. This evaluation was documented in the NRC SE of March 16, 2001 (Reference 4). CEOG incorporated the key NRC recommendations identified in Reference 4 and issued Reference 3, which included the staff's SE. Thus, Reference 3 represents the latest NRC-approved version of CE NPSD-683.

In Reference 4, the staff concludes that while the contents of the report are technically acceptable, the report leaves the description of certain key methodology details up to the licensee applying for a license amendment to relocate the P-T limits and LTOP system limits into a PTLR. These details are collectively identified in Section 5.0 of the SE of Reference 4 as 26 action items that would need to be addressed in a PTLR license amendment request that invoked the methods of the TR. These 26 action items collectively serve the dual purpose of fulfilling the plant-specific PTLR criteria from Attachment 1 of GL 96-03 and covering specific PTLR methodology details that were left open to plant-specific PTLR requests. Therefore, licensees requesting a license amendment to relocate the P-T limits and LTOP system limits into a PTLR, using the methodology of Reference 3, would need to specifically address the 26 action items in their plant-specific PTLR submittals.

The methodology in Reference 3 meets the minimum technical requirements for approved methodologies specified in Attachment 1 to GL 96-03. The licensee's use of Reference 3 as the proposed methodology for the SONGS 2 and 3 PTLRs meets the criterion in GL 96-03 specifying that an approved methodology be used for the development of the PTLR. The PTLRs for SONGS 2 and 3 are, therefore, acceptable with respect to this criterion.

3.3 Evaluation of the PTLR Contents Against the Seven Criteria for PTLR Contents in Attachment 1 of GL 96-03 and the 26 Action Items from Reference 4.

Attachment 1 of GL 96-03 contains seven technical criteria (PTLR criteria) that the contents of PTLRs should conform to if license amendments requesting relocation of TS P-T limits and LTOP system limits into PTLRs are to be approved by the staff. The staff's evaluations of the contents of the SONGS 2 and 3 PTLRs against the criteria in GL 96-03 are given in the subsections that follow. The licensee, in Reference 2, provided responses to the 26 action items by either directing to the location of information in Reference 1or by providing supplemental information.

The following discussion addresses the contents of Reference 1 and Reference 2, in the context of how they address the applicable regulatory requirements, the seven PTLR criteria, and the associated action items. Disposition of all 26 original action items is documented herein. The discussion is organized according to the seven PTLR criteria.

PTLR Criterion 1

PTLR Criterion 1 states that the PTLR contents should include the neutron fluence values that are used in the calculation of the adjusted reference temperature (ART) values for the P-T limit calculations. Accurate and reliable neutron fluence values are required in order to satisfy the provisions of GDC 14, 30, and 31 of 10 CFR Part 50, Appendix A, as well as the specific fracture toughness requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.61. Furthermore, several specific aspects of the PTLR methodology for determining the fluence values must also be identified under this criterion. The PTLR methodology must describe the neutron fluence calculation methods, including computer codes and formulas used to calculate neutron fluence. These fluence methodology details and PTLR Criterion 1 provisions are collectively reflected in Reference 3, action items (1) and (2) in Reference 4, with the fulfillment of PTLR Criterion 1. These action items are as stated in References 3 and 4. However, action item 1 has changed since the issuance of Reference 4 due to updates in regulatory guidance

for calculating neutron fluence. Therefore, the following statement of the original action item is followed by an explanation of its current provisions:

(1) Describe the methodology used to calculate the neutron fluence values for the reactor vessel materials, including a description of whether or not the methodology is consistent with the guidance of Draft Regulatory Guide DG-1053, a description of the computer codes used to calculate the neutron fluence values, and a description of how the computer codes for calculating the neutron fluence values were benchmarked.

The guidance of Draft Guide-1053 is no longer used for the fluence calculation methodology. Neutron fluence values are now calculated in accordance with the guidance in RG 1.190. Therefore, this action item must now be addressed by including a description of whether or not the fluence methodology is consistent with RG 1.190.

(2) Provide the values of neutron fluence used for the [ART] calculations, including the values of neutron fluence for the inner [diameter] (ID), 1/4T, and 3/4T locations of the RV.

The RV fluence methodology and calculational results are presented in the SONGS 2 and 3 PTLRs. The methodology is identical for both units. The proposed methodology uses the discrete ordinate transport (DORT) code for the calculation of the vessel fluence. The geometry is input in a (r, θ) and (r, z) two-dimensional configuration. There are two distinguishable core loading patterns for cycles 1-10 that used two separate inputs. Pin-wise relative power in (r, θ) and (r, z) are used to simulate three-dimensional power distributions. The fission spectrum from the outer assemblies is used to determine the spectrum of the neutron source, because the two cycle patterns have distinct neutron spectra. The BUGLE-93 cross sections were used by DORT to calculate the macroscopic energy dependent cross sections for all materials used in the analysis. The (r, θ), (r, z) configuration and the BUGLE-93 cross sections are based on the ENDF/B-VI library, as recommended in RG 1.190.

Approximations used include a P_3 expansion for anisotropic scattering and S_{10} for the quadrature expansion, which exceeds the RG 1.190 guidance for the quadrature expansion. The (r, z) solution is expanded 35 cm above and below the active core height. In the (r, θ) space an eighth core geometry was used. The flux synthesis used the same process described in the RG 1.190. All phases of the fluence calculations are in agreement with the guidance in RG 1.190.

The first 10 cycles in SONGS 2 were monitored with the 7° azimuthal-angle surveillance capsule. The capsule analysis included fluence calculations to the end of cycle 10, corresponding to 13.28 effective full power years (EFPY) of facility operation. Calculations also included 32 EFPY (i.e., to the end of current license period). Finally, peak RV fluence values were estimated by interpolation for 20 EFPY. The dosimeter measurement uncertainties are consistent with the Framatome data base uncertainties.

For SONGS 2, the peak neutron fluence value for 32 EFPY at the vessel-wetted surface is 4.37×10^{19} n/cm^2 (E > 1.0 MeV), corresponding to the lower shell plates. This corresponds to a fluence value of 4.147×10^{19} n/cm^2 (E > 1.0 MeV) at clad/base metal interface,

2.472 x 10^{19} n/cm^2 (E > 1.0 MeV) at the 1/4T location in the vessel wall, where T equals the vessel wall thickness, and 0.878 x 10^{19} n/cm^2 (E > 1.0 MeV) at the 3/4T vessel wall location.

For SONGS 3, the peak neutron fluence value for 32 EFPY at the vessel-wetted surface is 4.19x10^{19} n/cm^2 (E > 1.0 MeV), corresponding to the intermediate shell plates. This corresponds to a fluence value of 3.976 x 10^{19} n/cm^2 (E > 1.0 MeV) at clad/base metal interface, 2.37 x 10^{19} n/cm^2 (E > 1.0 MeV) at the 1/4T location in the vessel wall, and 0.842 x 10^{19} n/cm^2 (E > 1.0 MeV) at the 3/4T vessel wall location.

Based on the above discussion of the information contained in References 1 and 2, the staff concludes that the fluence methodology and calculations adhere to the guidance in RG 1.190 and that the licensee has satisfied the provisions of PTLR Criterion 1 from Attachment 1 of GL 96-03 and associated action items (1) and (2). The fluence values cited by the licensee are, therefore, acceptable.

PTLR Criterion 2

Part 50 of 10 CFR, Appendix H, provides the staff's requirements for designing and implementing RV materials surveillance programs. The rule requires that RV material surveillance programs for operating reactors must meet the requirements of American Society for Testing and Materials (ASTM) Standard Procedure E-185, "Standard Practice for Conducting Surveillance Tests for Light Water Power Reactor Vessels." The rule requires that the program design and the surveillance capsule withdrawal schedule for the program must meet the edition of the ASTM E-185 that is current on the issue date of the ASME Code to which the RV was purchased, although the rule permits more recent versions of ASTM E-185 to be used, up through the 1982 Edition (ASTM E-185-82).

To ensure conformance with these requirements, PTLR Criterion 2 states that the PTLR should either provide the RV surveillance capsule withdrawal schedule or provide references, by title and number, for the documents containing the RV surveillance capsule withdrawal schedule. The criterion also states that the PTLR should reference, by title and number, any applicable surveillance capsule reports that have been placed on the docket by the licensee requesting approval of the PTLR for its units. These specific PTLR Criterion 2 provisions are reflected in two action items associated with the fulfillment of PTLR Criterion 2, which are identified as action items (3) and (4) in Section 5.0 of Reference 4:

(3) Either provides the surveillance capsule withdrawal schedule in the proposed PTLR for the amendment, or reference in the PTLR, by title and number, the documents in which the withdrawal schedule is located.

The licensee provided surveillance capsule withdrawal schedules in Table 2-2 of Section 2.0 of the SONGS 2 and 3 PTLRs. The licensee also referenced in Section 2.0 of the SONGS 2 and 3 PTLRs, the documents that describe the RV surveillance program and surveillance capsule withdrawal schedules for SONGS 2 and 3. The RV surveillance programs for SONGS 2 and 3 are described in References 5, 6, and 7. This meets the provision in PTLR Criterion 2 stating that the PTLR must provide the RV material surveillance capsule withdrawal schedule, or reference by title and number documents in which the schedule is located. The applicant has based the RV surveillance capsule withdrawal schedules for SONGS 2 and 3 on the applicable withdrawal schedule requirements of ASTM E-185-73. The staff confirmed that

the surveillance program and the current withdrawal status for the RV surveillance capsules for SONGS 2 and 3 are in compliance with the RV surveillance program requirements and capsule withdrawal schedule requirements of ASTM E-185-73 and 10 CFR Part 50, Appendix H. Based on this assessment, the staff determined that the SONGS 2 and 3 PTLRs meet the first provision of PTLR Criterion 2 and associated action item (3).

> (4) Reference the surveillance capsule reports by title and number if the ART values are calculated using [RV] surveillance data.

The licensee referenced the RV surveillance capsule evaluation reports that have been docketed for SONGS 2 and 3 in Section 2.0 of the SONGS 2 and 3 PTLRs. The surveillance capsule evaluation reports for SONGS 2 are provided in References 8, 9, and 10. The surveillance capsule evaluation reports for SONGS 3 are provided in References 11, 12, and 13. Based on this assessment, the staff determined that the SONGS 2 and 3 PTLRs meet the second provision of PTLR Criterion 2 and associated action item (4).

Based on the above discussion, the staff concludes that the licensee has satisfied the requirements of 10 CFR Part 50, Appendix H, and meets the provisions of PTLR Criterion 2 from Attachment 1 of GL 96-03 and associated action items (3) and (4).

PTLR Criterion 3

The LTOP system protects the RCPB by ensuring that the RCS system pressure remains below the applicable P-T limits required by 10 CFR Part 50, Appendix G. PTLR Criterion 3 in GL 96-03 states that the PTLR must provide LTOP system limit curves or setpoint values for PWRs. Furthermore, several specific aspects of the PTLR methodology regarding LTOP system limits must also be identified under this criterion. The PTLR methodology must describe how the LTOP system limits are calculated, applying system thermal-hydraulics and fracture mechanics. These specific LTOP analysis methodology details and PTLR Criterion 3 provisions are reflected in eleven action items associated with the fulfillment of PTLR Criterion 3, which are identified as action items (5) through (15) in Section 5.0 of Reference 4:

> (5) Provide a description of the analytical method used in the energy addition transient analysis.

Section 3.2.1.3 of the SONGS 2 and 3 PTLRs provides a description of the analytical method used in the energy addition transient.

> (6) Provide a description of the analytical method used in the mass addition transient analysis, if different from that in Section 3.3.5 of the TR.

Section 3.2.1.2 of the SONGS 2 and 3 PTLRs provides a description of the analytical method used in the mass addition transient. (In its response to RAI 1 for this action item, the licensee indicated that Section 3.2.1.2 of the SONGS 2 and 3 PTLRs is revised to remove the statement "...and the equivalent mass addition that results from energy additions," in order to be consistent with the supporting analysis of record.)

> (7) Provide a description of the method for selection of relief valve setpoints.

Section 3.2.1.1 of the SONGS 2 and 3 PTLRs provides a method for selection of relief valve setpoints.

(8) Provide a justification for use of subcooled water conditions or a steam volume in the pressurizer.

Section 3.2.1 of the SONGS 2 and 3 PTLRs provides adequate justification for the use of subcooled water conditions or a steam volume in the pressurizer.

(9) Provide a justification for a less conservative method for determination of decay heat contribution if the method used is less conservative than the "most conservative method" described in the TR.

The analyses of record that support the SONGS 2 and 3 LTOP energy addition and mass addition transients were performed using methodologies that preceded approval of the methods described in the TR. The SONGS energy addition analysis used an assumed value of 1 percent decay heat, relevant to approximately 3.5 hours post shutdown. This is considered a conservative estimate of the cooldown time needed to achieve the LTOP enable temperature and is, therefore, conservative for the decay heat contribution. The SONGS mass addition transient does not include a decay heat contribution.

(10) Provide justification for operator action time used in transient mitigation or termination.

The transient analyses in the SONGS 2 and 3 PTLRs do not assume operator action for mitigation of the transients.

(11) Provide correlations used for developing Power Operated Relief Valve (PORV) discharge characteristics.

SONGS 2 and 3 do not have PORVs and, therefore, do not credit pressurizer PORV discharge characteristics in the LTOP transient analysis.

(12) Provide spring relief valve discharge characteristics if different from those described in the TR or if the peak transient pressure is above the set pressure of the valve plus 10 percent.

Section 3.2.1.1 of the SONGS 2 and 3 PTLRs provides spring relief valve discharge characteristics.

(13) Provide a description of how the reactor coolant temperature instrumentation uncertainty was accounted for.

The P-T limit curves for SONGS 2 and 3 account for RCS temperature instrumentation uncertainty. As discussed in the licensee’s response to RAI 6 under PTLR Criterion 5, a conservative temperature adjustment of 18.5 °F is added to the analytical temperature limits to conservatively increase the allowable temperature limits for the P-T limit values provided in the SONGS 2 and 3 PTLRs.

(14) Provide a justification for the mass and energy addition transient mitigation which credit presence of nitrogen in the pressurizer.

SONGS 2 and 3 do not credit the presence of nitrogen in the pressurizer in the LTOP transient analysis.

(15) Identify and explain any other deviation from the methodology included in Section 3.0 of the TR.

There are no other deviations from the PTLR methodology of Reference 3, with respect to this PTLR criterion.

LTOP System Limits

Proposed TS 3.4.12.1 protects the design safety assumptions for the LTOP system by requiring that no more than two high pressure safety injection (HPSI) pumps shall be operable, the safety injection tanks shall be isolated or depressurized to less than the limit specified in the PTLRs and at least one of the following overpressure protection systems shall be operable, the shutdown cooling relief valve with the relief valve isolation valves open, or the RCS depressurized with an RCS vent greater than or equal to 5.6 square inches. The PTLRs specify the LTOP enable temperature. TS 3.4.12.1 is applicable in Mode 4 when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLRs, in Mode 5, and in Mode 6 when the RCS is not vented.

The following assumptions assure a conservative estimate for the enable temperature. These assumptions are the same as those in the SONGS 2 and 3 Updated Final Safety Analysis Report (UFSAR):

- the pressurizer is water solid,
- the RCS boundary does not expand,
- the RCS metal is adiabatic,
- the RCS letdown is isolated,
- all injection pumps attain rated speed instantaneously,
- only one relief valve is used in the mitigation,
- no operator action is credited, and
- conservative energy and mass additions are assumed including full heat output from the pressurizer heaters and decay heat is increased by 10% and assumed constant through the transient.

3.4 Transient Analyses

The actual maximum pressure limits are determined from mass and heat injection transients to quantify conservative maximum pressures for low temperature.

The following assumptions assure a conservative analysis:

- the Shut Down Cooling System (SDCS) is assumed isolated at the start of the transient,
- the SDCS valve opening profile is conservative relative to the ASME model,
- no reactor coolant pump seal leakage or controlled leak-off is assumed,

- the RCS is isothermal and neither heated nor cooled by the mass addition, and
- the initial conditions are chosen to maximize the pressure transient.

LTOP, Mass Addition Transient

The method and results of the mass addition transient are those in the SONGS 2 and 3 UFSAR. The HPSI mass addition was developed using two-pump delivery with no losses, which is the maximum volumetric delivery at any pressure. The method is more conservative than the CE NSSS method described in Reference 3 and the results bound those in Reference 3, which has been approved by the staff.

LTOP Heat Addition Transient

The computer code used for the heat addition transient is OVERP and is described in WCAP-15688, Revision 00, "CE NSSS LTOP Energy Addition Transient Analysis Methodology," Westinghouse Electric Company, LLC, May 2001. In addition to the LTOP system limit assumptions stated above, for heat addition it is assumed that: (1) the steam generator secondary is 100 °F hotter than the primary coolant, and (2) one RCP starts and instantaneously reaches rated speed. Based on the results of the analysis of this transient, the NRC staff concludes that the large relief capacity of the shutdown cooling system valve reduces the pressure very fast, with the peak pressure remaining below the 10 CFR Part 50, Appendix G limit.

Based on the above discussion, the staff concludes that the licensee has satisfied the provisions of PTLR Criterion 3 from Attachment 1 of GL 96-03 and associated action items (5) through (15).

PTLR Criterion 4

Part 50 of 10 CFR, Appendix G, requires that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron embrittlement on the fracture toughness of RV beltline materials. For P-T limits, the effects of neutron embrittlement on the fracture toughness of RV beltline materials is defined in terms of the shift in the reference nil-ductility temperature (RT_{NDT}) resulting from neutron irradiation over a given period of facility operation, expressed in EFPY. The adjusted RT_{NDT} (ART) value resulting from neutron embrittlement over a certain period of facility operation is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term. NRC RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides the staff's recommended methodologies for calculating ART values used for P-T limit calculations. ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the

copper and nickel contents, the fluence, and the calculational procedures. Appendix G to Section XI of the ASME Code requires that licensees determine the ART at the 1/4T and 3/4T locations.

Section 50.61 of 10 CFR provides the regulatory requirements for implementing pressurized thermal shock (PTS) assessments for PWRs and for calculating the RT_{PTS} values required for these assessments. The rule requires RT_{PTS} values for base metal materials (i.e., RV plate or forging materials) and axial weld (longitudinal weld) materials to remain below 270 °F through the expiration of the operating license for the PWR facility. The rule requires RT_{PTS} values for circumferential weld (girth weld) materials to remain below 300 °F through the expiration of the operating license for a PWR facility. The methods in 10 CFR 50.61 for calculating the RT_{PTS} values for PWR RVs are consistent with the methods for calculating ART values in RG 1.99, Revision 2.

To ensure conformance with the requirements of 10 CFR Part 50, Appendix G, PTLR Criterion 4 states that PTLR contents should identify the limiting ART values at the 1/4T and 3/4T locations in the wall of the RV. To ensure compliance with the PTS requirements of 10 CFR 50.61, PTLR Criterion 4 also states that the PTLR contents should identify the limiting RT_{PTS} value for the RV, as required by 10 CFR 50.61. These specific PTLR Criterion 4 provisions are reflected in two action items associated with the fulfillment of PTLR Criterion 4, which are identified as action items (16) and (17) in Section 5.0 of Reference 4:

(16) Identify the limiting materials and corresponding ART values for both the quarter-thickness (1/4T) and three-quarter-thickness (3/4T) locations of the R.V. shell.

The licensee provided the calculated ART values for the limiting beltline materials in Table 4-1 of Section 4.0 of the SONGS 2 and 3 PTLRs for 32 EFPY, which represents the time of expiration for the current SONGS 2 and 3 FOLs. Fluence calculations for the wetted surface, 1/4T, and 3/4T locations, as well as fluence factor calculations, were provided in Tables 4-2 and 4-3 of Section 4.0 of the SONGS 2 and 3 PTLRs. Calculations of the 1/4T and 3/4T ART values for all SONGS 2 and 3 RV beltline materials were provided in Table 4-4 of Section 4.0 of the SONGS 2 and 3 PTLRs. This table includes, for each beltline material, calculations out to 13.28, 20, and 32 EFPY of facility operation. Each calculation includes the initial RT_{NDT} value, the ΔRT_{NDT} value, and the final ART value taking into consideration the shift in the RT_{NDT} value and margin term for each corresponding EFPY value. The table also includes the parameters used in these calculations, including the chemistry factors, fluence values, fluence factors, and margin terms.

The limiting material at the 1/4T location for the SONGS 2 RV is lower shell plate C-6404-5 and at the 3/4T location is lower shell plate C-6404-4. For the SONGS 3 RV, the limiting material at both the 1/4T and 3/4T locations is intermediate shell plate C-6802-1. The staff has independently confirmed this and is in agreement with licensee's finding.

For SONGS 2, the licensee determined that the limiting material ART values at the 1/4T and 3/4T locations are 137.3 °F and 116.6 °F, respectively, and for SONGS 3, the limiting material ART values at the 1/4T and 3/4T locations are 145.8 °F and 125.5 °F, respectively. The staff has independently confirmed this and is in agreement with licensee's finding.

The staff also confirmed that the ART calculations in Table 4-4 of Section 4.0 of the SONGS 2 and 3 PTLRs were in conformance with RG 1.99, Rev. 2, for all of the SONGS 2 and 3 beltline materials.

Based on this assessment, the staff determined that the SONGS 2 and 3 PTLRs meet the first provision of PTLR Criterion 4 and associated action item (16).

> (17) For [PWR] facilities, identify the limiting RT_{PTS} values for the [RV] as calculated in accordance with the methods and criteria of 10 CFR 50.61.

The licensee discusses the PTS assessments in Section 4.0 of the SONGS 2 and 3 PTLRs. Here, the licensee summarizes the requirements and screening criteria of 10 CFR 50.61 and stated that the limiting beltline materials for PTS in the SONGS 2 and 3 RVs are SONGS 2 lower shell plate C-6404-5 and SONGS 3 intermediate shell plate C-6802-1. The licensee determined that the RT_{PTS} values at the expiration of the operating licenses for SONGS 2 and 3 (i.e., at 32 EFPY) are 146.3 °F for Unit 2 and 154.6 °F for Unit 3. The licensee indicated that RT_{PTS} values were calculated for all of the SONGS 2 and 3 beltline materials in Table 4-4 of Section 4.0 of the SONGS 2 and 3 PTLRs. Calculations were based on a 32 EFPY peak fluence at the vessel clad/base metal interface of 4.147×10^{19} n/cm^2 (E > 1.0 MeV) in the lower shell for SONGS 2 and 3.976×10^{19} n/cm^2 (E > 1.0 MeV) in the intermediate shell for SONGS 3. The staff independently confirmed that the limiting beltline materials for PTS were SONGS 2 lower shell plate C-6404-5 and SONGS 3 intermediate shell plate C-6802-1. The staff also independently confirmed the RT_{PTS} values at the expiration of the SONGS 2 and 3 operating licenses for these limiting materials. Based on this assessment, the staff determined that the SONGS 2 and 3 PTLRs meet the second provision of PTLR Criterion 4 and associated action item (17).

Based on the above discussion, the staff concludes that the licensee has satisfied the requirements of 10 CFR Part 50, Appendix G; 10 CFR 50.61 pertaining to PTLR Criterion 4; and that the ART calculations are consistent with the methodologies of RG 1.99, Revision 2, and, therefore, meet the provisions of PTLR Criterion 4 from Attachment 1 of GL 96-03 and associated action items (16) and (17).

PTLR Criterion 5

Section IV.A.2 of 10 CFR Part 50, Appendix G, requires that the P-T limits for operating reactors and the minimum temperatures established for the highly stressed regions of RVs (i.e., for the RV flange and stud assemblies) be met for all conditions. The rule also requires that the P-T limits for operating reactors must be at least as conservative as those that would be generated if the methods of analysis in the ASME Code, Section XI, Appendix G, were used to generate the P-T limit curves. Table 1 of 10 CFR Part 50, Appendix G, provides a summary of the required criteria for generating the P-T limits for operating reactors.

To ensure that PTLRs are in conformance with the above requirements, PTLR Criterion 5 states that the PTLR contents should provide the P-T limit curves for heatup and cooldown operations, core critical operations, and pressure testing conditions for operating light-water reactors. Furthermore, several specific aspects of the PTLR methodology for generating the plant-specific P-T limit curves must also be identified under this criterion. These P-T limit methodology details and PTLR Criterion 5 provisions are collectively reflected in five action

items associated with the fulfillment of PTLR Criterion 5, which are identified as action items (18) through (22) in Reference 4. The following statements of these action items is followed by an explanation of the current applicability criteria, with respect to PTLR Criterion 5:

(18) Ensure that the ferritic RV materials that have accumulated neutron fluences in excess of 1.0×10^{17} n/cm^2 (E > 1 MeV) will be assessed according to Section 4.0 of the CE Topical Report NPSD-683, Rev. 6, regardless of whether the materials are located within the region immediately surrounding the active core.

(19) Identify which method (i.e., K_{IC} [static plane strain fracture toughness] or K_{IA} [dynamic/crack arrest fracture toughness]) will be used to calculate the reference stress intensity factor (K_{IR}) values for the RV as a function of temperature.

The information addressing action items (18) and (19) were addressed in the licensee's response to RAI 1. As indicated in the portion of the response to RAI 1 addressing action item (18), the licensee stated that, for SONGS 2 and 3, all RV materials located within the region immediately surrounding the active core were evaluated according to Section 4.0 of CE NPSD-683-A, Rev. 6, as prescribed by PTLR Criterion 4, to determine the material with the limiting ART value at the 1/4T and 3/4T locations. For the RV materials located above and below the region immediately surrounding the active core, the licensee stated that the neutron fluence is much lower than the neutron fluence to which the RV materials located within the region immediately surrounding the active core are exposed. The initial RT_{NDT} values of the RV materials located above and below the active core region is comparable to the initial RT_{NDT} values of the materials located within the active core region. Based on these facts, the licensee concluded that materials located above and below the active core region would never exhibit a limiting ART value. As indicated in the portion of the response to RAI 1 addressing action item (19), the licensee stated that K_{IC} was used to calculate the reference stress intensity factor, K_{IR}, values for the RV.

(20) If ASME Code Case N-640 and K_{IC} are being used as the basis for calculating the K_{IR} reference fracture toughness values, submit an exemption request [pursuant to the alternative program provisions of 10 CFR 50.60(b)] to use the methods of ASME Code Case N-640 and apply them to the P-T limit calculations.

This action item originally dealt with the use of ASME Code Case N-640 and K_{IC} as the basis for calculating the K_{IR} reference fracture toughness values. It is no longer applicable to the fulfillment of PTLR Criterion 5 for P-T limit curve calculations due to updates in the allowable editions and addenda of the ASME Code, Section XI, Appendix G, which have been incorporated by reference in 10 CFR Part 50. Therefore, this action item no longer needs to be addressed in the plant-specific PTLR submittal, and the submittal of a 10 CFR Part 50 exemption request to use the methods of ASME Code Case N-640 for P-T limit curve calculations is no longer required.

(21) (Applicable only if the CE NSSS methods for calculating K_{IM} [applied stress intensity, K, due to pressure loading] and K_{IT} [applied K due to thermal loading] factors, as stated in Section 5.4 of CE NPSD-683, Revision 6, are being used as

the basis for generating the P-T limits for their facilities). Apply for an exemption against requirements of Section IV.A.2. of Appendix G to Part 50 to apply the CE NSSS methods to their P-T curves. This is consistent with the *"note"* on page 5-15 of CE NPSD-683, Revision 6. Exemption requests to apply the CE NSSS to the generation of P-T limit curves should be submitted pursuant to the provision of 10 CFR 50.60(b) and will be evaluated on a case-by-case basis against the exemption request acceptance criteria of 10 CFR 50.12; and

This action item originally dealt with the use of the CE NSSS methods for calculating both the K_{IM} and K_{IT} factors, as stated in Section 5.4 of CE NPSD-683-A, Rev. 6, as the basis for generating P-T limit curves. Currently this action item only applies the use of the CE NSSS methods for calculating the K_{IM} factor as the basis for generating P-T limit curves. It is no longer applicable to the use of the CE NSSS methods for calculating the K_{IT} factor as the basis for generating P-T limit curves due to updates in the allowable editions and addenda of the ASME Code, Section XI, Appendix G, which have been incorporated by reference in 10 CFR Part 50. This action item is satisfied through the submittal of an exemption request addressing the use of the CE NSSS methods for calculating the K_{IM} factor as the basis for generating P-T limit curves.

In RAI 2, the staff asked the licensee to supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a discussion of the methodology for calculating the stress intensity factors, K_{IM} and K_{IT}. In response, the licensee indicated that the CE NSSS methods given in Reference 3 were used for calculating the K_{IM} and K_{IT} stress intensity factors. The CE NSSS methods given in Reference 3 for calculating K_{IT} are now codified in 10 CFR Part 50-endorsed editions and addenda of the ASME Code, Section XI, Appendix G. In addition, in accordance with the response to RAI 1, regarding action item (21), the licensee included the application for the exemption specified in action item (21) from the requirements of 10 CFR Part 50, Appendix G. The requested exemption would allow for the application of the CE NSSS methods of Reference 3 for calculating K_{IM} values to the calculation of P-T limits, in lieu of the methodology cited in the ASME Code, Section XI, Appendix G. This exemption request was submitted pursuant to the provisions of 10 CFR 50.60(b) and 10 CFR 50.12, and the authorization of the licensee’s exemption request is documented in an exemption from the applicable requirements of 10 CFR Part 50, Appendix G.

(22) Include in their PTLRs the P-T limit curves for heatup, cooldown, criticality, and hydrostatic and leak testing of their reactors.

The licensee provided P-T limit data for heatup and cooldown operations, core critical operations, and hydrostatic testing in Tables 5-1 through 5-3 of Section 5.0 of the SONGS 2 and 3 PTLRs. The P-T limit curves corresponding to these data points were provided in Figures 5-1 through 5-3 of Section 5.0 of the SONGS 2 and 3 PTLRs. This meets the provisions of PTLR Criterion 5 which specifies that the PTLR includes the P-T limit curves for reactor heatup, cooldown, critical operations, and pressure testing conditions. This also addresses the provisions of the associated action item (22).

In RAI 4, the staff asked the licensee to supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a discussion of how the calculation of the proposed P-T limit curves in the SONGS 2 and 3 PTLRs account for the most limiting conditions in the reactor vessel during the heat-up and cool-down transients. For this RAI, the staff specifically requested a discussion of (a) how the

calculation of the P-T limit curves addresses heat-up and cool-down transients, taking into consideration the different conditions at the 1/4T and 3/4T crack depth locations, and (b) how the calculation of the P-T limit curves addresses the assessment of the 1/4T crack depth location for steady-state conditions in addition to heat-up and cool-down conditions. In addition, the staff requested that the licensee supplement the P-T limit curves in the SONGS 2 and 3 PTLRs with a P-T limit curve representing the 1/4T location under steady-state conditions. In response to RAI 4, the licensee provided, in graphical form, the enveloping procedure used for developing P-T limits for the 1/4T and 3/4T crack depth locations for the heat-up and cool-down transients. The licensee also included the P-T limit curve representing the 1/4T location under steady-state conditions. This supplemental information ensures that the proposed P-T limit curves in the SONGS 2 and 3 PTLRs account for the most limiting conditions in the RV during all possible thermal transients. The staff determined that the above information satisfied resolution of RAI 4.

Section 5.0 of the SONGS 2 and 3 PTLRs provided a footnote indicating that the P-T limit values were adjusted for instrument uncertainty and RCS pressure and elevation effects. The P-T limit values for SONGS 2 and 3 are based on pressurizer pressure and RCS temperature, as interpreted by the pressure and temperature instrumentation. In RAI 6, the staff asked the licensee to supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a detailed discussion of how instrument uncertainties are treated in the development of the PTLR P-T limit curves for SONGS 2 and 3.

In response to RAI 6, the licensee indicated that the adjustments to the P-T limit values for instrument uncertainty ensure that the analytical beltline P-T limits are conservatively interpreted by the pressurizer pressure and RCS temperature instrumentation. The pressure values are adjusted using pressure correction factors (PCFs), and the temperature values are adjusted for temperature instrumentation uncertainty. The PCFs applied to the P-T limit values consist of three components: (1) pressure differential corresponding to water head between the pressurizer water level and the reference point in the reactor vessel, (2) flow-induced pressure drop between the reactor vessel downcomer and the surge nozzle in the hot leg, and (3) pressurizer pressure instrumentation loop uncertainty. These three components are individually established using conservative assumptions, then summed to yield the PCF. The PCF values are subtracted from the analytical pressure limits for the beltline to conservatively reduce the allowable pressure limit. The licensee provided the PCFs that are applied to the analytical pressure limits to yield the P-T limit values provided in the SONGS 2 and 3 PTLRs. To account for temperature instrumentation uncertainty, a temperature adjustment of 18.5 °F is added to the analytical temperature limits to increase the allowable temperature limits for the P-T limit values provided in the SONGS 2 and 3 PTLRs. The staff determined that the above information satisfied resolution of RAI 6.

The current P-T limit curves established in the SONGS 2 and 3 TS are effective through 20 EFPY of facility operation. The proposed P-T limit curves provided in the SONGS 2 and 3 PTLRs are effective through 32 EFPY of facility operation. In RAI 7, the staff asked the licensee to indicate whether the changes to the P-T limit curves incorporated into the proposed SONGS 2 and 3 PTLRs from the P-T limit curves incorporated into the SONGS 2 and 3 TS reflected only the increase in the EFPY for which the curves would be applied. Furthermore, the staff asked the licensee to indicate whether other factors, such as different parameters or methods, contributed to the changes to the curves. If factors other than the increase in the

EFPY did contribute to the changes to the curves, the staff asked the licensee to provide a detailed discussion of how those other factors affected the proposed P-T limit curves incorporated into the proposed SONGS 2 and 3 PTLRs.

In response to RAI 7, the licensee indicated that the changes to the P-T limit curves included in Section 5.0 of the SONGS 2 and 3 PTLRs reflect the combined effect of the increase in the EFPY for which the curves are valid and the use of the K_{IC} fracture toughness to calculate the reference stress intensity factor, K_{IR}, values for the RV, instead of the K_{IA} fracture toughness. The staff determined that the above information satisfied resolution of RAI 7.

In RAI 3, the staff asked the licensee to supplement the SONGS 2 and 3 PTLRs with data for the through-wall thermal gradients (ΔT) and K_{IT} factors for the 1/4T and 3/4T crack depth locations during thermal transients (i.e., during heatup and cooldown operations), as well as the values for K_{IM}. These data were necessary for the NRC staff to perform independent calculations of the P-T limits to verify that the proposed P-T limit curves contained in the SONGS 2 and 3 PTLRs are bounded by the requirements of 10 CFR Part 50, Appendix G, as modified by the exemption authorization to apply the methodology of Reference 3 for determining the K_{IM} values. In response to RAI 3, the licensee provided the requested data. The staff determined that this data satisfied resolution of RAI 3.

Using the information provided by the licensee in References 1 and 2, the staff evaluated the licensee's revised P-T limit curves for acceptability by performing a set of check calculations by using the methodologies referenced in the ASME Code, Section XI, Appendix G, as modified by the CE NSSS methodologies given in Reference 3. The staff's independent calculations confirmed the licensee's determination regarding how the limiting RV beltline materials contributed to the definition of the SONGS 2 and 3 RV P-T limit curves. The staff verified that the licensee's proposed P-T limit curves satisfied the requirements in Section IV.A.2 of Appendix G to 10 CFR Part 50, as modified by the exemption authorization to apply the methodology of Reference 3. Specifically, the staff concluded that the P-T limit curves submitted by the licensee appropriately accounted for the limiting conditions defined by the material properties of the limiting beltline materials and were at least as conservative as those that would be generated by the staff's application of the methodology specified in the ASME Code, Section XI, Appendix G, as modified by the CE NSSS methodologies given in Reference 3 for determining K_{IM} values.

Based on the above discussion, the staff concludes that the licensee has satisfied the requirements of 10 CFR Part 50, Appendix G, pertaining to PTLR Criterion 5, as modified by the exemption discussed above, which authorizes the use of the methodology of Reference 3 for generating the P-T limits curves in the SONGS 2 and 3 PTLRs. Furthermore, the staff concludes that the licensee's PTLR submittal, as supplemented, meets the provisions of PTLR Criterion 5 from Attachment 1 of GL 96-03 and associated action items (18) through (22), as applicable.

PTLR Criterion 6

Section IV.A.2 of 10 CFR Part 50, Appendix G, requires that the P-T limits for operating reactors and the minimum temperature requirements for the highly stressed regions of the RVs (i.e., RV flange and stud assemblies) be met for all conditions. Table 1 of 10 CFR Part 50,

Appendix G, provides the required criteria for meeting the minimum temperature requirements for the stressed regions of the RV.

PTLR Criterion 6 states that the minimum temperature requirements of 10 CFR Part 50, Appendix G, shall be incorporated into the P-T limit curves, and the PTLR shall identify minimum temperatures on the P-T limit curves such as the minimum boltup temperature and the hydrotest temperature. This PTLR criterion is reflected in one action item, which is identified as action item (23) in Section 5.0 of Reference 4.

(23) Demonstrate how the P-T curves for pressure testing conditions and normal operations with the core critical and not-critical will be in compliance with the appropriate minimum temperature requirements as given in Table 1 to Appendix G to Part 50.

The licensee provided a listing and brief description of the minimum temperature requirements that have been incorporated into the P-T limit curves for SONGS 2 and 3. The minimum temperature requirements are listed in Table 6-1 of Section 6.0 of the SONGS 2 and 3 PTLRs. However, the discussion and list of minimum temperature requirements in the PTLRs did not adequately demonstrate how the P-T limit curves for pressure testing conditions and normal operations with the core critical and with the core not critical would be in compliance with the minimum temperature requirements, as given in Table 1 of 10 CFR Part 50, Appendix G. In RAI 5, the staff asked the licensee to supplement Section 6.0 of the SONGS 2 and 3 PTLRs with a discussion concerning how the P-T limit curves would meet all of the minimum temperature requirements mandated by Table 1 of 10 CFR Part 50, Appendix G. The staff requested that the licensee include in its discussion the value for the highest reference temperature of the material in the closure flange region and how this value is applied, along with the minimum permissible hydrostatic test temperature, to determine the minimum temperature requirements being applied to the SONGS 2 and 3 P-T limit curves.

In response to RAI 5, the licensee provided a discussion of how the P-T limit curves for pressure testing conditions and normal operations with the core critical and with the core not critical would be in compliance with the minimum temperature requirements, as given in Table 1 in 10 CFR Part 50, Appendix G. This discussion included detailed calculations demonstrating how the minimum temperature requirements that are applied to the proposed P-T limit curves in the SONGS 2 and 3 PTLRs are determined by applying the highest reference temperature of the material in the closure flange region along with the minimum permissible hydrostatic test temperature, as required by Table 1 in 10 CFR Part 50, Appendix G. The staff verified that the licensee’s application of the highest flange reference temperature and the minimum permissible hydrostatic test temperature to the proposed P-T limit curves meets the minimum temperature requirements specified in Table 1 in 10 CFR Part 50, Appendix G. Based on this assessment, the staff determined that the above information satisfied resolution of RAI 5.

Based on the above discussion, the staff concludes that the licensee’s PTLR submittal, as supplemented, meets the provisions of PTLR Criterion 6 from Attachment 1 of GL 96-03 and associated action item (23).

PTLR Criterion 7

RG 1.99, Rev. 2, provides the staff's recommended methods for calculating the ART values for RV beltline materials. These ART values are calculated for the 1/4T and 3/4T locations in the vessel wall. The ASME Code, Section XI, Appendix G, and 10 CFR Part 50, Appendix G, require that these values be used for the calculations of P-T limit curves for reactors. Part 50 of 10 CFR, Appendix G, also requires that the ART values include the applicable results of the RV material surveillance program of 10 CFR Part 50, Appendix H. ART values for ferritic RV base metal and weld materials increase as a function of accumulated neutron fluence and the quantity of alloying elements in the materials, copper and nickel in particular. The procedures of the RG specify the use of a chemistry factor (CF) as a means for quantifying the effect of the alloying elements on the ART values. Furthermore, the RG specifies that a CF be calculated and input into the calculation of the final ART value for each beltline material. The RG cites one of two methods to determine the CF values for the RV beltline base metal and weld materials: (1) Regulatory Position 1.1 in the RG allows the licensee to determine the CF values from applicable tables in the RG as a function of copper and nickel content or, (2) Regulatory Position 2.1 allows the use of applicable RV surveillance data to determine the CF values if the base metal or weld materials are represented in a licensee's RV material surveillance program and if two or more credible surveillance data sets become available for the reactor in question. The criteria for determining the credibility of the RV surveillance data sets are defined in the RG. In accordance with the requirements of 10 CFR Part 50, Appendix G, the RG states that if the procedure of Regulatory Position 2.1 results in a higher ART value than that given by using the procedure of Regulatory Position 1.1, the surveillance data should be used for determining the CF and ART. If the procedure of Regulatory Position 2.1 results in a lower value for the ART, the CF and ART results of either procedure may be used.

To ensure that PTLRs are in conformance with the above regulatory requirements and guidelines, PTLR Criterion 7 states that if surveillance data are used in the calculations of the ART values, the PTLR contents should include the surveillance data and calculations of the CF values for the RV base metal and weld materials, as well as an evaluation of the credibility of the surveillance data against the credibility criteria of RG 1.99, Rev. 2. These PTLR Criterion 7 provisions are reflected in three action items associated with the fulfillment of PTLR Criterion 7, which are identified as action items (24) through (26) in Section 5.0 of Reference 4.

(24) Include in their PTLRs the supplemental surveillance data and calculations of the chemistry factors if surveillance data are used for the calculations of the adjusted reference temperatures.

(25) Provide the evaluation of whether the surveillance data are credible in accordance with the credibility criteria of RG 1.99, Rev. 2.

The SONGS 2 and 3 material surveillance programs are designed to meet the requirements of 10 CFR Part 50, Appendix H, and include the following RV materials: for SONGS 2, the RV surveillance materials are Base Metal Plate C-6404-2 and Weld 9-203 (Heat No. 90069), and for SONGS 3, the RV surveillance materials are Base Metal Plate C-6802-1 and Weld 9-203 (Heat No. 90069).

For SONGS 2, the licensee provided the supplemental surveillance data in Section 7.0 of the SONGS 2 PTLR. This data table includes the measured shift in the RT_{NDT} values for the surveillance materials, CF values for the surveillance materials, as well as an analysis of the credibility of the surveillance data. In RAI 8, the staff asked the licensee: (1) to clarify whether the credibility analysis of the SONGS 2 surveillance data demonstrated that the surveillance data were credible according to the RG 1.99, Rev. 2, criteria for determining surveillance data credibility, (2) to indicate whether the methods of Regulatory Position 1.1 of RG 1.99, Rev. 2, were used to determine CF and ART values for the limiting materials, (3) explain why this was an acceptable method for determining the CF and ART values, if the surveillance data were credible, and (4) to supplement Section 7.0 of the SONGS 2 PTLR with a description of the methods used for determining the CF values for the surveillance materials and detailed calculations of these CF values.

In response to RAI 8, the licensee confirmed that a credibility analysis was performed in accordance with RG 1.99, Rev. 2, for the SONGS 2 surveillance materials and the analysis demonstrated that all of the credibility criteria of the RG were met for the SONGS 2 surveillance base metal Plate C-6404-2 and surveillance Weld 9-203. The staff verified that the licensee's credibility analysis of the SONGS 2 surveillance data met the credibility criteria of RG 1.99, Rev. 2. The licensee also confirmed, in its response to RAI 8, that the ART values for all of the SONGS 2 RV beltline materials, including the limiting materials, were calculated in accordance with Regulatory Position 1.1 of RG 1.99, Rev. 2. The licensee explained that their use of Regulatory Position 1.1 of the RG for the limiting material ART calculations was based upon a determination that the ART values for the surveillance materials, as calculated using the procedure of Regulatory Position 2.1 in RG 1.99, Rev. 2, were less conservative than the ART value of the limiting material, determined using the procedure of Regulatory Position 1.1 in RG 1.99, Rev. 2. This conservatism was necessary because the surveillance base metal material, plate C-6404-2, was not representative of the material with the limiting ART value at the 1/4T location, lower shell plate C-6404-5.

In addition, the licensee provided a description of the methods used for determining the CF values for the surveillance materials, including detailed calculations of these CF values. The licensee indicated that surveillance material CF values from Table 7-1 of the SONGS 2 PTLR were obtained from Tables 1 and 2 of RG 1.99, Rev. 2 (Regulatory Position 1.1). The licensee included the material chemistry data (copper and nickel content) that were used to arrive at these CF values. The staff verified that these CF values were based on appropriate chemistry data. The licensee's calculations of the surveillance material CF values in Reference 2, were appropriately based on Regulatory Position 2.1 of RG 1.99, Rev. 2. The staff verified the validity of these calculations. These CF calculations demonstrate that the SONGS 2 limiting material ART values, as determined using the methods of Regulatory Position 1.1 of RG 1.99, Rev. 2, are more conservative than the surveillance material ART values.

For SONGS 3, the ART value for the limiting material was determined using surveillance data, as specified in Regulatory Position 2.1 of RG 1.99, Rev. 2. The licensee provided the supplemental surveillance data in Table 7-1 of the SONGS 3 PTLR. The credibility analysis of the surveillance data for SONGS 3 was provided in Table 7-2 of the SONGS 3 PTLR.
Based on the data provided in Section 7.0 of the SONGS 3 PTLR, the staff determined that the licensee's use and application of the available surveillance data for the determination of the CF

and ART values for the limiting material satisfied the provisions of RG 1.99, Rev. 2, PTLR Criterion 7, and the associated action items. Therefore, RAI 8 did not apply to the application of the surveillance data for the CF calculations for SONGS 3.

Regulatory Position 2.1 of RG 1.99, Rev. 2, states that if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the corresponding vessel weld, the measured value of ΔRT_{NDT} for the surveillance weld should be adjusted by multiplying it by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld. In RAI 9, the staff asked the licensee to indicate whether the copper and nickel content of the surveillance weld differs from that of the vessel weld for SONGS 2 and 3. If any such difference in copper and nickel content did exist between the surveillance weld and corresponding vessel weld for either unit, the staff requested that the licensee supplement Section 7.0 of the SONGS 2 or 3 PTLRs with detailed calculations of the adjustments to the measured values for ΔRT_{NDT} for the surveillance welds and indicate whether these adjusted ΔRT_{NDT} values were used in the determination of the CF values for the SONGS 2 and 3 surveillance welds.

In response to RAI 9, the licensee stated that for both SONGS 2 and 3 the copper and nickel content of the surveillance weld (Weld 9-203 for each unit) differs from the copper and nickel content of the as-deposited vessel Weld 9-203. The licensee provided detailed calculations of the adjustments to the measured values for ΔRT_{NDT} for the surveillance welds based on these differing weld chemistries, as prescribed by Regulatory Position 2.1 of RG 1.99, Rev. 2. The staff verified the accuracy of these calculations. The licensee then confirmed, through step-by-step calculations of the CF values for the surveillance welds, that the adjusted ΔRT_{NDT} values were used in the determination of the CF values for the SONGS 2 and 3 surveillance welds. The staff determined that the above information satisfied resolution of RAI 9.

Based on this assessment, the staff determined that the above information satisfied resolution of RAI 8 and that the SONGS 2 PTLR, as supplemented, meets the provisions of PTLR criterion 7 and associated action items (24) and (25).

(26) In addition, if licensees seek to use surveillance data from supplemental plant sources, licensees must:

(a) Identify the source(s) of the data.

(b) Either identify by title and number the SE report that approved the use of the supplement data, along with a justification of why the data is applicable; or compare the licensee's data with the data from the supplemental plants(s) for both the radiation environments (i.e., neutron spectrums and irradiation temperatures) and the surveillance test results, and pursuant to Section III.C of Appendix H to Part 50, submit the proposed integrated surveillance program and evaluation of the data to the NRC for review and approval.

Action Item (26) does not specifically apply to SONGS 2 and 3 because the licensee does not use surveillance data from supplemental plant sources in the SONGS 2 and 3 PTLR.

Based on the above discussions, the staff concludes that the licensee has satisfied the requirements of 10 CFR Part 50, Appendix G, pertaining to PTLR Criterion 7. The staff also concludes that the licensee's credibility analysis for the SONGS 2 and 3 surveillance data and calculations of the CF values for the limiting materials and the surveillance materials are consistent with the methodologies of RG 1.99, Rev. 2. Furthermore, the staff concludes that the licensee's PTLR submittal, as supplemented, meets the provisions of PTLR Criterion 7 from Attachment 1 of GL 96-03 and associated action items (24) through (26), as applicable.

4.0 SUMMARY

The staff has completed its review of SCE's license amendment request to remove the P-T limit curves and LTOP system limits for SONGS 2 and 3 from the LCOs in the SONGS 2 and 3 TS and incorporate them into PTLRs that will be controlled through the implementation of administrative controls specified in TS Section 5.7.1.6. On the basis of the staff's review, the staff has determined that the 32 EFPY P-T limits for SONGS 2 and 3 are acceptable and that the contents of SCE's PTLRs for SONGS 2 and 3, as supplemented by letter dated January 12, 2006, conform to the staff's technical criteria for PTLRs, as defined in Attachment 1 of GL 96-03. The staff has also determined that References 1 and 2 satisfy the requirements of 10 CFR Part 50, Appendix G, as modified by exemption, which authorizes the use of the methodology of Reference 3 for generating the P-T limits curves in the SONGS 2 and 3 PTLRs. Furthermore, the staff has determined that the changes to the SONGS 2 and 3 TS for implementing the PTLRs meet the technical criteria of GL 96-03 and are consistent with NUREG-1432, Revision 3, as amended by TSTF-419, for Combustion Engineering plants. Based on this evaluation, the staff concludes that the PTLRs for SONGS 2 and 3 are acceptable and that SCE is authorized to remove the actual P-T limits curves and LTOP system limits from the TS LCOs and incorporate them into a PTLR for each unit that will be administratively controlled by Section 5.7.1.6 of the SONGS 2 and 3 TS. Upon issuance of this license amendment, pursuant to TS Requirement 5.7.1.6c, the licensee will be required to provide the PTLR to the NRC upon issuance for each reactor vessel fluence period and for any PTLR revision or supplement thereto.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published March 1, 2005 (70 FR 9996). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Southern California Edison Company (SCE) Letter to the NRC Document Control Desk, “San Onofre Nuclear Generating Station Units 2 and 3, Docket Nos. 50-361 and 50-362, Proposed Change Number NPF-10/15-551, License Amendment Request, ‘Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)’,” January 28, 2005.

2. Southern California Edison Company (SCE) Letter to the NRC Document Control Desk, “San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362, Proposed Change Number NPF-10/15-551, License Amendment Request, ‘Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)’,” January 12, 2006.

3. Combustion Engineering Owners Group Topical Report CE NPSD-683-A, Revision 6, “Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications,” May 2001.

4. Letter from S. A. Richards, Project Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, to R. Bernier, Chairman, Combustion Engineering Owners Group, “Safety Evaluation of Topical Report CE NPSD-683, Revision 6, ‘Development of a RCS Pressure and Temperature Limits Report [PTLR] for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications’ (TAC No. MA9561),” March 16, 2001.

5. “Program for Irradiation Surveillance of San Onofre Reactor Vessel Materials, San Onofre Nuclear Generating Station, Units 2 and 3,” Combustion Engineering, Inc., Windsor, Connecticut, S-NLM-002, Revision 2, October 1974.

6. “Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of San Onofre Unit 2 Reactor Vessel Materials,” Combustion Engineering, Inc., Windsor, Connecticut, TR-S-MCM-001, May 1, 1978.

7. “Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of San Onofre Unit 3 Reactor Vessel Materials,” Combustion Engineering, Inc., Windsor, Connecticut, TR-S-MCM-003, June 25, 1979.

8. “Analysis of the 263-degree Capsule, Southern California Edison Company, San Onofre Unit 2 Nuclear Generating Station, Reactor Vessel Material Surveillance Program,” BAW-2408, Revision 1, February 2004.

9. A. Ragl, "Southern California Edison Company, San Onofre Unit 2, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program," Combustion Engineering, Inc., Windsor, Connecticut, S-TR-MCS-002, May 27, 1978.

10. "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the San Onofre Nuclear Generating Station Unit 2 (SONGS-2)," (97-Degree Capsule), Battelle Columbus Report dated December, 1988.

11. "Analysis of the 263-degree Capsule, Southern California Edison Company, San Onofre Unit 3 Nuclear Generating Station, Reactor Vessel Material Surveillance Program," BAW-2454, Lynchburg, Virginia, January 2004.

12. A. Ragl, "Southern California Edison Company, San Onofre Unit 3, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program," Combustion Engineering, Inc., Windsor, Connecticut, TR-S-MCM-004, November 30, 1979.

13. "Analysis of the Southern California Edison Company San Onofre Unit 3 Reactor Vessel Surveillance Capsule Removed from the 97° Location," WCAP-12920, Rev. 2, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, May 1994.

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