ENCLOSURE 2

Westinghouse Non-Proprietary document: "Response to NRC Request for Additional Information on WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report" and WCAP-16167-NP, Rev. 0, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report"

Response to NRC Request for Additional Information on WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report"; and WCAP-16167-NP, Rev. 0, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report"

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Revision 0

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1.0 BACKGROUND

In November 2004, WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report" and WCAP-16167-NP, Rev. 0, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report" were provided to Southern California Edison. These reports were then submitted to the NRC by SCE as part of a request for license amendment.

The NRC has reviewed these submittals and has compiled a list of requests for additional information (RAIs). This document provides responses to these RAIs.

2.0 QUALITY ASSURANCE

This work was completed under the requirements of the Westinghouse Quality Assurance Program (Reference 9). References are provided at the end of this document following the RAI responses.

3.0 RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Each NRC RAI is listed by number and is followed by a response.

3.1 RAI #1

In the staff's safety evaluation (SE) on topical report CE-NPSD-683, Revision 6, dated March 16, 2001, the staff included 26 action items that would need to be addressed in a pressure-temperature (P-T) limits report (PTLR) license amendment request that invoked the methods of the topical report. Your PTLR submittal of January 28, 2005, does not specifically identify how the proposed San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3) PTLRs resolve the action items in the SE of March 16, 2001.

The staff requests that you supplement your application with your responses to these 26 action items. If your PTLR submittal already includes information that satisfies any of these action items, please specify which information in the PTLR satisfies resolution of a particular action item. If the PTLR does not include information which satisfies a particular action item, please provide supplemental information which satisfies resolution of the particular action item of concern.

The staff recognizes that several of these action items have become obsolete due to updates in the allowable editions and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, which have been incorporated by reference in Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50). If such an action item falls under this category please designate it as such.

Response:

Responses to the 26 action items of the SE are provided with respect to the SONGS Unit 2 and SONGS Unit 3 PTLRs prepared by Westinghouse in WCAP-16005-NP, Revision 3 for Unit 2 and WCAP-16167-NP, Revision 0 for Unit 3 (References 1 and 2, respectively). For each numbered response below the NRC action item is paraphrased and is followed by a statement as to how the action has been addressed in the respective PTLR, if it has become obsolete, or if it is addressed in a response to another RAI.

1) *Describe the methodology used to calculate the neutron fluence values for the reactor vessel materials.*

   a. *Describe whether the methodology used is consistent with Draft Regulatory Guide 1053.*

      Section 1.0 of References 1 and 2 state that the neutron fluence methodology is consistent with the guidance of Regulatory Guide 1.190.

   b. *Describe the computer codes used to calculate the neutron fluence.*

      The computer codes used to calculate the neutron fluence are described in Section 1.1.1 of References 1 and 2.

   c. *Describe how the computer codes used to calculate the neutron fluence were benchmarked.*

      The computer codes used to calculate the neutron fluence were benchmarked as described in Section 1.4 of References 1 and 2.

2) *Provide the values of neutron fluence used for the adjusted reference temperature calculations including the values for the inner surface (ID), ¼ T and ¾ T locations.*

   a. For Unit 2, Section 4, Tables 4-2 and 4-3 and Subsection 4.4 of Reference 1 provide the following peak values at 32 EFPY: clad/base metal interface is 4.147×10^{19} n/cm^2; ¼ T is 2.472×10^{19} n/cm^2; ¾ T is 0.878×10^{19} n/cm^2.

   b. For Unit 3, Section 4, Tables 4-2 and 4-3 and Subsection 4.4 of Reference 2 provide the following peak values at 32 EFPY: clad/base metal interface is 3.976×10^{19} n/cm^2; ¼ T is 2.370×10^{19} n/cm^2; ¾ T is 0.8419×10^{19} n/cm^2.

3) *Provide the surveillance capsule withdrawal schedule in the PTLR or by reference.*

   The surveillance capsule withdrawal schedule is provided in Section 2, Table 2-2 of References 1 and 2.

4) *Reference the surveillance capsule reports by title and number if the RT_{NDT} values are calculated using RPV surveillance capsule data.*

   The title and number of the surveillance capsule reports are given in References 4 and 15 of Reference 1 (Unit 2) and Reference 2 (Unit 3).

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

This information is provided in Section 3.2.1.3 of the References 1 and 2, and also in the Reference 18 document of these references.

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 topical report.*

This information is provided in Section 3.2.1.2 of References 1 and 2. Also, this section of References 1 and 2 will be revised to remove the statement "...and the equivalent mass addition that results from energy additions" in order to be consistent with the supporting analyses of record. The reference to equivalent mass additions incorrectly implies additional contributions to the transient are provided by the pressurizer heaters and by decay heat.

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

This information is provided in Section 3.2.1.1 of References 1 and 2.

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

The RCS is considered water solid for the transients as described in Section 3.2.1 of References 1 and 2.

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 topical report;*

The analyses of record that support the SONGS Units 2 and 3 LTOP energy addition and mass addition transients were performed using methodologies that preceded approval of the methods described in the topical report. The SONGS energy addition analysis used an assumed value of 1% 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. As noted in Action Item 6 of the staff's safety evaluation, the SONGS mass addition transient does not include decay heat contribution.

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

The transient analyses of References 1 and 2 that support SONGS Units 2 and 3 do not assume operator action for mitigation of the transients.

11) *Provide correlations used for developing power operated relief valve (PORV) discharge characteristics.*

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

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

This information is provided in Section 3.2.1.1 of References 1 and 2.

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

Refer to the response to RAI #6.

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

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

15) *Identify and explain any other deviation from the methodology included in Section 3.0 of the topical report.*

There are no other deviations from the methodology described in Section 3.0 of References 1 and 2.

16) *With respect to the methods used to calculate the adjusted reference temperature, identify the limiting materials and corresponding RT_{NDT} values at ¼ T and ¾ T (where "T" is vessel thickness).*

a. For Unit 2, the RT_{NDT} values at 32 EFPY at the ¼ T and ¾ T locations are provided in Section 4 of Reference 1. Those values are 137.3°F for lower shell plate C-6405-5 and 116.6°F for lower shell plate C-6405-4, respectively.

b. For Unit 3, the RT_{NDT} values at 32 EFPY at the ¼ T and ¾ T locations are provided in Section 4 of Reference 2. Those values are 145.8°F and 125.5°F respectively for intermediate shell plate C-6802-1.

17) *Identify the limiting material and corresponding RT_{PTS} value calculated in accordance with 10 CFR 50.61.*

a. For Unit 2, the RT_{PTS} value at 32 EFPY at the limiting location is provided in Section 4 of Reference 1. That value is 146.3°F for lower shell plate C-6405-5.

b. For Unit 3, the RT_{PTS} value at 32 EFPY at the limiting location is provided in Section 4 of Reference 2. That value is 154.6°F for intermediate shell plate C-6802-1.

18) *Ensure that the ferritic RPV materials that have accumulated neutron fluence in excess of $1x10^{17}$ n/cm^2 will be assessed according to Section 4 of the topical report (Reference 4), regardless of whether the materials are located within the region immediately surrounding the active core.*

a. For both Units 2 and 3, each ferritic plate and weld material located within the region immediately surrounding the active core was evaluated to identify the limiting material at the ¼ T and ¾ T locations. The results of that evaluation are provided in Table 4-4 of References 1 and 2. The adjusted RT_{NDT} values for the limiting materials were used to establish the heatup and cooldown limits. All of the materials assessed in Table 4-4 receive a high neutron fluence being adjacent to the active core and, therefore, are considered in the determination of the limiting beltline material.

b. For the ferritic plate and weld materials located above and below the region immediately surrounding the active core, the neutron fluence is much lower than the fluence relative to the materials immediately surrounding the active core region. The initial RT_{NDT} values of those ferritic materials are comparable to the initial RT_{NDT} values for the materials assessed in Table 4-4. The adjustment to RT_{NDT} will, therefore, also be smaller than for the materials surrounding the active core. Hence, the ferritic plate and weld materials located above and below the region immediately surrounding the active core in the SONGS Units will never be limiting with respect to establishing the heat-up and cool-down limits. (Note that these additional materials comprise the upper shell course plates and welds that are located immediately above the intermediate shell course plates and welds. Each of the lower and intermediate shell course plates and welds are assessed in Table 4-4.)

c. The initial RT_{NDT} values for the ferritic plate and weld materials located above and below the region immediately surrounding the active core are used to establish other aspects of the heat-up and cool-down limits. These other limits include the bolt-up temperature, the lowest service temperature, and the flange limits.

19) *Identify which method (i.e., K_{IC} or K_{IA}) will be used to calculate the reference intensity factor (K_{IR}) values for the RPV as a function of temperature.*

The reference stress intensity factor (SIF) used in both Units 2 and 3 is the equation for K_{IC} given in Appendix G of the ASME Code (Reference 3). It is expressed as:

$$K_{Ic} = 33.2 + 20.734 e^{0.02(T - RT_{NDT})}$$

The parameter T is the temperature of the material at the hypothetical crack tip (°F), RT_{NDT} is the material nil-ductility transition reference temperature (°F), and K_{IC} is the crack initiation fracture toughness (ksi$\sqrt{\text{in}}$).

20) *If Code Case N-640 and K_{IC} are used as the basis for calculating the K_{IR} values; submit an exemption request to use the methods of Code Case N-640 and apply them to the P-T limit calculations.*

The determination of the Pressure-Temperature (P-T) curves is consistent with Section XI of the 2000 Edition of the ASME Code. This Code Edition has incorporated the K_{IC} criteria for the allowable fracture toughness. The Code of Federal Regulations part 50.55a has

approved this ASME Code Edition in Reference 8. Therefore, Code Case N-640 is not used and an exemption request is not submitted.

21) *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.*

[ ]a, c

The justification to support an exemption request for use of the CE NSSS methodology of pressure stress K_{IM} is included as Appendix A of this document.

22) *Include in the PTLRs the P-T curves for heatup, cooldown, criticality, and hydrostatic and leak tests of the reactors.*

P-T curves for heatup, cooldown, hydrostatic test and criticality have been included in the Westinghouse Calculations of References 6 and 7. Details of heatup, cooldown and hydrostatic test curves for the uncorrected case are included as Figures 6 through 27 of this document. Allowable P-T limit curves for the criticality condition were obtained from References 6 and 7 and are shown in Figures 28 through 33 of this document.

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.*

Minimum temperature requirements have been incorporated into the final composite P-T curve limits in Figures 5.1 through 5.3 of the PLTR reports (References 1 and 2). These figures demonstrate that the P-T curves comply with Appendix G to 10 CFR 50.

Please refer to RAI #5 of this document for a further discussion of how compliance is demonstrated.

24) *With respect to the evaluation of plant specific surveillance data, licensees need to include in their PTLRs the supplemental surveillance data and calculations of the chemistry factors if the surveillance data are used for the calculations of the adjusted reference temperatures.*

a. For Unit 2, the determination of chemistry factors is discussed in Section 4 of Reference 1 and the surveillance data are discussed in Section 7. The chemistry factors were determined in accordance with Regulatory Guide 1.99, Revision 2, using Regulatory Position 1.1. The bases for the chemical content of the vessel and surveillance plates and welds are provided in the responses to Generic Letter 92-01 (Reference 16 in Reference 1). Additional

information is also provided in the responses to RAIs #8 and #9. Surveillance data were evaluated but they are not used for the calculations of the adjusted reference temperatures.

b. For Unit 3, the determination of chemistry factors is discussed in Section 4 of Reference 2 and the surveillance data are discussed in Section 7. The chemistry factors were determined in accordance with Regulatory Guide 1.99, Revision 2, using both Regulatory Position 1.1 and 2.1. Additional information is also provided in the response to RAI #9. No supplemental surveillance data were used.

25) *Provide the evaluation of whether the surveillance data are credible in accordance with the credibility criteria of RG 1.99, Revision 02.*

a. For Unit 2, the credibility of the surveillance data is established in Section 7 of Reference 1. Additional information is also provided in the responses to RAIs #8 and #9.

b. For Unit 3, the credibility of the surveillance data is established in Section 7 of Reference 2. Additional information is also provided in the response to RAI #9.

26) *In addition, if licensees seek to use surveillance data from supplemental plant sources, licensees must identify the source of the data, and either identify by title and number the safety evaluation report ....; or compare the licensee's data ... and submit the proposed integrated surveillance program and evaluation of the data for the NRC for review and approval.*

The licensee does not seek to use surveillance data from supplemental plant sources for either Units 2 or 3. In either case there is a plant-specific surveillance program. Four surveillance capsules remain in each unit, and the materials in the surveillance capsules were selected in accordance with ASTM E185-73. In neither case was it necessary or feasible to obtain surveillance data from another reactor vessel surveillance program.

3.2 RAI #2

*The ASME Code, Section XI, Appendix G provides a methodology for calculating stress intensity factors corresponding to membrane tension (K_{IM}) and thermal stress (K_{IT}) for the postulated axial defect. Calculations of K_{IT} are based on stress influence coefficients from finite element modeling (FEM) analyses for inside (¼T) and outside (¾T) surface flaws. Calculations of the maximum allowable K_{IM} are based on a closed-form solution to an equation such as $2K_{IM} + K_{IT} < K_{IC}$, where K_{IT} has been determined from solutions based on stress influence coefficients, and K_{IC} was determined using the equation representing the analytical approximation to the lower bound fracture toughness curve, K_{IC} (in ksi*sqrt(in.)) = 33.2 + 20.734exp[0.02(T - RT_{NDT})], where RT_{NDT} is the material nil- ductility transition reference temperature and T is the actual temperature of the material.*

The Combustion Engineering (CE) nuclear steam supply system (NSSS) methodology differs from the ASME Code, Section XI, Appendix G methodology in several respects. The CE NSSS methodology for calculating K_{IT} is based on thermal influence coefficients from FEM analyses, as opposed to stress influence coefficients. Furthermore, the CE NSSS methodology for calculating K_{IM} does not involve a closed-form solution based on calculations of K_{IT} and K_{IC} factors, and instead applies FEM methods for estimating the K_{IM} factors.

Please supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a discussion of the specific methodologies that will be applied in the PTLRs for SONGS 2 and 3 for calculating stress intensity factors at the 1/4T and 3/4T crack depth locations:

a. *Discuss the methodology for calculating the thermal stress intensity factor, K_{IT}.*
b. *Discuss the methodology for calculating the stress intensity factor corresponding to membrane tension resulting from pressure loading of the reactor vessel, K_{IM}. Please specify whether K_{IM} is determined by obtaining a closed-form solution (as prescribed by the ASME Code, Section XI, Appendix G) or determined by applying FEM methods (as prescribed by the CE NSSS methodology).*

Per your response to action item 21 in RAI 1, if your methodology applies the CE NSSS method for calculating K_{IM} stress intensity values, then your application will need to include a request for an exemption from the requirements of 10 CFR Part 50, Appendix G for P-T limits. The need for an exemption for calculating P-T limits using the CE NSSS method is specified in the fourth paragraph (pages 20- 21) of Section 2.5.4 and in action item 21 (page 27) of Section 5.0 of the SE on topical report CE-NPSD-683, Revision 6, dated March 16, 2001. The CE Owners Group (CEOG) agreed to this requirement in their final version of topical report CE-NPSD-683, Revision 6. The requirement for the exemption is specified in the "Note" on page 5-15 of the topical report.

Response:

Calculations of crack tip stress intensity factors due to thermal loads, K_{IT}, are based on thermal influence coefficients developed using two dimensional (2D) finite element analyses with linear, quadratic and cubic temperature profiles across the vessel wall thickness. The influence coefficients are computed for postulated inside (¼ thickness) and outside (¾ thickness) surface flaws and are then corrected to account for the 3D elliptical geometry using the procedures of Appendix A of Section XI of the ASME Code.

a, c

[ ]

a, c

Using the K_{IM} values due to the applied pressure loading, the ASME Code procedure using the criteria:

$$2K_{IM} + K_{IT} \leq K_{IC}$$

was followed in computing the maximum allowable for thermal transient cases, and

$$1.5K_{IM} \leq K_{IC}$$

for hydrostatic test cases.

Per the conditions listed in the safety evaluation for topical report CE NPSD-683 (Reference 4) a request for exemption from the requirements of 10CFR Part 50, Appendix G, is provided in Appendix A of this document.

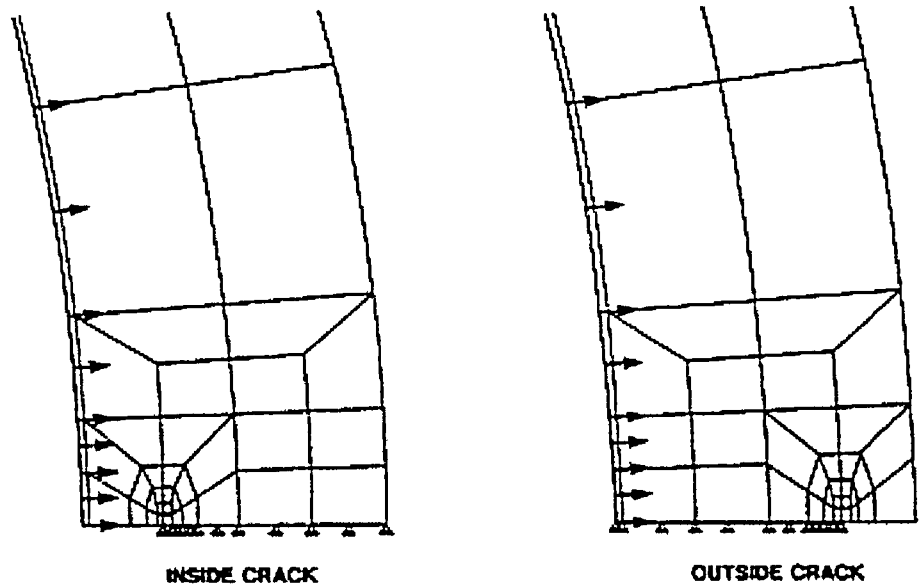


Figure 1: FE Models used in Computing K_{IM} due to Internal Pressure Loading

3.3 RAI #3

In support of the NRC staff's review of the P-T limit curves contained in the PTLR submittal, please supplement your application with data for the through-wall thermal gradients (ΔT) and thermal stress intensities (K_{IT}) for the 1/4T and 3/4T crack depth locations. These data are necessary for the NRC staff to perform independent calculations of P-T limits to verify that the P-T limit curves are at least as conservative as those that would be obtained as a result of applying the methods of 10 CFR Part 50, Appendix G, or as modified using the CE NSSS methodology. In addition, if you are requesting to use the CE NSSS methodology for K_{IM} determinations, please submit the plant-specific K_{IM} data to support the staff's review of these calculations.

Response:

a, c

Figure 2: Unit 2 Heatup 60F/hr Through-wall Thermal Gradients at Crack Tips

a, c

Figure 3: Unit 2 Heatup Thermal Stress Intensity Factors K_{IT} at Outside Crack Tip

a, c

Figure 4: Unit 2 Cooldown 100F/hr Through-wall Thermal Gradients at Crack Tips

a, c

Figure 5: Unit 2 Cooldown Thermal SIF K_{IT} at Inside Crack Tip

The membrane stress intensity factor coefficients K_{IM} due to unit (1000 psi) internal pressure loading for SONGS reactor vessel geometry with a base metal thickness of 8.625 inches for inside and outside crack tip locations are given in the following table.

Table 1: Pressure Stress Influence Coefficients, K_{IM}

a, c

Per the topical report (Reference 4) a request for exemption from the requirements of 10CFR Part 50, Appendix G, is provided in Appendix A of this document.

3.4 RAI #4

In all cases P-T limit curves must be determined using the most limiting conditions in the reactor vessel. For heatup and cooldown transients the application of the PTLR methodology and calculations of P-T limits must always take into consideration the different conditions at the 1/4T and 3/4T locations during the thermal transient, and the resulting P-T limit curves must always represent the most limiting of these conditions.

Please supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a discussion of how the P-T limit curves account for the most limiting conditions in the reactor vessel. The discussion should address the following points:

a. Please discuss how the calculation of P-T limit curves for SONGS 2 and 3 addresses heatup and cooldown transients, specifically taking into consideration the different conditions at the 1/4T and 3/4T crack depth locations.

b. Please discuss how the calculation of P-T limit curves for SONGS 2 and 3 addresses the assessment of the 1/4T location for steady state conditions in addition to the 1/4T and 3/4T locations under heatup and cooldown transient conditions. Please supplement the P-T limit curves for SONGS 2 and 3 with a P-T limit curve representing the 1/4T location under steady state conditions.

<u>**Response:**</u>

a, c

a, c

a, c

Figure 6: Unit 2 Steady-state P-T Curve

a, c

Figure 7: Unit 2 Heatup P-T Curves

a, c

Figure 8: Unit 2 Heatup P-T Curve Enveloped for 5F/hr Transient

a, c

Figure 9: Unit 2 Heatup P-T Curve Enveloped for 10F/hr Transient

a, c

Figure 10: Unit 2 Heatup P-T Curve Enveloped for 30F/hr Transient

a, c

Figure 11: Unit 2 Heatup P-T Curve Enveloped for 40F/hr Transient

a, c

Figure 12: Unit 2 Heatup P-T Curve Enveloped for 60F/hr Transient

a, c

Figure 13: Unit 2 Heatup P-T Curve Envelopes

a, c

Figure 14: Unit 2 Cooldown P-T Curves

a, c

Figure 15: Unit 2 Cooldown P-T Curve Enveloped for 100F/hr Transient

a, c

Figure 16: Unit 2 Cooldown P-T Curve Envelopes

a, c

Figure 17: Unit 3 Steady-state P-T Curve

a, c

Figure 18: Unit 3 Heatup P-T Curves

a, c

Figure 19: Unit 3 Heatup P-T Curve Enveloped for 5F/hr Transient

a, c

Figure 20: Unit 3 Heatup P-T Curve Enveloped for 10F/hr Transient

a, c

Figure 21: Unit 3 Heatup P-T Curve Enveloped for 30F/hr Transient

a, c

Figure 22: Unit 3 Heatup P-T Curve Enveloped for 40F/hr Transient

a, c

Figure 23: Unit 3 Heatup P-T Curve Enveloped 60F/hr Transient

a, c

Figure 24: Unit 3 Heatup P-T Curve Envelopes

a, c

Figure 25: Unit 3 Cooldown P-T Curves

a, c

Figure 26: Unit 3 Cooldown P-T Curve Enveloped for 100F/hr

a, c

Figure 27: Unit 3 Cooldown P-T Curve Envelopes

3.5 RAI #5

Table 1 of 10 CFR Part 50, Appendix G, specifies six different minimum temperature requirements that must be met when generating the pressure-temperature (P-T) limits for U.S. operating pressurized water reactors (PWRs):

a. *Those for pressure test conditions with the Reactor Coolant System (RCS) pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure.*
b. *Those for pressure test conditions with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure.*
c. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure, at times the reactor is not in the critical operating mode.*
d. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure at times the reactor is not in the critical operating mode.*
e. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure at times the reactor is in the critical operating mode.*
f. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure at times the reactor is in the critical operating mode.*

Criterion 6 in Attachment 1 to GL 96-03 states that the above minimum temperature requirements of 10 CFR Part 50, Appendix G shall be incorporated into the P-T limit curves, and PTLRs shall identify minimum temperatures on the P-T limit curves such as the minimum boltup temperature and the hydrotest temperature.

Section 6.0 of the SONGS 2 and 3 PTLRs, provides a listing and brief discussion of the minimum temperature requirements that have been incorporated into the P-T limit curves for SONGS 2 and 3. However, the discussion does not adequately demonstrate how the P-T limit curves for pressure testing conditions and normal operations with the core critical and core not critical will be in compliance with the appropriate minimum temperature requirements as given in Table 1 to Appendix G to 10 CFR Part 50. This information is needed to satisfy action item 23 from staff's safety evaluation (SE) on topical report CE-NPSD-683, Revision 6.

Per your response to action item 23 in RAI 1, update Section 6.0 of the PTLRs for SONGS 2 and 3 to provide a discussion on how the P-T limit curves will meet all of the minimum temperature requirements mandated by Table 1 of 10 CFR Part 50, Appendix G. Include in this discussion the value for the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload and how this value is applied along with minimum permissible hydrostatic test temperature to determine minimum temperature requirements that will be applied to the P-T limit curves for SONGS 2 and 3. This information is necessary to ensure that the SONGS 2 and 3 P-T limit curves will continue to comply with the minimum temperature requirements of Table 1 of 10 CFR Part 50, Appendix G, and that the PTLR will conform to the provisions of Criterion 6 in Attachment 1 to Generic Letter (GL) 96-03.

Response:

The P-T limit curves for pressure testing conditions and normal operations with the core critical and core not critical are in compliance with the appropriate minimum temperature requirements given in Table 1 of Appendix G to 10 CFR Part 50. This is demonstrated by the explanation and figures provided hereafter. Each of the six minimum temperature requirements a) through f) are specifically identified for clarity.

Unit 2

Design pressure	= 2,500 psia (2,485.3 psig)	
Normal operating pressure	= 2,250 psia (2,235.3 psig)	
Preservice hydrostatic pressure	= 3,125 psia (3,110.3 psig)	
Minimum bolt-up temperature	= 65°F	(Ref.12)
Flange region RT_{NDT}	= 20°F	(Ref.11)
Initial piping, pumps and valves RT_{NDT}	= 90°F	(Ref.13)
Adjusted RT_{NDT} at ¼ t for 32 EFPY	= 137.3°F	
Adjusted RT_{NDT} at ¾ t for 32 EFPY	= 116.6°F	
20% Preservice hydrostatic pressure	= 0.2 (3,125 psia) = 625 psia	

Preservice hydrostatic pressure with correction for instrument uncertainty
= 20% preservice hydro pressure + RCS instrument uncertainty
= 625 psia - 97.8 psi = 527.2 psia

Inservice hydrostatic pressure= 1.1 (Operating Pressure) + Pressurizer instrument uncertainty
= 1.1 (2,250 psia) + 81 psi = 2,556 psia

a, c

a, c

Figure 28 shows the minimum pressure and temperature requirements for hydrostatic test and heatup transients for Unit 2. Minimum requirements for cooldown transients for control room and remote shutdown panel are shown in Figures 29 and 30, respectively.

Unit 3

Design Pressure	= 2,500 psia (2,485.3 psig)	
Normal Operating Pressure	= 2,250 psia (2,235.3 psig)	
Preservice Hydrostatic Pressure	= 3,125 psia (3,110.3 psig)	
Minimum Bolt-up Temperature	= 65°F	(Ref.12)
Flange Initial RT_{NDT}	= 40°F	(Ref.15)
Initial piping, pumps and valves RT_{NDT}	= 90°F	(Ref.13 & 16)
Adjusted RT_{NDT} at ¼ t for 32 EFPY	= 145.8°F	
Adjusted RT_{NDT} at ¾ t for 32 EFPY	= 125.5°F	

20% Preservice hydrostatic pressure = 0.2 (3,125 psia) = 625 psia

Preservice hydrostatic pressure with correction for instrument uncertainty
= 20% preservice hydro pressure + RCS instrument uncertainty
= 625 psia - 97.8 psi = 527.2 psia

Inservice hydrostatic pressure= 1.1 (Operating Pressure) + Pressurizer instrument uncertainty
= 1.1 (2,250 psia) + 81 psi = 2,556 psia

a, c

a, c

Figure 31 shows the minimum pressure and temperature requirements for hydrostatic test and heatup transients for Unit 3. Minimum requirements for cooldown transients for control room and remote shutdown panel are shown in Figures 32 and 33, respectively.

a, c

Figure 28: Unit 2 Heatup P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 29: Unit 2 Cooldown P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 30: Unit 2 Cooldown P-T Curves with Min. Temperature Requirements, Remote Shutdown Panel

a, c

Figure 31: Unit 3 Heatup P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 32: Unit 3 Cooldown P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 33: Unit 3 Cooldown P-T Curves with Min. Temperature Requirements, Remote Shutdown Panel

3.6 RAI #6

Section 5.0 of the PTLRs for SONGS 2 and 3 provides a footnote indicating that pressure and temperature limit values are adjusted for instrument uncertainty, and for RCS pressure and elevation effects. Please 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. Include in this discussion numerical values for the instrument uncertainties as well as numerical values for factors that compensate for RCS pressure and elevation effects. Please discuss how these factors are applied in the calculation of the P-T limit curves.

Response:

Tables 5-1, 5-2 and 5-3 of the SONGS PTLRs (References 1 and 2) each have a footnote explaining that the tabulated values have been adjusted for instrument uncertainty and for RCS pressure and elevation effects. Additional detail as below will be added to Section 5 of References 1 and 2 to explain these footnotes.

The calculated reactor vessel pressure and temperature limit values are adjusted for instrument uncertainty, and for RCS pressure and elevation effects. Section 3.4.2 of the topical report (Reference 4) provides a description of the development of these RCS pressure and elevation effects.

These adjustments ensure that the analytical beltline P-T limits are conservatively interpreted by pressurizer pressure and RCS temperature instrumentation. The pressure values are adjusted using pressure correction factors (PCF), and the temperature values are adjusted for temperature instrumentation uncertainty.

The pressure correction factors applied to Tables 5-1, 5-2 and 5-3 consist of three components:

1. pressure differential corresponding to water head between the pressurizer water level and the reference point in the reactor vessel (ΔP_{ELEV}),
2. flow-induced pressure drop between the reactor vessel downcomer and the surge nozzle in the hot leg (ΔP_{FLOW}; a value that depends on the number of operating RCPs), and
3. pressurizer pressure instrumentation loop uncertainty (ΔP_{INSTR}).

These components are individually established using conservative assumptions, then summed into the PCF. The PCF values are subtracted from the analytical values to conservatively reduce the allowable pressure limit. The explicit PCF values used for the SONGS units are dependent upon the number of operating RCPs and on the available instrumentation. Thus, the following PCFs are applied to the analytical pressure limits:

Pressure Correction Factors
(with instrument uncertainty)

		Low Range	Wide Range	
Control Room:		≤ 700 psia		
≤ 340 °F (2 RCP Operating)	PCF =	97.8	151.3	psid
> 340 °F (3 RCP Operating)	PCF =	117.8	171.3	psid
Remote Shutdown Panel:		≤ 1600 psia		
≤ 340 °F (2 RCP Operating)	PCF =	146.3	151.3	psid
> 340 °F (3 RCP Operating)	PCF =	166.3	171.3	psid

The data in Tables 5-1, 5-2 and 5-3 of References 1 and 2 are also adjusted for temperature instrumentation uncertainty. For SONGS, a conservative temperature uncertainty of 18.5°F is added to the analytical values for both the control room and the remote shutdown panel instrumentation.

3.7 RAI #7

The proposed P-T limit curves included in Section 5.0 of the PTLRs for SONGS 2 and 3 are proposed to be effective through 32 effective full power years of operation (EFPY). The existing P-T limit curves contained in the Technical Specifications (TS) are stated to be effective through 20 EFPY. Confirm whether the changes to the P-T limit curves included in Section 5.0 of the PTLRs for SONGS 2 and 3 reflect only the increase in the EFPY for which the curves will be applied. If there are other factors, such as different parameters or methods, which contribute to the changes to the curves, provide a detailed discussion of these factors and how they affect the PTLR P-T limit curves.

Response:

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, via a higher RT_{NDT} shift from 20 to 32 EFPY, in conjunction with the use of the crack arrest allowable, K_{IC}, instead of the crack initiation allowable, K_{IA}. The P-T limit curves included in Section 5.0 of the PTLRs are effective through 32 effective full power years of operation and are based on a 32 year RT_{NDT} shift using the K_{IC} allowable. The P-T limit curves contained in the Technical Specifications (TS) are effective through 20 EFPY and are based on a RT_{NDT} shift of 20 years using the K_{IA} allowable.

3.8 RAI #8

Criterion 7 of the Table in Attachment 1 to GL 96-03 specifies that an analysis of the credibility of the surveillance data must be provided in the PTLR. Regulatory Position 2.1 of Regulatory Guide (RG) 1.99, Revision 2 specifies that when two or more credible surveillance data sets become available from the reactor in question, they may be used to determine the Adjusted Reference Temperature (ART) values. If the procedure of Regulatory Position 2.1 for determining the ART values based on the surveillance data results in a higher value for the ART than that given by using the procedures of Regulatory Position 1.1 of the RG, RG 1.99, Revision 2 specifies that the surveillance data should be used for the ART and chemistry factor determination. If the procedure of Regulatory Position 2.1 results in a lower value for the ART, either may be used.

Please confirm that the credibility analysis of the SONGS 2 surveillance data from Section 7.0 of the SONGS 2 PTLR demonstrated that the surveillance data sets for SONGS 2 are credible.

Section 7.0 of the SONGS 2 PTLR states that the surveillance data were not used to generate a chemistry factor in accordance with the methodology prescribed in Regulatory Position 2.1 of RG 1.99, Revision 2. Please confirm whether the ART values for the limiting materials were calculated using the procedure of Regulatory Position 1.1 of RG 1.99, Revision 2.

If the procedure of Regulatory Position 1.1 of RG 1.99, Revision 2 was used to calculate the ART values for the limiting materials, please indicate why this is an acceptable procedure, given the credibility of the surveillance data.

Please supplement Section 7.0 of the PTLR for SONGS 2 with the following information:

a. *Table 7-1 of the SONGS 2 PTLR provides chemistry factors for the two surveillance materials plate C-6404-2 and weld 9-203. Please indicate how these chemistry factors were derived.*
b. *There is no explicit calculation in the SONGS 2 PTLR demonstrating that chemistry factor values for the limiting materials derived from the tables in RG 1.99, Revision 2 would result in limiting ART values that are more conservative than those determined using chemistry factors derived from surveillance data. Per your response to action item 24 in RAI 1 please supplement Section 7.0 of the PTLR for SONGS 2 with detailed calculations of the chemistry factors for each of the surveillance materials based on the calculation methods specified in Regulatory Position 2.1 of RG 1.99, Revision 2.*

The calculations of the chemistry factors for the surveillance materials for SONGS 3, provided in Table 7-1 of the SONGS 3 PTLR represent an acceptable format for presenting surveillance material chemistry factor calculations.

Response:

It is confirmed that a credibility analysis was performed in accordance with Regulatory Guide 1.99, Revision 2, and it was determined that all of the credibility criteria were met for the SONGS 2 surveillance base plate (Plate C-6404-2) and surveillance weld (Weld 9-203). It is also confirmed that ART values for all the SONGS 2 reactor vessel beltline materials were calculated using Regulatory Position 1.1 of RG 1.99, Revision 2. The results of that calculation are reported in Table 4-4 of Reference 1. Additional details are provided in the specific responses to questions 8.a and 8.b.

Information on the selection of materials for inclusion in the Unit 2 surveillance program is provided in Reference 17. The selection followed the procedures of ASTM E-185-73, Annex A-1. Two surveillance capsules have been removed from Unit 2, those from the 97° and 263° locations. Data from those capsules were used to compute an Adjusted Reference Temperature (ART) using Regulatory Guide 1.99, Revision 2, Regulatory Position 2.1. The ART values for plate C-6404-2 and the surveillance weld determined from the surveillance data were less than that predicted for plate C-6404-5 using Position 1.1 of RG 1.99. That is, the two surveillance data sets do not represent the limiting vessel material. The responses to questions 8.a and 8.b below provide additional information regarding the treatment of the surveillance data.

The heatup and cooldown limits and the assessment of RT_{PTS} require the determination of the highest (i.e., limiting) value of ART. These values are reported in Section 4 of Reference 1. The limiting material (highest ART value at 1/4T or 3/4T based on Regulatory Position 1.1 of RG 1.99, Revision 2) for the SONGS 2 reactor vessel beltline is plate C-6404-5. The ART for plate C-6404-2 based on the credible surveillance data and Regulatory Position 2.1 is lower than that for the limiting plate. The ART for the surveillance weld based on the credible surveillance data and Regulatory Position 2.1 is also lower than that for the limiting plate. The predicted values of RT_{PTS} are in the same relative order as the ART values. Hence, the procedure used produces a more conservative value for the SONGS 2 reactor vessel than that obtained using the credible surveillance data and Regulatory Position 2.1. New surveillance data will be reviewed in accordance with the requirements in effect.

a. Response to RAI #8.a

The chemistry factors for the surveillance materials provided in Table 7-1 of Reference 1 are obtained from Tables 1 and 2 of RG 1.99, Revision 2, based on the following copper and nickel content:

CF = 65°F for Plate C-6404-2, based on 0.10% copper content and 0.60% nickel content per Reference 18. Also, refer to Table 2 in the response to 8.b, below.

CF = 31.1°F for Weld 9-203, based on 0.03% copper content and 0.14% nickel content. These represent the average values of the 97° and the 263° surveillance capsules per Reference 20. Baseline analysis showed 0.03% copper content and 0.12% nickel content (Reference 20), which corresponds to a slightly lower CF = 29.8°F.

b. Response to RAI #8.b

Two analyses were performed by Combustion Engineering Chattanooga Laboratory on Unit 2 reactor vessel beltline plates and welds (Reference 18), including intermediate shell Plate C-6404-2 and lower shell Plate C-6404-5. The copper and nickel content based on these analyses results are provided in Table 2.

Table 2: Copper and Nickel Content by Weight (percent)

Plate	First Analysis		Second Analysis		Average	
	Cu	Ni	Cu	Ni	Cu	Ni
C-6404-2	0.10	0.58	0.10	0.60	0.10	0.59
C-6404-5	0.11	0.62	0.11	0.67	0.11	0.65

Per Table 2 of Regulatory Guide 1.99 (Reference 19), the chemistry factor for Plate C-6404-2 is 65°F, and the chemistry factor for C-6404-5 is 75°F based on the average Cu and Ni values provided in Table 2 above.

b.1 ART for Intermediate Shell Plate C-6404-2

The following two surveillance data sets are available for intermediate shell Plate C- 6404-2:

i) The 97° surveillance capsule, which was removed from the reactor vessel during the Cycle 4 refueling outage, i.e., at the end of Fuel Cycle 3, and

ii) The 263° surveillance capsule, which was removed from the reactor vessel during the Cycle 11 refueling outage, i.e., at the end of Fuel Cycle 10.

The SONGS Unit 2 surveillance data were evaluated to show that the base plate and the weld met the five credibility criteria provided in RG 1.99, Revision 2. The Adjusted Reference Temperature (ART) can be calculated for intermediate shell Plate C-6404-2 and Weld 9-203 based on the two credible surveillance data sets as follows:

- Per Reference 20, the fluence values are 0.507 for the 97° surveillance capsule, and 2.188 for the 263° surveillance capsule (units:10^{19} n/cm^2). The corresponding fluence factor ($ff = f^{(0.28 - 0.1 \log f)}$) values are calculated per Position 1.1. It follows that:

  ff = 0.810 based on the 97° surveillance capsule, and
  ff = 1.212 based on the 263° surveillance capsule

  The sum of the squares, Σff^2, is given by:

  $$\Sigma ff^2 = 0.81^2 + 1.212^2 = 2.127$$

- Per Reference 8.4, the measured ΔRT_{NDT} values are:

  $\Delta RT_{NDT} = 41°F$ for the 97° surveillance capsule
  $\Delta RT_{NDT} = 88°F$ for the 263° surveillance capsule

  It follows that:

  $$\Sigma (ff \times \Delta RT_{NDT}) = 0.810 \times 41 + 1.212 \times 88 = 139.9°F$$

- Per Position 2.1, the chemistry factor, CF, is given by:

  $$CF = \Sigma (ff \times \Delta RT_{NDT}) / \Sigma ff^2 = 65.8°F$$

- The values of f, ff and ΔRT_{NDT} projected for 32 EFPY are calculated using Position 1.1 methodology with x = 2.375 inches at the 1/4 T location:

  $f = 2.4436(\times 10^{19}$ n/cm$^2)$
  ff = 1.2405
  $\Delta RT_{NDT} = CF \times ff = 81.6°F$

- The projected Adjusted Reference Temperature (ART) for Plate C-6404-2 for 32 EFPY at the 1/4 T location is calculated based on Position 1.1 as follows:

  ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin

where

  Initial RT_{NDT} = 20°F per Reference 18
  Margin = 17°F per Position 2.1

Therefore,

  ART= 20 + 81.6 + 17 = **118.6°F**

In the above analysis, the fluence value for the intermediate shell plates was used.

b.2 ART for Lower Shell Plate C-6404-5

- Per Table 2 of Reference 19, CF = 75°F for lower shell Plate C-6404-5.
- Per Reference 8.2, the projected surface fluence for 32 EFPY = 4.3707 ($x10^{19}$ n/cm^2).
- Using Position 1.1 calculation methodology, the projected fluence for 32 EFPY at the 1/4T location for lower shell plates is:

  f = 2.4717 ($x10^{19}$ n/cm^2)
  ff = 1.2434

- The projected shift, ΔRT_{NDT}, is given by:

  ΔRT_{NDT} = CF x ff = 93.3°F
  Initial RT_{NDT} = 10°F (Reference 18)
  Margin = 34°F (per Position 1.1)

- It follows that ART for Plate C-6404-5 projected for 32 EFPY at the 1/4 T location is given by:

  ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin
  = 10 + 93.3 + 34 = **137.3°F**

In the calculation above, the fluence values for the lower shell plates were used.

Based on the analyses results in Sections b.1 and b.2 above, the ART for Plate C-6404-5 (137.3°F) is higher then that for Plate C-6404-2 (118.6°F), so Plate C-6404-5 is bounding.

b.3 ART for Lower Shell Weld 9-203

- Per Reference 18, the results of the chemical analysis performed by Combustion Engineering on Weld 9-203 showed an as-deposited copper content of 0.07% and nickel content of 0.29% by weight. The chemistry factor CF = 69 per Table 1 of Reference 19.

- Using the values of f = 2.4717 ($x10^{19}$ n/cm^2), and ff = 1.2434 calculated for the lower shell at the 1/4T location in Section b.2 above, ART for Weld 9-203 can be calculated as follows:

  ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin

where

Initial RT_{NDT} = -60°F per Reference 18
Margin = 56°F per Position 1.1
ΔRT_{NDT} = CF x ff = 85.8°F

Therefore,

ART= -60 + 85.8 + 56 = **81.8°F**

- Alternatively, two sets of surveillance data exist for Weld 9-203. Using Position 2.1, Weld 9-203 ART of 8.4°F is calculated in the response to RAI #9.

Therefore, the value of CF = 69°F calculated based on Position 1.1 is higher than that based on Position 2.1 using the surveillance data, CF = 32.5. Hence, the bounding estimate of ART for Weld 9-203 is 81.8°F. Note that the ART value of 137.3°F for Plate C-6404-5 is actually bounding and was used to generate the P-T curves.

New surveillance data will be reviewed in accordance with the requirements in effect.

3.9 RAI #9

Regulatory Position 2.1 of RG 1.99, Revision 2 states that if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, the measured values of ΔRT should be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld.

Please indicate in the SONGS 2 and 3 PTLRs whether the copper and nickel content of the surveillance weld differs from that of the vessel weld. If so, please supplement Section 7.0 of the PTLRs for SONGS 2 and 3 with detailed calculations for determining the adjustments to the measured values for DRT for the surveillance weld, and indicate whether these adjusted values of DRT were used in the determination of the chemistry factor for the surveillance weld.

Response:

SONGS Unit 2

The copper and nickel content for the surveillance weld for Unit 2 differs from the as-deposited vessel weld, Weld 9-203, as shown in Table 3. Table 3 also provides the chemistry factor obtained using Table 2, RG 1.99, Revision 2.

Table 3: SONGS 2 Best Estimate Weld Chemical Composition

Weld	Cu %	Ni %	CF (°F)
Vessel Weld 9-203$^{(1)}$	0.07	0.29	69
Surveillance Weld$^{(2)}$	0.03	0.12	30
Surveillance Weld$^{(3)}$	0.03	0.15	32

(1) Reference 18.
(2) Reference 21.
(3) Reference 18 provided these values based on analysis of the 97° capsule.

The value CF = 31.1°F was reported in Reference 20, and is considered reasonable based on the data in Table 3 above.

Table 4 provides the fluence values, the fluence factors and the measured ΔRT_{NDT} values for the 97° and the 263° surveillance capsules. The table also provides the adjusted ΔRT_{NDT} values obtained per Position 2.1 as follows:

Adjusted ΔRT_{NDT} = measured ΔRT_{NDT} values multiplied by the ratio of chemistry factors; where the ratio is the vessel weld chemistry factor divided by the surveillance weld chemistry factor. It follows that the ratio of chemistry factors = 69/31.1= 2.22.

Table 4: Adjusted Unit 2 Weld ART Based on Chemical Composition

	$f^{(1)}$ (10^{19} n/cm^2)	ff	Measured ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} (°F)
97° capsule	0.507	0.810	4.0	8.9
263° capsule	2.188	1.212	23.0	51.0

(2) The fluence values were obtained from Reference 20.

Using the two sets of data in Table 4, the value of ART for 32 EFPY is calculated as follows:

$\Sigma\, ff^2 = 2.127$
$\Sigma(ff \times \Delta RT_{NDT}) = 0.810 \times 8.9 + 1.212 \times 51.0 = 69.1°F$

Per Position 2.1, the chemistry factor, CF, is given by:

$$CF = \Sigma\,(ff \times \Delta RT_{NDT}) / \Sigma\, ff^2 = 32.5°F$$

Per Reference 20, the extrapolated fluence value (f) for the lower shell for 32 EFPY is 2.472 ($\times 10^{19}$ n/cm^2), and the corresponding fluence factor (ff) is 1.243. It follows that:

$\Delta RT_{NDT} = 32.5 \times 1.243 = 40.4°F$

Therefore, the projected ART value for 32 EFPY is given by:

ART $=$ Initial $RT_{NDT} + \Delta RT_{NDT} +$ Margin
$= -60 + 40.4 + 28 =$ **8.4°F**

where

Initial RT_{NDT} = -60°F per Reference 18
Margin = 28°F per Position 2.1

The projected ART value of 8.4°F calculated for Weld 9-203 for 32 EFPY is significantly lower than the limiting ART of 137.3°F for Plate C-6404-5, calculated as part of the response to RAI #8 above. The ART value of 137.3°F was used to generate the P-T curves.

<u>SONGS Unit 3</u>

The copper and nickel content for the surveillance weld for Unit 3, Weld 9-203, differs from the as-deposited vessel weld as shown in Table 5. Table 5 also provides the chemistry factor obtained using Table 2, RG 1.99, Revision 2.

For the purpose of this calculation, the vessel as-deposited weld CF = 33.6°F and surveillance weld CF = 27.2°F are conservatively used to maximize the adjustment to ΔRT_{NDT}.

Table 5: SONGS 3 Best Estimate Weld Chemical Composition

	Cu %	Ni %	CF (°F)
Vessel Weld 9-203 (1)	0.06	0.04	33.6
Vessel Weld 9-203 (1)	0.05	0.04	30.6
Surveillance Weld (1)	0.03	0.11	29.2
Surveillance Weld (1)	0.03	0.09	27.9
Surveillance Weld (2)	0.03	0.08	27.2

(1) Reference 18.
(2) Reference 22.

Examination of Table 5 shows that the available chemical analysis measurements for the surveillance weld and the vessel weld are essentially the same, and no adjustment to the values of ΔRT_{NDT} is warranted. Table 6 provides the fluence values, the fluence factors and the measured ΔRT_{NDT} values for the 97° and the 263° surveillance capsules.

Table 6: Unit 3 Surveillance Weld ΔRT_{NDT}

	f (1) (10^{19} n/cm^2)	ff	Measured ΔRT_{NDT} (°F)
97° capsule	0.8	0.937	29
263° capsule	2.471	1.243	72

(1) The fluence values were obtained from Reference 23.

Using the two sets of data in Table 6, the value of ART for 32 EFPY is calculated as follows:

$$\Sigma\, ff^2 = 2.424$$
$$\Sigma\,(ff \times \Delta RT_{NDT}) = 0.937 \times 29 + 1.243 \times 72 = 116.7°F$$

Per Position 2.1, the chemistry factor, CF, is given by:

$$CF = \frac{\Sigma\,(ff \times \Delta RT_{NDT})}{\Sigma\, ff^2} = 48.1°F$$

Per Reference 23, the extrapolated fluence value (f) for the lower shell for 32 EFPY is 2.37 (x10^{19} n/cm^2), and the corresponding fluence factor (ff) is 1.233. It follows that:

$$\Delta RT_{NDT} = 48.1 \times 1.233 = 59.3°F$$

Therefore, the projected ART value for 32 EFPY is given by:

$$\begin{aligned} ART &= \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \\ &= -50 + 59.3 + 28 = \mathbf{37.3°F} \end{aligned}$$

where

Initial RT_{NDT} = -50°F	The higher of the two values reported in Reference 18

and

Margin= 28°F	Per Position 2.1.

The projected ART value of 37.3°F calculated for Weld 9-203 for 32 EFPY is significantly lower than the limiting ART of 145.8°F for Plate C-6802-1, which was used to generate the P-T curves for Unit 3. The projected ART value calculated using Position 1.1 is 33.3°F, which is similar to that calculated using Position 2.1.

4.0 REFERENCES

1. WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report," dated November 2004, Westinghouse Electric Company.
2. WCAP-16167-NP, Rev. 00, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report," dated November 2004, Westinghouse Electric Company.
3. ASME Code Section XI, Division 1, Appendix G, July 1999.
4. CEOG Task 1174 Final Report CE NPSD-683-A, Rev. 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.
5. ABB Report 063-PENG-ER-096, Rev. 00, "Technical Methodology Paper Comparing ABB/CE PT Curve to ASME Section III, Appendix G," January 22, 1998.
6. Westinghouse CalcNote CN-CI-02-54, Rev. 03, "SONGS Unit 2 RCS Pressure-Temperature Limits and LTOP Enable Temperatures for 32 EFPY," October 08, 2004.
7. Westinghouse CalcNote CN-CI-04-37, Rev. 01, "SONGS Unit 3 RCS Pressure-Temperature Limits and LTOP Enable Temperatures for 32 EFPY," October 29, 2004.
8. Code of Federal Regulations 10 CFR Part 50.55a Codes and Standards, Section 1 Background, September 26, 2002.
9. Westinghouse Policies & Procedures, Nuclear Services Edition, Revision 21.
10. CE Instruction Manual, "Reactor Vessel Assembly, San Onofre Unit No. 2, Southern California Edison," Book No. 71170, Vol. 1.
11. SCE Calculation No. M-0011-071, Rev. 2, "SONGS Unit 2 Adjusted Referenced Temperature for 20 & 32 EFPY," November 20, 2003.
12. SCE Calculation No. M-0011-063, Rev. 01, "Revised PT Curves for 20 EFPY," May 31, 1994
13. Westinghouse Correspondence, S-MCM-80-126, "Materials Data for SoCal 2&3 Pressure Temperature Limit Curves," February 8, 1980.
14. CE Design Report CENC-1292 "Analytical Report for Southern California, San Onofre Unit 3 Reactor Vessel," August 1977.
15. Attachment I, Calculation M-DSC-373, "Reactor Pressure Vessel Minimum Bolt-up Temperature" to Southern California Edison Letter to U.S. Nuclear Regulatory Commission, "Docket Nos. 50-361 and 50-362, Proposed Technical Specification Change Number NPF-10/15-516, Reduce the Minimum Bolt-up Temperature for Reactor Vessel Head Bolts when they are Tensioned, San Onofre Nuclear Generating Station Units 2 and 3," sent May 3, 2000.
16. Westinghouse Correspondence S-MCM-130, "RCP Material Data for San Onofre III," August 1, 1980.
17. CE Report TR-S-MCM-001-P (SCE Document Control / Transmittal Number C780501G0078),"Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of San Onofre - Unit 2 Reactor Vessel Materials."
18. Letter from Walter C. March, SCE, to U. S. Nuclear Regulatory Commission Dated June 22, 1994. Subject: Docket Nos. 50-361 and 50-362, Revision to Supplemental Response to Generic Letter 92-01, Revision 1,"Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" San Onofre Nuclear Generating Station Units 2 and 3.

19. Regulatory Guide 1.99, Revision 2.
20. SCE Report SO23-901-C264, Revision 1,"Analysis of the 263° Capsule, Southern California Edison Company, San Onofre Unit 2 Nuclear Generating Station."
21. CE Report S-TR-MCS-002 dated May 27, 1978 (SCE Document CDCC 56396), "Southern California Edison San Onofre Unit 2, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program."
22. CE Report TR-S-MCM-004 (SCE Document Control / Transmittal Number C791130G0097),"Southern California Edison San Onofre Unit 3, Evaluation of Baseline Specimens."
23. SCE Report SO23-901-C274, Rev. 1, "Analysis of the 263° Capsule, Southern California Edison Company, San Onofre Unit 3 Nuclear Generating Station."

Appendix A

Justification for Exemption to Apply Alternate Method of Calculating the Stress Intensity Factor K_{IM} due to Internal Pressure Loading for San Onofre Units 2 and 3

Introductory Statements:

Pursuant to 10CFR50.90, Southern California Edison (SCE) hereby requests the use of an alternate methodology for calculation of crack tip stress intensity factor K_{IM} for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 reactor vessel beltline regions subjected to internal pressure loading. This alternate methodology by Westinghouse has been used in topical report CE NPSD-683, Revision 6, dated March 16, 2001. This methodology was developed and used in 1998 for Indian Point Unit 3 in a previous submittal report (Docket No. 50-286, TAC No. M99928). The approach was reviewed by the NRC and concluded to be acceptable (Section 2.5.4 paragraph 4 of the report).

A justification for the application of this alternate method to SONGS is given below.

Justification for Exemption

10 CFR 50.60(b) allows usage of alternatives to the requirements described in Appendix G and H of 10 CFR 50 when the exemption is granted by the NRC.

In accordance with 10 CFR 50.12(a), Southern California Edison (SCE) requests an exemption from the regulations of 10CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation." The exemption request would allow SONGS Units 2 and 3 to use an alternate methodology for calculation of crack tip stress intensity factor K_{IM} for SONGS reactor vessel beltline regions subjected to internal pressure loading, in lieu of the methodology cited in ASME Boiler and Pressure Vessel Code, Appendix G.

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10CFR50 which are:

1) authorized by law;
2) will not present an undue risk to the public health and safety;
3) consistent with the common defense and security; and
4) special circumstances, as defined by 10 CFR 50.12(a)(2) are present.

The standards for the exemption are justified, as described below.

1) **The requested exemption is authorized by law.**

   The NRC is authorized by law to grant this exemption. Requirements in 10 CFR 50.60 state that the use of alternative methods to 10 CFR 50, Appendix G is acceptable when an exemption is granted by the NRC.

2) **The requested exemption does not present an undue risk to the public health and safety.**

The proposed exemption request has no impact on the safe operation of the plant. An exemption from the requirements would allow the use of an alternate methodology to calculate the membrane loading stress intensity factor. Specifically, this methodology uses a finite element base influence function under internal pressure loading. The results of this methodology are comparable to the results obtained using the ASME Appendix G methodology as demonstrated below.

ASME Code uses the procedure in Article G-2214 for axial surface flaws. In this procedure, stress intensity factor under internal pressure loading is given by (U.S. Customary units)

$$K_{IM} = M_M\left(\frac{pR_i}{t}\right)$$

The magnification factor M_M for inside axial flaws is given by:

$$M_M = 1.85 \quad \text{for } \sqrt{t} < 2 \text{ or } t < 4$$

$$M_M = 0.926\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464 \text{ or } 4 \le t \le 12$$

$$M_M = 3.21 \text{ for } \sqrt{t} > 3.464 \text{ or } t > 12$$

and for outside flaws by:

$$M_M = 1.77 \quad \text{for } \sqrt{t} < 2 \text{ or } t < 4$$

$$M_M = 0.893\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464 \text{ or } 4 \le t \le 12$$

$$M_M = 3.09 \text{ for } \sqrt{t} > 3.464 \text{ or } t > 12$$

where:

K_{IM} is the stress intensity factor for membrane loads (ksi√in);
p is the internal pressure (ksi);
R_i is the vessel inner radius (in); and
t is the vessel wall thickness (in).

a, c

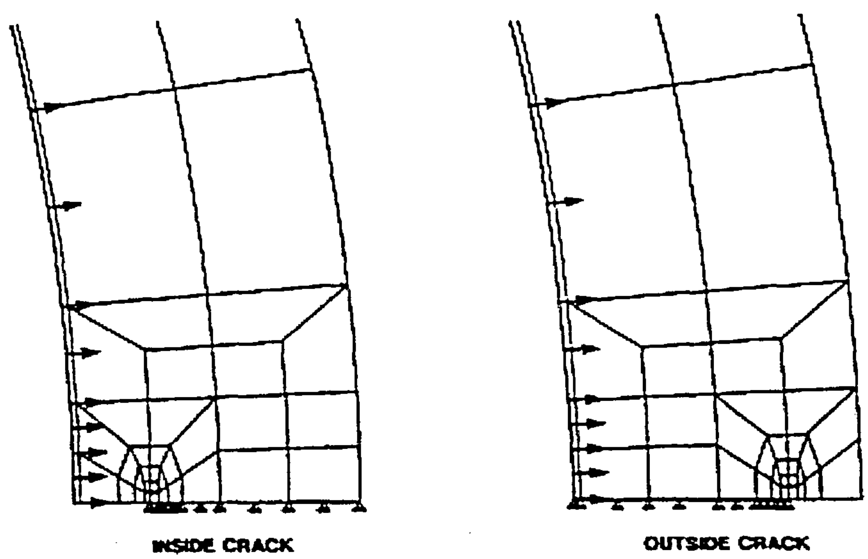
[ ]

$$k_{IM} = M_M\left(\frac{R_i}{t}\right)$$

and the applied stress intensity factor K_{IM} for any given pressure, p, in ksi, is given by

$$K_{IM} = pk_{IM}$$

a, c



FEMs used in Computing k_{IM} due to Internal Pressure Loading

a, c

[ ]

Therefore, this exemption request does not present an undue risk to the public health and safety.

3) **The requested exemption will not endanger the common defense and security.**

   The common defense and security are not affected by this exemption request.

4) **Special Circumstances, as defined by 10 CFR 50.12(a)(2) are present.**

   10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption unless special circumstances are present. This exemption meets the special circumstances listed in 10 CFR 50.12(a)(2)(ii).

   10 CFR 50.12(a)(2)(ii) – Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

   The primary purpose of 10 CFR 50.60 is to protect the reactor vessel against non-ductile failure. The use of the Westinghouse alternate methodology requested by this exemption provides greater operational flexibility while still maintaining reactor vessel integrity. In addition, use of the Westinghouse methodology to generate pressure-temperature curves yields comparable results to the use of the ASME Appendix G methodology. Therefore, the reactor vessel is protected against non-ductile failure and the underlying purpose of the rule is achieved.

Conclusion

The use of the Westinghouse alternate methodology to calculate the membrane stress intensity factor, K_{IM}, provides comparable results to that of the ASME Section XI, Division I, Appendix G, and provides adequate protection of the reactor vessel against non-ductile failure.

Response to NRC Request for Additional Information on WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report"; and WCAP-16167-NP, Rev. 0, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report"

December 14, 2005

Revision 0

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1.0 BACKGROUND

In November 2004, WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report" and WCAP-16167-NP, Rev. 0, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report" were provided to Southern California Edison. These reports were then submitted to the NRC by SCE as part of a request for license amendment.

The NRC has reviewed these submittals and has compiled a list of requests for additional information (RAIs). This document provides responses to these RAIs.

2.0 QUALITY ASSURANCE

This work was completed under the requirements of the Westinghouse Quality Assurance Program (Reference 9). References are provided at the end of this document following the RAI responses.

3.0 RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Each NRC RAI is listed by number and is followed by a response.

3.1 RAI #1

In the staff's safety evaluation (SE) on topical report CE-NPSD-683, Revision 6, dated March 16, 2001, the staff included 26 action items that would need to be addressed in a pressure-temperature (P-T) limits report (PTLR) license amendment request that invoked the methods of the topical report. Your PTLR submittal of January 28, 2005, does not specifically identify how the proposed San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 and 3) PTLRs resolve the action items in the SE of March 16, 2001.

The staff requests that you supplement your application with your responses to these 26 action items. If your PTLR submittal already includes information that satisfies any of these action items, please specify which information in the PTLR satisfies resolution of a particular action item. If the PTLR does not include information which satisfies a particular action item, please provide supplemental information which satisfies resolution of the particular action item of concern.

The staff recognizes that several of these action items have become obsolete due to updates in the allowable editions and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, which have been incorporated by reference in Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50). If such an action item falls under this category please designate it as such.

Response:

Responses to the 26 action items of the SE are provided with respect to the SONGS Unit 2 and SONGS Unit 3 PTLRs prepared by Westinghouse in WCAP-16005-NP, Revision 3 for Unit 2 and WCAP-16167-NP, Revision 0 for Unit 3 (References 1 and 2, respectively). For each numbered response below the NRC action item is paraphrased and is followed by a statement as to how the action has been addressed in the respective PTLR, if it has become obsolete, or if it is addressed in a response to another RAI.

1) *Describe the methodology used to calculate the neutron fluence values for the reactor vessel materials.*

   a. *Describe whether the methodology used is consistent with Draft Regulatory Guide 1053.*

      Section 1.0 of References 1 and 2 state that the neutron fluence methodology is consistent with the guidance of Regulatory Guide 1.190.

   b. *Describe the computer codes used to calculate the neutron fluence.*

      The computer codes used to calculate the neutron fluence are described in Section 1.1.1 of References 1 and 2.

   c. *Describe how the computer codes used to calculate the neutron fluence were benchmarked.*

      The computer codes used to calculate the neutron fluence were benchmarked as described in Section 1.4 of References 1 and 2.

2) *Provide the values of neutron fluence used for the adjusted reference temperature calculations including the values for the inner surface (ID), ¼ T and ¾ T locations.*

   a. For Unit 2, Section 4, Tables 4-2 and 4-3 and Subsection 4.4 of Reference 1 provide the following peak values at 32 EFPY: clad/base metal interface is 4.147×10^{19} n/cm^2; ¼ T is 2.472×10^{19} n/cm^2; ¾ T is 0.878×10^{19} n/cm^2.

   b. For Unit 3, Section 4, Tables 4-2 and 4-3 and Subsection 4.4 of Reference 2 provide the following peak values at 32 EFPY: clad/base metal interface is 3.976×10^{19} n/cm^2; ¼ T is 2.370×10^{19} n/cm^2; ¾ T is 0.8419×10^{19} n/cm^2.

3) *Provide the surveillance capsule withdrawal schedule in the PTLR or by reference.*

   The surveillance capsule withdrawal schedule is provided in Section 2, Table 2-2 of References 1 and 2.

4) *Reference the surveillance capsule reports by title and number if the RT_{NDT} values are calculated using RPV surveillance capsule data.*

   The title and number of the surveillance capsule reports are given in References 4 and 15 of Reference 1 (Unit 2) and Reference 2 (Unit 3).

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

This information is provided in Section 3.2.1.3 of the References 1 and 2, and also in the Reference 18 document of these references.

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 topical report.*

This information is provided in Section 3.2.1.2 of References 1 and 2. Also, this section of References 1 and 2 will be revised to remove the statement "...and the equivalent mass addition that results from energy additions" in order to be consistent with the supporting analyses of record. The reference to equivalent mass additions incorrectly implies additional contributions to the transient are provided by the pressurizer heaters and by decay heat.

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

This information is provided in Section 3.2.1.1 of References 1 and 2.

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

The RCS is considered water solid for the transients as described in Section 3.2.1 of References 1 and 2.

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 topical report;*

The analyses of record that support the SONGS Units 2 and 3 LTOP energy addition and mass addition transients were performed using methodologies that preceded approval of the methods described in the topical report. The SONGS energy addition analysis used an assumed value of 1% 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. As noted in Action Item 6 of the staff's safety evaluation, the SONGS mass addition transient does not include decay heat contribution.

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

The transient analyses of References 1 and 2 that support SONGS Units 2 and 3 do not assume operator action for mitigation of the transients.

11) *Provide correlations used for developing power operated relief valve (PORV) discharge characteristics.*

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

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

This information is provided in Section 3.2.1.1 of References 1 and 2.

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

Refer to the response to RAI #6.

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

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

15) *Identify and explain any other deviation from the methodology included in Section 3.0 of the topical report.*

There are no other deviations from the methodology described in Section 3.0 of References 1 and 2.

16) *With respect to the methods used to calculate the adjusted reference temperature, identify the limiting materials and corresponding RT_{NDT} values at ¼ T and ¾ T (where "T" is vessel thickness).*

a. For Unit 2, the RT_{NDT} values at 32 EFPY at the ¼ T and ¾ T locations are provided in Section 4 of Reference 1. Those values are 137.3°F for lower shell plate C-6405-5 and 116.6°F for lower shell plate C-6405-4, respectively.

b. For Unit 3, the RT_{NDT} values at 32 EFPY at the ¼ T and ¾ T locations are provided in Section 4 of Reference 2. Those values are 145.8°F and 125.5°F respectively for intermediate shell plate C-6802-1.

17) *Identify the limiting material and corresponding RT_{PTS} value calculated in accordance with 10 CFR 50.61.*

a. For Unit 2, the RT_{PTS} value at 32 EFPY at the limiting location is provided in Section 4 of Reference 1. That value is 146.3°F for lower shell plate C-6405-5.

b. For Unit 3, the RT_{PTS} value at 32 EFPY at the limiting location is provided in Section 4 of Reference 2. That value is 154.6°F for intermediate shell plate C-6802-1.

18) *Ensure that the ferritic RPV materials that have accumulated neutron fluence in excess of $1x10^{17}$ n/cm^2 will be assessed according to Section 4 of the topical report (Reference 4), regardless of whether the materials are located within the region immediately surrounding the active core.*

a. For both Units 2 and 3, each ferritic plate and weld material located within the region immediately surrounding the active core was evaluated to identify the limiting material at the ¼ T and ¾ T locations. The results of that evaluation are provided in Table 4-4 of References 1 and 2. The adjusted RT_{NDT} values for the limiting materials were used to establish the heatup and cooldown limits. All of the materials assessed in Table 4-4 receive a high neutron fluence being adjacent to the active core and, therefore, are considered in the determination of the limiting beltline material.

b. For the ferritic plate and weld materials located above and below the region immediately surrounding the active core, the neutron fluence is much lower than the fluence relative to the materials immediately surrounding the active core region. The initial RT_{NDT} values of those ferritic materials are comparable to the initial RT_{NDT} values for the materials assessed in Table 4-4. The adjustment to RT_{NDT} will, therefore, also be smaller than for the materials surrounding the active core. Hence, the ferritic plate and weld materials located above and below the region immediately surrounding the active core in the SONGS Units will never be limiting with respect to establishing the heat-up and cool-down limits. (Note that these additional materials comprise the upper shell course plates and welds that are located immediately above the intermediate shell course plates and welds. Each of the lower and intermediate shell course plates and welds are assessed in Table 4-4.)

c. The initial RT_{NDT} values for the ferritic plate and weld materials located above and below the region immediately surrounding the active core are used to establish other aspects of the heat-up and cool-down limits. These other limits include the bolt-up temperature, the lowest service temperature, and the flange limits.

19) *Identify which method (i.e., K_{IC} or K_{IA}) will be used to calculate the reference intensity factor (K_{IR}) values for the RPV as a function of temperature.*

The reference stress intensity factor (SIF) used in both Units 2 and 3 is the equation for K_{IC} given in Appendix G of the ASME Code (Reference 3). It is expressed as:

$$K_{Ic} = 33.2 + 20.734 e^{0.02(T - RT_{NDT})}$$

The parameter T is the temperature of the material at the hypothetical crack tip (°F), RT_{NDT} is the material nil-ductility transition reference temperature (°F), and K_{IC} is the crack initiation fracture toughness (ksi√in).

20) *If Code Case N-640 and K_{IC} are used as the basis for calculating the K_{IR} values, submit an exemption request to use the methods of Code Case N-640 and apply them to the P-T limit calculations.*

The determination of the Pressure-Temperature (P-T) curves is consistent with Section XI of the 2000 Edition of the ASME Code. This Code Edition has incorporated the K_{IC} criteria for the allowable fracture toughness. The Code of Federal Regulations part 50.55a has

approved this ASME Code Edition in Reference 8. Therefore, Code Case N-640 is not used and an exemption request is not submitted.

21) *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.*

a, c

The justification to support an exemption request for use of the CE NSSS methodology of pressure stress K_{IM} is included as Appendix A of this document.

22) *Include in the PTLRs the P-T curves for heatup, cooldown, criticality, and hydrostatic and leak tests of the reactors.*

P-T curves for heatup, cooldown, hydrostatic test and criticality have been included in the Westinghouse Calculations of References 6 and 7. Details of heatup, cooldown and hydrostatic test curves for the uncorrected case are included as Figures 6 through 27 of this document. Allowable P-T limit curves for the criticality condition were obtained from References 6 and 7 and are shown in Figures 28 through 33 of this document.

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.*

Minimum temperature requirements have been incorporated into the final composite P-T curve limits in Figures 5.1 through 5.3 of the PLTR reports (References 1 and 2). These figures demonstrate that the P-T curves comply with Appendix G to 10 CFR 50.

Please refer to RAI #5 of this document for a further discussion of how compliance is demonstrated.

24) *With respect to the evaluation of plant specific surveillance data, licensees need to include in their PTLRs the supplemental surveillance data and calculations of the chemistry factors if the surveillance data are used for the calculations of the adjusted reference temperatures.*

a. For Unit 2, the determination of chemistry factors is discussed in Section 4 of Reference 1 and the surveillance data are discussed in Section 7. The chemistry factors were determined in accordance with Regulatory Guide 1.99, Revision 2, using Regulatory Position 1.1. The bases for the chemical content of the vessel and surveillance plates and welds are provided in the responses to Generic Letter 92-01 (Reference 16 in Reference 1). Additional

information is also provided in the responses to RAIs #8 and #9. Surveillance data were evaluated but they are not used for the calculations of the adjusted reference temperatures.

b. For Unit 3, the determination of chemistry factors is discussed in Section 4 of Reference 2 and the surveillance data are discussed in Section 7. The chemistry factors were determined in accordance with Regulatory Guide 1.99, Revision 2, using both Regulatory Position 1.1 and 2.1. Additional information is also provided in the response to RAI #9. No supplemental surveillance data were used.

25) *Provide the evaluation of whether the surveillance data are credible in accordance with the credibility criteria of RG 1.99, Revision 02.*

a. For Unit 2, the credibility of the surveillance data is established in Section 7 of Reference 1. Additional information is also provided in the responses to RAIs #8 and #9.

b. For Unit 3, the credibility of the surveillance data is established in Section 7 of Reference 2. Additional information is also provided in the response to RAI #9.

26) *In addition, if licensees seek to use surveillance data from supplemental plant sources, licensees must identify the source of the data, and either identify by title and number the safety evaluation report ....; or compare the licensee's data ... and submit the proposed integrated surveillance program and evaluation of the data for the NRC for review and approval.*

The licensee does not seek to use surveillance data from supplemental plant sources for either Units 2 or 3. In either case there is a plant-specific surveillance program. Four surveillance capsules remain in each unit, and the materials in the surveillance capsules were selected in accordance with ASTM E185-73. In neither case was it necessary or feasible to obtain surveillance data from another reactor vessel surveillance program.

3.2 RAI #2

*The ASME Code, Section XI, Appendix G provides a methodology for calculating stress intensity factors corresponding to membrane tension (K_{IM}) and thermal stress (K_{IT}) for the postulated axial defect. Calculations of K_{IT} are based on stress influence coefficients from finite element modeling (FEM) analyses for inside (¼T) and outside (¾T) surface flaws. Calculations of the maximum allowable K_{IM} are based on a closed-form solution to an equation such as $2K_{IM} + K_{IT} < K_{IC}$, where K_{IT} has been determined from solutions based on stress influence coefficients, and K_{IC} was determined using the equation representing the analytical approximation to the lower bound fracture toughness curve, K_{IC} (in ksi*sqrt(in.)) = $33.2 + 20.734\exp[0.02(T - RT_{NDT})]$, where RT_{NDT} is the material nil- ductility transition reference temperature and T is the actual temperature of the material.*

The Combustion Engineering (CE) nuclear steam supply system (NSSS) methodology differs from the ASME Code, Section XI, Appendix G methodology in several respects. The CE NSSS methodology for calculating K_{IT} is based on thermal influence coefficients from FEM analyses, as opposed to stress influence coefficients. Furthermore, the CE NSSS methodology for calculating K_{IM} does not involve a closed-form solution based on calculations of K_{IT} and K_{IC} factors, and instead applies FEM methods for estimating the K_{IM} factors.

Please supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a discussion of the specific methodologies that will be applied in the PTLRs for SONGS 2 and 3 for calculating stress intensity factors at the 1/4T and 3/4T crack depth locations:

a. *Discuss the methodology for calculating the thermal stress intensity factor, K_{IT}.*
b. *Discuss the methodology for calculating the stress intensity factor corresponding to membrane tension resulting from pressure loading of the reactor vessel, K_{IM}. Please specify whether K_{IM} is determined by obtaining a closed-form solution (as prescribed by the ASME Code, Section XI, Appendix G) or determined by applying FEM methods (as prescribed by the CE NSSS methodology).*

Per your response to action item 21 in RAI 1, if your methodology applies the CE NSSS method for calculating K_{IM} stress intensity values, then your application will need to include a request for an exemption from the requirements of 10 CFR Part 50, Appendix G for P-T limits. The need for an exemption for calculating P-T limits using the CE NSSS method is specified in the fourth paragraph (pages 20- 21) of Section 2.5.4 and in action item 21 (page 27) of Section 5.0 of the SE on topical report CE-NPSD-683, Revision 6, dated March 16, 2001. The CE Owners Group (CEOG) agreed to this requirement in their final version of topical report CE-NPSD-683, Revision 6. The requirement for the exemption is specified in the "Note" on page 5-15 of the topical report.

Response:

Calculations of crack tip stress intensity factors due to thermal loads, K_{IT}, are based on thermal influence coefficients developed using two dimensional (2D) finite element analyses with linear, quadratic and cubic temperature profiles across the vessel wall thickness. The influence coefficients are computed for postulated inside (¼ thickness) and outside (¾ thickness) surface flaws and are then corrected to account for the 3D elliptical geometry using the procedures of Appendix A of Section XI of the ASME Code.

a, c

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a, c

Using the K_{IM} values due to the applied pressure loading, the ASME Code procedure using the criteria:

$$2K_{IM} + K_{IT} \leq K_{IC}$$

was followed in computing the maximum allowable for thermal transient cases, and

$$1.5K_{IM} \leq K_{IC}$$

for hydrostatic test cases.

Per the conditions listed in the safety evaluation for topical report CE NPSD-683 (Reference 4) a request for exemption from the requirements of 10CFR Part 50, Appendix G, is provided in Appendix A of this document.

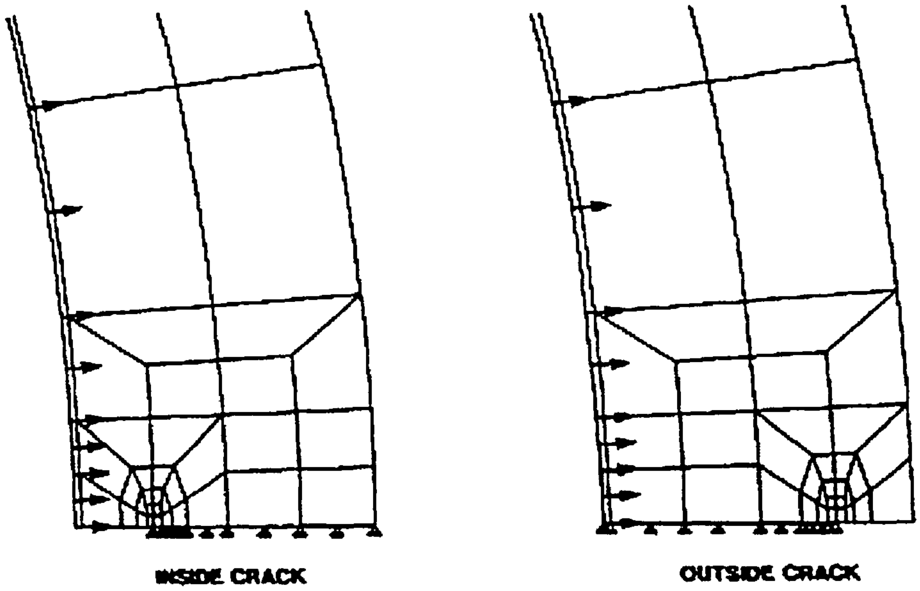

Figure 1: FE Models used in Computing K_{IM} due to Internal Pressure Loading

3.3 RAI #3

In support of the NRC staff's review of the P-T limit curves contained in the PTLR submittal, please supplement your application with data for the through-wall thermal gradients (ΔT) and thermal stress intensities (K_{IT}) for the 1/4T and 3/4T crack depth locations. These data are necessary for the NRC staff to perform independent calculations of P-T limits to verify that the P-T limit curves are at least as conservative as those that would be obtained as a result of applying the methods of 10 CFR Part 50, Appendix G, or as modified using the CE NSSS methodology. In addition, if you are requesting to use the CE NSSS methodology for K_{IM} determinations, please submit the plant-specific K_{IM} data to support the staff's review of these calculations.

Response:

a, c

Figure 2: Unit 2 Heatup 60F/hr Through-wall Thermal Gradients at Crack Tips

a, c

Figure 3: Unit 2 Heatup Thermal Stress Intensity Factors K_{IT} at Outside Crack Tip

a, c

Figure 4: Unit 2 Cooldown 100F/hr Through-wall Thermal Gradients at Crack Tips

a, c

Figure 5: Unit 2 Cooldown Thermal SIF K_{IT} at Inside Crack Tip

The membrane stress intensity factor coefficients K_{IM} due to unit (1000 psi) internal pressure loading for SONGS reactor vessel geometry with a base metal thickness of 8.625 inches for inside and outside crack tip locations are given in the following table.

Table 1: Pressure Stress Influence Coefficients, K_{IM}

a, c

Per the topical report (Reference 4) a request for exemption from the requirements of 10CFR Part 50, Appendix G, is provided in Appendix A of this document.

3.4 RAI #4

In all cases P-T limit curves must be determined using the most limiting conditions in the reactor vessel. For heatup and cooldown transients the application of the PTLR methodology and calculations of P-T limits must always take into consideration the different conditions at the 1/4T and 3/4T locations during the thermal transient, and the resulting P-T limit curves must always represent the most limiting of these conditions.

Please supplement Section 5.0 of the SONGS 2 and 3 PTLRs with a discussion of how the P-T limit curves account for the most limiting conditions in the reactor vessel. The discussion should address the following points:

a. Please discuss how the calculation of P-T limit curves for SONGS 2 and 3 addresses heatup and cooldown transients, specifically taking into consideration the different conditions at the 1/4T and 3/4T crack depth locations.

b. Please discuss how the calculation of P-T limit curves for SONGS 2 and 3 addresses the assessment of the 1/4T location for steady state conditions in addition to the 1/4T and 3/4T locations under heatup and cooldown transient conditions. Please supplement the P-T limit curves for SONGS 2 and 3 with a P-T limit curve representing the 1/4T location under steady state conditions.

Response: a, c

a, c

a, c

Figure 6: Unit 2 Steady-state P-T Curve

a, c

Figure 7: Unit 2 Heatup P-T Curves

a, c

Figure 8: Unit 2 Heatup P-T Curve Enveloped for 5F/hr Transient

a, c

Figure 9: Unit 2 Heatup P-T Curve Enveloped for 10F/hr Transient

a, c

Figure 10: Unit 2 Heatup P-T Curve Enveloped for 30F/hr Transient

a, c

Figure 11: Unit 2 Heatup P-T Curve Enveloped for 40F/hr Transient

a, c

Figure 12: Unit 2 Heatup P-T Curve Enveloped for 60F/hr Transient

a, c

Figure 13: Unit 2 Heatup P-T Curve Envelopes

a, c

Figure 14: Unit 2 Cooldown P-T Curves

a, c

Figure 15: Unit 2 Cooldown P-T Curve Enveloped for 100F/hr Transient

a, c

Figure 16: Unit 2 Cooldown P-T Curve Envelopes

a, c

Figure 17: Unit 3 Steady-state P-T Curve

a, c

Figure 18: Unit 3 Heatup P-T Curves

a, c

Figure 19: Unit 3 Heatup P-T Curve Enveloped for 5F/hr Transient

a, c

Figure 20: Unit 3 Heatup P-T Curve Enveloped for 10F/hr Transient

a, c

Figure 21: Unit 3 Heatup P-T Curve Enveloped for 30F/hr Transient

a, c

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Figure 22: Unit 3 Heatup P-T Curve Enveloped for 40F/hr Transient

a, c

[ ]

Figure 23: Unit 3 Heatup P-T Curve Enveloped 60F/hr Transient

a, c

Figure 24: Unit 3 Heatup P-T Curve Envelopes

a, c

Figure 25: Unit 3 Cooldown P-T Curves

a, c

Figure 26: Unit 3 Cooldown P-T Curve Enveloped for 100F/hr

a, c

Figure 27: Unit 3 Cooldown P-T Curve Envelopes

3.5 RAI #5

Table 1 of 10 CFR Part 50, Appendix G, specifies six different minimum temperature requirements that must be met when generating the pressure-temperature (P-T) limits for U.S. operating pressurized water reactors (PWRs):

a. *Those for pressure test conditions with the Reactor Coolant System (RCS) pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure.*
b. *Those for pressure test conditions with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure.*
c. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure, at times the reactor is not in the critical operating mode.*
d. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure at times the reactor is not in the critical operating mode.*
e. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure at times the reactor is in the critical operating mode.*
f. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure at times the reactor is in the critical operating mode.*

Criterion 6 in Attachment 1 to GL 96-03 states that the above minimum temperature requirements of 10 CFR Part 50, Appendix G shall be incorporated into the P-T limit curves, and PTLRs shall identify minimum temperatures on the P-T limit curves such as the minimum boltup temperature and the hydrotest temperature.

Section 6.0 of the SONGS 2 and 3 PTLRs, provides a listing and brief discussion of the minimum temperature requirements that have been incorporated into the P-T limit curves for SONGS 2 and 3. However, the discussion does not adequately demonstrate how the P-T limit curves for pressure testing conditions and normal operations with the core critical and core not critical will be in compliance with the appropriate minimum temperature requirements as given in Table 1 to Appendix G to 10 CFR Part 50. This information is needed to satisfy action item 23 from staff's safety evaluation (SE) on topical report CE-NPSD-683, Revision 6.

Per your response to action item 23 in RAI 1, update Section 6.0 of the PTLRs for SONGS 2 and 3 to provide a discussion on how the P-T limit curves will meet all of the minimum temperature requirements mandated by Table 1 of 10 CFR Part 50, Appendix G. Include in this discussion the value for the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload and how this value is applied along with minimum permissible hydrostatic test temperature to determine minimum temperature requirements that will be applied to the P-T limit curves for SONGS 2 and 3. This information is necessary to ensure that the SONGS 2 and 3 P-T limit curves will continue to comply with the minimum temperature requirements of Table 1 of 10 CFR Part 50, Appendix G, and that the PTLR will conform to the provisions of Criterion 6 in Attachment 1 to Generic Letter (GL) 96-03.

Response:

The P-T limit curves for pressure testing conditions and normal operations with the core critical and core not critical are in compliance with the appropriate minimum temperature requirements given in Table 1 of Appendix G to 10 CFR Part 50. This is demonstrated by the explanation and figures provided hereafter. Each of the six minimum temperature requirements a) through f) are specifically identified for clarity.

Unit 2

Design pressure	= 2,500 psia (2,485.3 psig)	
Normal operating pressure	= 2,250 psia (2,235.3 psig)	
Preservice hydrostatic pressure	= 3,125 psia (3,110.3 psig)	
Minimum bolt-up temperature	= 65°F	(Ref.12)
Flange region RT_{NDT}	= 20°F	(Ref.11)
Initial piping, pumps and valves RT_{NDT}	= 90°F	(Ref.13)
Adjusted RT_{NDT} at ¼ t for 32 EFPY	= 137.3°F	
Adjusted RT_{NDT} at ¾ t for 32 EFPY	= 116.6°F	
20% Preservice hydrostatic pressure	= 0.2 (3,125 psia) = 625 psia	

Preservice hydrostatic pressure with correction for instrument uncertainty
= 20% preservice hydro pressure + RCS instrument uncertainty
= 625 psia - 97.8 psi = 527.2 psia

Inservice hydrostatic pressure= 1.1 (Operating Pressure) + Pressurizer instrument uncertainty
= 1.1 (2,250 psia) + 81 psi = 2,556 psia

a, c

a, c

Figure 28 shows the minimum pressure and temperature requirements for hydrostatic test and heatup transients for Unit 2. Minimum requirements for cooldown transients for control room and remote shutdown panel are shown in Figures 29 and 30, respectively.

Unit 3

Design Pressure	= 2,500 psia (2,485.3 psig)	
Normal Operating Pressure	= 2,250 psia (2,235.3 psig)	
Preservice Hydrostatic Pressure	= 3,125 psia (3,110.3 psig)	
Minimum Bolt-up Temperature	= 65°F	(Ref.12)
Flange Initial RT_{NDT}	= 40°F	(Ref.15)
Initial piping, pumps and valves RT_{NDT}	= 90°F	(Ref.13 & 16)
Adjusted RT_{NDT} at ¼ t for 32 EFPY	= 145.8°F	
Adjusted RT_{NDT} at ¾ t for 32 EFPY	= 125.5°F	

20% Preservice hydrostatic pressure = 0.2 (3,125 psia) = 625 psia

Preservice hydrostatic pressure with correction for instrument uncertainty
= 20% preservice hydro pressure + RCS instrument uncertainty
= 625 psia - 97.8 psi = 527.2 psia

Inservice hydrostatic pressure= 1.1 (Operating Pressure) + Pressurizer instrument uncertainty
= 1.1 (2,250 psia) + 81 psi = 2,556 psia

a, c

a, c

Figure 31 shows the minimum pressure and temperature requirements for hydrostatic test and heatup transients for Unit 3. Minimum requirements for cooldown transients for control room and remote shutdown panel are shown in Figures 32 and 33, respectively.

a, c

Figure 28: Unit 2 Heatup P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 29: Unit 2 Cooldown P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 30: Unit 2 Cooldown P-T Curves with Min. Temperature Requirements, Remote Shutdown Panel

a, c

Figure 31: Unit 3 Heatup P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 32: Unit 3 Cooldown P-T Curves with Min. Temperature Requirements, Control Room

a, c

Figure 33: Unit 3 Cooldown P-T Curves with Min. Temperature Requirements, Remote Shutdown Panel

3.6 RAI #6

Section 5.0 of the PTLRs for SONGS 2 and 3 provides a footnote indicating that pressure and temperature limit values are adjusted for instrument uncertainty, and for RCS pressure and elevation effects. Please 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. Include in this discussion numerical values for the instrument uncertainties as well as numerical values for factors that compensate for RCS pressure and elevation effects. Please discuss how these factors are applied in the calculation of the P-T limit curves.

Response:

Tables 5-1, 5-2 and 5-3 of the SONGS PTLRs (References 1 and 2) each have a footnote explaining that the tabulated values have been adjusted for instrument uncertainty and for RCS pressure and elevation effects. Additional detail as below will be added to Section 5 of References 1 and 2 to explain these footnotes.

The calculated reactor vessel pressure and temperature limit values are adjusted for instrument uncertainty, and for RCS pressure and elevation effects. Section 3.4.2 of the topical report (Reference 4) provides a description of the development of these RCS pressure and elevation effects.

These adjustments ensure that the analytical beltline P-T limits are conservatively interpreted by pressurizer pressure and RCS temperature instrumentation. The pressure values are adjusted using pressure correction factors (PCF), and the temperature values are adjusted for temperature instrumentation uncertainty.

The pressure correction factors applied to Tables 5-1, 5-2 and 5-3 consist of three components:

1. pressure differential corresponding to water head between the pressurizer water level and the reference point in the reactor vessel (ΔP_{ELEV}),
2. flow-induced pressure drop between the reactor vessel downcomer and the surge nozzle in the hot leg (ΔP_{FLOW}; a value that depends on the number of operating RCPs), and
3. pressurizer pressure instrumentation loop uncertainty (ΔP_{INSTR}).

These components are individually established using conservative assumptions, then summed into the PCF. The PCF values are subtracted from the analytical values to conservatively reduce the allowable pressure limit. The explicit PCF values used for the SONGS units are dependent upon the number of operating RCPs and on the available instrumentation. Thus, the following PCFs are applied to the analytical pressure limits:

Pressure Correction Factors
(with instrument uncertainty)

		Low Range	Wide Range	
Control Room:		≤ 700 psia		
≤ 340 °F (2 RCP Operating)	PCF =	97.8	151.3	psid
> 340 °F (3 RCP Operating)	PCF =	117.8	171.3	psid
Remote Shutdown Panel:		≤ 1600 psia		
≤ 340 °F (2 RCP Operating)	PCF =	146.3	151.3	psid
> 340 °F (3 RCP Operating)	PCF =	166.3	171.3	psid

The data in Tables 5-1, 5-2 and 5-3 of References 1 and 2 are also adjusted for temperature instrumentation uncertainty. For SONGS, a conservative temperature uncertainty of 18.5°F is added to the analytical values for both the control room and the remote shutdown panel instrumentation.

3.7 RAI #7

The proposed P-T limit curves included in Section 5.0 of the PTLRs for SONGS 2 and 3 are proposed to be effective through 32 effective full power years of operation (EFPY). The existing P-T limit curves contained in the Technical Specifications (TS) are stated to be effective through 20 EFPY. Confirm whether the changes to the P-T limit curves included in Section 5.0 of the PTLRs for SONGS 2 and 3 reflect only the increase in the EFPY for which the curves will be applied. If there are other factors, such as different parameters or methods, which contribute to the changes to the curves, provide a detailed discussion of these factors and how they affect the PTLR P-T limit curves.

Response:

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, via a higher RT_{NDT} shift from 20 to 32 EFPY, in conjunction with the use of the crack arrest allowable, K_{IC}, instead of the crack initiation allowable, K_{IA}. The P-T limit curves included in Section 5.0 of the PTLRs are effective through 32 effective full power years of operation and are based on a 32 year RT_{NDT} shift using the K_{IC} allowable. The P-T limit curves contained in the Technical Specifications (TS) are effective through 20 EFPY and are based on a RT_{NDT} shift of 20 years using the K_{IA} allowable.

3.8 RAI #8

Criterion 7 of the Table in Attachment 1 to GL 96-03 specifies that an analysis of the credibility of the surveillance data must be provided in the PTLR. Regulatory Position 2.1 of Regulatory Guide (RG) 1.99, Revision 2 specifies that when two or more credible surveillance data sets become available from the reactor in question, they may be used to determine the Adjusted Reference Temperature (ART) values. If the procedure of Regulatory Position 2.1 for determining the ART values based on the surveillance data results in a higher value for the ART than that given by using the procedures of Regulatory Position 1.1 of the RG, RG 1.99, Revision 2 specifies that the surveillance data should be used for the ART and chemistry factor determination. If the procedure of Regulatory Position 2.1 results in a lower value for the ART, either may be used.

Please confirm that the credibility analysis of the SONGS 2 surveillance data from Section 7.0 of the SONGS 2 PTLR demonstrated that the surveillance data sets for SONGS 2 are credible.

Section 7.0 of the SONGS 2 PTLR states that the surveillance data were not used to generate a chemistry factor in accordance with the methodology prescribed in Regulatory Position 2.1 of RG 1.99, Revision 2. Please confirm whether the ART values for the limiting materials were calculated using the procedure of Regulatory Position 1.1 of RG 1.99, Revision 2.

If the procedure of Regulatory Position 1.1 of RG 1.99, Revision 2 was used to calculate the ART values for the limiting materials, please indicate why this is an acceptable procedure, given the credibility of the surveillance data.

Please supplement Section 7.0 of the PTLR for SONGS 2 with the following information:

a. *Table 7-1 of the SONGS 2 PTLR provides chemistry factors for the two surveillance materials plate C-6404-2 and weld 9-203. Please indicate how these chemistry factors were derived.*
b. *There is no explicit calculation in the SONGS 2 PTLR demonstrating that chemistry factor values for the limiting materials derived from the tables in RG 1.99, Revision 2 would result in limiting ART values that are more conservative than those determined using chemistry factors derived from surveillance data. Per your response to action item 24 in RAI 1 please supplement Section 7.0 of the PTLR for SONGS 2 with detailed calculations of the chemistry factors for each of the surveillance materials based on the calculation methods specified in Regulatory Position 2.1 of RG 1.99, Revision 2.*

The calculations of the chemistry factors for the surveillance materials for SONGS 3, provided in Table 7-1 of the SONGS 3 PTLR represent an acceptable format for presenting surveillance material chemistry factor calculations.

Response:

It is confirmed that a credibility analysis was performed in accordance with Regulatory Guide 1.99, Revision 2, and it was determined that all of the credibility criteria were met for the SONGS 2 surveillance base plate (Plate C-6404-2) and surveillance weld (Weld 9-203). It is also confirmed that ART values for all the SONGS 2 reactor vessel beltline materials were calculated using Regulatory Position 1.1 of RG 1.99, Revision 2. The results of that calculation are reported in Table 4-4 of Reference 1. Additional details are provided in the specific responses to questions 8.a and 8.b.

Information on the selection of materials for inclusion in the Unit 2 surveillance program is provided in Reference 17. The selection followed the procedures of ASTM E-185-73, Annex A-1. Two surveillance capsules have been removed from Unit 2, those from the 97° and 263° locations. Data from those capsules were used to compute an Adjusted Reference Temperature (ART) using Regulatory Guide 1.99, Revision 2, Regulatory Position 2.1. The ART values for plate C-6404-2 and the surveillance weld determined from the surveillance data were less than that predicted for plate C-6404-5 using Position 1.1 of RG 1.99. That is, the two surveillance data sets do not represent the limiting vessel material. The responses to questions 8.a and 8.b below provide additional information regarding the treatment of the surveillance data.

The heatup and cooldown limits and the assessment of RT_{PTS} require the determination of the highest (i.e., limiting) value of ART. These values are reported in Section 4 of Reference 1. The limiting material (highest ART value at 1/4T or 3/4T based on Regulatory Position 1.1 of RG 1.99, Revision 2) for the SONGS 2 reactor vessel beltline is plate C-6404-5. The ART for plate C-6404-2 based on the credible surveillance data and Regulatory Position 2.1 is lower than that for the limiting plate. The ART for the surveillance weld based on the credible surveillance data and Regulatory Position 2.1 is also lower than that for the limiting plate. The predicted values of RT_{PTS} are in the same relative order as the ART values. Hence, the procedure used produces a more conservative value for the SONGS 2 reactor vessel than that obtained using the credible surveillance data and Regulatory Position 2.1. New surveillance data will be reviewed in accordance with the requirements in effect.

a. Response to RAI #8.a

The chemistry factors for the surveillance materials provided in Table 7-1 of Reference 1 are obtained from Tables 1 and 2 of RG 1.99, Revision 2, based on the following copper and nickel content:

CF = 65°F for Plate C-6404-2, based on 0.10% copper content and 0.60% nickel content per Reference 18. Also, refer to Table 2 in the response to 8.b, below.

CF = 31.1°F for Weld 9-203, based on 0.03% copper content and 0.14% nickel content. These represent the average values of the 97° and the 263° surveillance capsules per Reference 20. Baseline analysis showed 0.03% copper content and 0.12% nickel content (Reference 20), which corresponds to a slightly lower CF = 29.8°F.

b. Response to RAI #8.b

Two analyses were performed by Combustion Engineering Chattanooga Laboratory on Unit 2 reactor vessel beltline plates and welds (Reference 18), including intermediate shell Plate C-6404-2 and lower shell Plate C-6404-5. The copper and nickel content based on these analyses results are provided in Table 2.

Table 2: Copper and Nickel Content by Weight (percent)

Plate	First Analysis		Second Analysis		Average	
	Cu	Ni	Cu	Ni	Cu	Ni
C-6404-2	0.10	0.58	0.10	0.60	0.10	0.59
C-6404-5	0.11	0.62	0.11	0.67	0.11	0.65

Per Table 2 of Regulatory Guide 1.99 (Reference 19), the chemistry factor for Plate C-6404-2 is 65°F, and the chemistry factor for C-6404-5 is 75°F based on the average Cu and Ni values provided in Table 2 above.

b.1 ART for Intermediate Shell Plate C-6404-2

The following two surveillance data sets are available for intermediate shell Plate C- 6404-2:

i) The 97° surveillance capsule, which was removed from the reactor vessel during the Cycle 4 refueling outage, i.e., at the end of Fuel Cycle 3, and

ii) The 263° surveillance capsule, which was removed from the reactor vessel during the Cycle 11 refueling outage, i.e., at the end of Fuel Cycle 10.

The SONGS Unit 2 surveillance data were evaluated to show that the base plate and the weld met the five credibility criteria provided in RG 1.99, Revision 2. The Adjusted Reference Temperature (ART) can be calculated for intermediate shell Plate C-6404-2 and Weld 9-203 based on the two credible surveillance data sets as follows:

- Per Reference 20, the fluence values are 0.507 for the 97° surveillance capsule, and 2.188 for the 263° surveillance capsule (units:10^{19} n/cm^2). The corresponding fluence factor ($ff = f^{(0.28 - 0.1 \log f)}$) values are calculated per Position 1.1. It follows that:

  ff = 0.810 based on the 97° surveillance capsule, and
  ff = 1.212 based on the 263° surveillance capsule

  The sum of the squares, $\Sigma\, ff^2$, is given by:

  $\Sigma\, ff^2 = 0.81^2 + 1.212^2 = 2.127$

- Per Reference 8.4, the measured ΔRT_{NDT} values are:

  $\Delta RT_{NDT} = 41°F$ for the 97° surveillance capsule
  $\Delta RT_{NDT} = 88°F$ for the 263° surveillance capsule

  It follows that:

  $\Sigma\,(ff \times \Delta RT_{NDT}) = 0.810 \times 41 + 1.212 \times 88 = 139.9°F$

- Per Position 2.1, the chemistry factor, CF, is given by:

  $CF = \Sigma\,(ff \times \Delta RT_{NDT}) / \Sigma\, ff^2 = 65.8°F$

- The values of f, ff and ΔRT_{NDT} projected for 32 EFPY are calculated using Position 1.1 methodology with x = 2.375 inches at the 1/4 T location:

  $f = 2.4436(\times 10^{19}$ n/cm$^2)$
  $ff = 1.2405$
  $\Delta RT_{NDT} = CF \times ff = 81.6°F$

- The projected Adjusted Reference Temperature (ART) for Plate C-6404-2 for 32 EFPY at the 1/4 T location is calculated based on Position 1.1 as follows:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

where

Initial RT_{NDT} = 20°F per Reference 18
Margin = 17°F per Position 2.1

Therefore,

ART= 20 + 81.6 + 17 = **118.6°F**

In the above analysis, the fluence value for the intermediate shell plates was used.

b.2 ART for Lower Shell Plate C-6404-5

- Per Table 2 of Reference 19, CF = 75°F for lower shell Plate C-6404-5.
- Per Reference 8.2, the projected surface fluence for 32 EFPY = 4.3707 ($x10^{19}$ n/cm^2).
- Using Position 1.1 calculation methodology, the projected fluence for 32 EFPY at the 1/4T location for lower shell plates is:

  f = 2.4717 ($x10^{19}$ n/cm^2)
  ff = 1.2434

- The projected shift, ΔRT_{NDT}, is given by:

  ΔRT_{NDT} = CF x ff = 93.3°F
  Initial RT_{NDT} = 10°F (Reference 18)
  Margin = 34°F (per Position 1.1)

- It follows that ART for Plate C-6404-5 projected for 32 EFPY at the 1/4 T location is given by:

  ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin
  = 10 + 93.3 + 34 = **137.3°F**

In the calculation above, the fluence values for the lower shell plates were used.

Based on the analyses results in Sections b.1 and b.2 above, the ART for Plate C-6404-5 (137.3°F) is higher then that for Plate C-6404-2 (118.6°F), so Plate C-6404-5 is bounding.

b.3 ART for Lower Shell Weld 9-203

- Per Reference 18, the results of the chemical analysis performed by Combustion Engineering on Weld 9-203 showed an as-deposited copper content of 0.07% and nickel content of 0.29% by weight. The chemistry factor CF = 69 per Table 1 of Reference 19.

- Using the values of f = 2.4717 ($x10^{19}$ n/cm^2), and ff = 1.2434 calculated for the lower shell at the 1/4T location in Section b.2 above, ART for Weld 9-203 can be calculated as follows:

  ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin

where

Initial RT_{NDT} = -60°F per Reference 18
Margin = 56°F per Position 1.1
ΔRT_{NDT} = CF x ff = 85.8°F

Therefore,

ART= -60 + 85.8 + 56 = **81.8°F**

- Alternatively, two sets of surveillance data exist for Weld 9-203. Using Position 2.1, Weld 9-203 ART of 8.4°F is calculated in the response to RAI #9.

Therefore, the value of CF = 69°F calculated based on Position 1.1 is higher than that based on Position 2.1 using the surveillance data, CF = 32.5. Hence, the bounding estimate of ART for Weld 9-203 is 81.8°F. Note that the ART value of 137.3°F for Plate C-6404-5 is actually bounding and was used to generate the P-T curves.

New surveillance data will be reviewed in accordance with the requirements in effect.

3.9 RAI #9

Regulatory Position 2.1 of RG 1.99, Revision 2 states that if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, the measured values of ΔRT should be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld.

Please indicate in the SONGS 2 and 3 PTLRs whether the copper and nickel content of the surveillance weld differs from that of the vessel weld. If so, please supplement Section 7.0 of the PTLRs for SONGS 2 and 3 with detailed calculations for determining the adjustments to the measured values for DRT for the surveillance weld, and indicate whether these adjusted values of DRT were used in the determination of the chemistry factor for the surveillance weld.

Response:

SONGS Unit 2

The copper and nickel content for the surveillance weld for Unit 2 differs from the as-deposited vessel weld, Weld 9-203, as shown in Table 3. Table 3 also provides the chemistry factor obtained using Table 2, RG 1.99, Revision 2.

Table 3: SONGS 2 Best Estimate Weld Chemical Composition

Weld	Cu %	Ni %	CF (°F)
Vessel Weld 9-203[(1)]	0.07	0.29	69
Surveillance Weld[(2)]	0.03	0.12	30
Surveillance Weld[(3)]	0.03	0.15	32

(1) Reference 18.
(2) Reference 21.
(3) Reference 18 provided these values based on analysis of the 97° capsule.

The value CF = 31.1°F was reported in Reference 20, and is considered reasonable based on the data in Table 3 above.

Table 4 provides the fluence values, the fluence factors and the measured ΔRT_{NDT} values for the 97° and the 263° surveillance capsules. The table also provides the adjusted ΔRT_{NDT} values obtained per Position 2.1 as follows:

Adjusted ΔRT_{NDT} = measured ΔRT_{NDT} values multiplied by the ratio of chemistry factors; where the ratio is the vessel weld chemistry factor divided by the surveillance weld chemistry factor. It follows that the ratio of chemistry factors = 69/31.1= 2.22.

Table 4: Adjusted Unit 2 Weld ART Based on Chemical Composition

	f(1) (10^{19} n/cm^2)	ff	Measured ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} (°F)
97° capsule	0.507	0.810	4.0	8.9
263° capsule	2.188	1.212	23.0	51.0

(2) The fluence values were obtained from Reference 20.

Using the two sets of data in Table 4, the value of ART for 32 EFPY is calculated as follows:

$\Sigma\, ff^2 = 2.127$
$\Sigma(ff \times \Delta RT_{NDT}) = 0.810 \times 8.9 + 1.212 \times 51.0 = 69.1$°F

Per Position 2.1, the chemistry factor, CF, is given by:

$$CF = \Sigma\,(ff \times \Delta RT_{NDT}) / \Sigma\, ff^2 = 32.5\text{°F}$$

Per Reference 20, the extrapolated fluence value (f) for the lower shell for 32 EFPY is 2.472 ($\times 10^{19}$ n/cm^2), and the corresponding fluence factor (ff) is 1.243. It follows that:

$$\Delta RT_{NDT} = 32.5 \times 1.243 = 40.4\text{°F}$$

Therefore, the projected ART value for 32 EFPY is given by:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$
$$= -60 + 40.4 + 28 = \mathbf{8.4°F}$$

where

Initial RT_{NDT} = -60°F per Reference 18
Margin = 28°F per Position 2.1

The projected ART value of 8.4°F calculated for Weld 9-203 for 32 EFPY is significantly lower than the limiting ART of 137.3°F for Plate C-6404-5, calculated as part of the response to RAI #8 above. The ART value of 137.3°F was used to generate the P-T curves.

SONGS Unit 3

The copper and nickel content for the surveillance weld for Unit 3, Weld 9-203, differs from the as-deposited vessel weld as shown in Table 5. Table 5 also provides the chemistry factor obtained using Table 2, RG 1.99, Revision 2.

For the purpose of this calculation, the vessel as-deposited weld CF = 33.6°F and surveillance weld CF = 27.2°F are conservatively used to maximize the adjustment to ΔRT_{NDT}.

Table 5: SONGS 3 Best Estimate Weld Chemical Composition

	Cu %	Ni %	CF (°F)
Vessel Weld 9-203 (1)	0.06	0.04	33.6
Vessel Weld 9-203 (1)	0.05	0.04	30.6
Surveillance Weld (1)	0.03	0.11	29.2
Surveillance Weld (1)	0.03	0.09	27.9
Surveillance Weld (2)	0.03	0.08	27.2

(1) Reference 18.
(2) Reference 22.

Examination of Table 5 shows that the available chemical analysis measurements for the surveillance weld and the vessel weld are essentially the same, and no adjustment to the values of ΔRT_{NDT} is warranted. Table 6 provides the fluence values, the fluence factors and the measured ΔRT_{NDT} values for the 97° and the 263° surveillance capsules.

Table 6: Unit 3 Surveillance Weld ΔRT_{NDT}

	f (1) (10^{19} n/cm^2)	ff	Measured ΔRT_{NDT} (°F)
97° capsule	0.8	0.937	29
263° capsule	2.471	1.243	72

(1) The fluence values were obtained from Reference 23.

Using the two sets of data in Table 6, the value of ART for 32 EFPY is calculated as follows:

$\Sigma\, ff^2 = 2.424$
$\Sigma\, (ff \times \Delta RT_{NDT}) = 0.937 \times 29 + 1.243 \times 72 = 116.7°F$

Per Position 2.1, the chemistry factor, CF, is given by:

$$CF = \frac{\Sigma\,(ff \times \Delta RT_{NDT})}{\Sigma\, ff^2} = 48.1°F$$

Per Reference 23, the extrapolated fluence value (f) for the lower shell for 32 EFPY is 2.37 ($\times 10^{19}$ n/cm^2), and the corresponding fluence factor (ff) is 1.233. It follows that:

$\Delta RT_{NDT} = 48.1 \times 1.233 = 59.3°F$

Therefore, the projected ART value for 32 EFPY is given by:

$$\begin{aligned} ART &= \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \\ &= -50 + 59.3 + 28 = \mathbf{37.3°F} \end{aligned}$$

where

Initial RT_{NDT} = -50°F — The higher of the two values reported in Reference 18

and

Margin= 28°F — Per Position 2.1.

The projected ART value of 37.3°F calculated for Weld 9-203 for 32 EFPY is significantly lower than the limiting ART of 145.8°F for Plate C-6802-1, which was used to generate the P-T curves for Unit 3. The projected ART value calculated using Position 1.1 is 33.3°F, which is similar to that calculated using Position 2.1.

4.0 REFERENCES

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5. ABB Report 063-PENG-ER-096, Rev. 00, "Technical Methodology Paper Comparing ABB/CE PT Curve to ASME Section III, Appendix G," January 22, 1998.
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8. Code of Federal Regulations 10 CFR Part 50.55a Codes and Standards, Section 1 Background, September 26, 2002.
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16. Westinghouse Correspondence S-MCM-130, "RCP Material Data for San Onofre III," August 1, 1980.
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18. Letter from Walter C. March, SCE, to U. S. Nuclear Regulatory Commission Dated June 22, 1994. Subject: Docket Nos. 50-361 and 50-362, Revision to Supplemental Response to Generic Letter 92-01, Revision 1,"Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" San Onofre Nuclear Generating Station Units 2 and 3.

19. Regulatory Guide 1.99, Revision 2.

20. SCE Report SO23-901-C264, Revision 1,"Analysis of the 263° Capsule, Southern California Edison Company, San Onofre Unit 2 Nuclear Generating Station."

21. CE Report S-TR-MCS-002 dated May 27, 1978 (SCE Document CDCC 56396), "Southern California Edison San Onofre Unit 2, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program."

22. CE Report TR-S-MCM-004 (SCE Document Control / Transmittal Number C791130G0097),"Southern California Edison San Onofre Unit 3, Evaluation of Baseline Specimens."

23. SCE Report SO23-901-C274, Rev. 1, "Analysis of the 263° Capsule, Southern California Edison Company, San Onofre Unit 3 Nuclear Generating Station."

Appendix A

Justification for Exemption to Apply Alternate Method of Calculating the Stress Intensity Factor K_{IM} due to Internal Pressure Loading for San Onofre Units 2 and 3

Introductory Statements:

Pursuant to 10CFR50.90, Southern California Edison (SCE) hereby requests the use of an alternate methodology for calculation of crack tip stress intensity factor K_{IM} for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 reactor vessel beltline regions subjected to internal pressure loading. This alternate methodology by Westinghouse has been used in topical report CE NPSD-683, Revision 6, dated March 16, 2001. This methodology was developed and used in 1998 for Indian Point Unit 3 in a previous submittal report (Docket No. 50-286, TAC No. M99928). The approach was reviewed by the NRC and concluded to be acceptable (Section 2.5.4 paragraph 4 of the report).

A justification for the application of this alternate method to SONGS is given below.

Justification for Exemption

10 CFR 50.60(b) allows usage of alternatives to the requirements described in Appendix G and H of 10 CFR 50 when the exemption is granted by the NRC.

In accordance with 10 CFR 50.12(a), Southern California Edison (SCE) requests an exemption from the regulations of 10CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation." The exemption request would allow SONGS Units 2 and 3 to use an alternate methodology for calculation of crack tip stress intensity factor K_{IM} for SONGS reactor vessel beltline regions subjected to internal pressure loading, in lieu of the methodology cited in ASME Boiler and Pressure Vessel Code, Appendix G.

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10CFR50 which are:

1) authorized by law;
2) will not present an undue risk to the public health and safety;
3) consistent with the common defense and security; and
4) special circumstances, as defined by 10 CFR 50.12(a)(2) are present.

The standards for the exemption are justified, as described below.

1) **The requested exemption is authorized by law.**

   The NRC is authorized by law to grant this exemption. Requirements in 10 CFR 50.60 state that the use of alternative methods to 10 CFR 50, Appendix G is acceptable when an exemption is granted by the NRC.

2) **The requested exemption does not present an undue risk to the public health and safety.**

The proposed exemption request has no impact on the safe operation of the plant. An exemption from the requirements would allow the use of an alternate methodology to calculate the membrane loading stress intensity factor. Specifically, this methodology uses a finite element base influence function under internal pressure loading. The results of this methodology are comparable to the results obtained using the ASME Appendix G methodology as demonstrated below.

ASME Code uses the procedure in Article G-2214 for axial surface flaws. In this procedure, stress intensity factor under internal pressure loading is given by (U.S. Customary units)

$$K_{IM} = M_M(\frac{pR_i}{t})$$

The magnification factor M_M for inside axial flaws is given by:

$$M_M = 1.85 \quad \text{for } \sqrt{t} < 2 \text{ or } t < 4$$

$$M_M = 0.926\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464 \text{ or } 4 \le t \le 12$$

$$M_M = 3.21 \text{ for } \sqrt{t} > 3.464 \text{ or } t > 12$$

and for outside flaws by:

$$M_M = 1.77 \quad \text{for } \sqrt{t} < 2 \text{ or } t < 4$$

$$M_M = 0.893\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464 \text{ or } 4 \le t \le 12$$

$$M_M = 3.09 \text{ for } \sqrt{t} > 3.464 \text{ or } t > 12$$

where:

K_{IM} is the stress intensity factor for membrane loads (ksi√in);
p is the internal pressure (ksi);
R_i is the vessel inner radius (in); and
t is the vessel wall thickness (in).

a, c

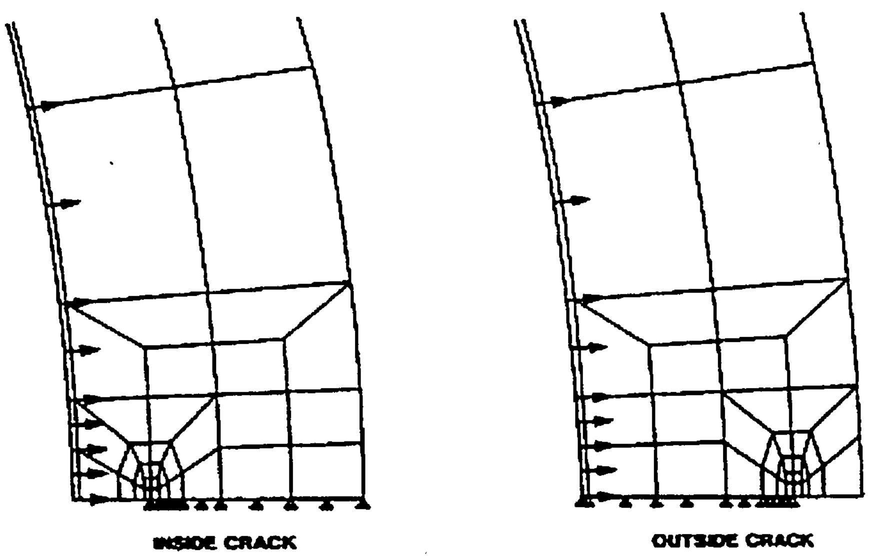
[ ]

$$k_{IM} = M_M(\frac{R_i}{t})$$

and the applied stress intensity factor K_{IM} for any given pressure, p, in ksi, is given by

$$K_{IM} = pk_{IM}$$

a, c



FEMs used in Computing k_{IM} due to Internal Pressure Loading

a, c

Therefore, this exemption request does not present an undue risk to the public health and safety.

3) **The requested exemption will not endanger the common defense and security.**

   The common defense and security are not affected by this exemption request.

4) **Special Circumstances, as defined by 10 CFR 50.12(a)(2) are present.**

   10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption unless special circumstances are present. This exemption meets the special circumstances listed in 10 CFR 50.12(a)(2)(ii).

   10 CFR 50.12(a)(2)(ii) – Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

   The primary purpose of 10 CFR 50.60 is to protect the reactor vessel against non-ductile failure. The use of the Westinghouse alternate methodology requested by this exemption provides greater operational flexibility while still maintaining reactor vessel integrity. In addition, use of the Westinghouse methodology to generate pressure-temperature curves yields comparable results to the use of the ASME Appendix G methodology. Therefore, the reactor vessel is protected against non-ductile failure and the underlying purpose of the rule is achieved.

Conclusion

The use of the Westinghouse alternate methodology to calculate the membrane stress intensity factor, K_{IM}, provides comparable results to that of the ASME Section XI, Division I, Appendix G, and provides adequate protection of the reactor vessel against non-ductile failure.

ENCLOSURE 3

Westinghouse authorization letter CAW-05-2078 to the U.S. Nuclear Regulatory Commission dated December 15, 2005 Affidavit, Proprietary Information Notice, and Copyright Notice.

Westinghouse

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Our ref: CAW-05-2078

December 15, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response to NRC Request for Additional Information on WCAP-16005-NP, Rev. 03, "San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report" and WCAP-16167-NP, Rev. 0, "San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report", dated December 14, 2005 (Proprietary/Non-Proprietary)

The proprietary information for which withholding is being requested in the above-referenced letter is further identified in Affidavit CAW-05-2078 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Southern California Edison Co.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2078, and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

for B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney
L. Feizollahi

CAW-05-2078

AFFIDAVIT

STATE OF CONNECTICUT:

ss:

COUNTY OF HARTFORD:

Before me, the undersigned authority, personally appeared Ian C. Rickard, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

I. C. Rickard,
Licensing Project Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 15th day
of December, 2005

Laurie J. White
Notary Public

My commission expires 8/31/09.

(1) I, Ian C. Rickard, dispose and say that I am a Licensing Project Manager, Westinghouse Electric Company LLC (Westinghouse), and as such I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(a) (v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Response to NRC Request for Additional Information on WCAP-16005-NP, Rev. 03, 'San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report' and WCAP-16167-NP, Rev. 0, San Onofre Nuclear Generating Station Unit 3 RCS Pressure and Temperature Limits Report' ", dated December 14, 2005 (Proprietary), being transmitted by the Southern California Edison Co. letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 enables Westinghouse to support utilities with NSSS plants in the preparation of pressure-temperature limit reports.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of its methodology from which the SONGS PTLR work is based.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar PTLR and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).