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# San Onofre Nuclear Generating Station Unit 2 RCS Pressure and Temperature Limits Report

Westinghouse

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# WCAP-16005-NP, Rev 00

# San Onofre
# Nuclear Generating Station Unit 2
# RCS Pressure and Temperature
# Limits Report

**March 2003**

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## TABLE OF CONTENTS

LIST OF TABLES ..... iii

LIST OF FIGURES ..... iii

LIST OF ACRONYMS ..... iv

ABSTRACT ..... v

INTRODUCTION ..... 1-1

1.0 NEUTRON FLUENCE VALUES ..... 1-1
- 1.1 INPUT DATA ..... 1-2
  - 1.1.1 Materials and Geometry ..... 1-2
  - 1.1.2 Cross Sections ..... 1-3
- 1.2 CORE NEUTRON SOURCE ..... 1-3
  - 1.2.1 Synthesized Three Dimensional Results ..... 1-3
  - 1.2.2 Calculated Activities and Measured Activities ..... 1-4
  - 1.2.3 Measurement / Calculation Ratios ..... 1-5
- 1.3. FLUENCE CALCLATION ..... 1-5
- 1.4 METHODOLOGY QUALIFICATION AND UNCERTAINTY ESTIMATES ..... 1-6
  - 1.4.1 Analytic Uncertainty Analysis ..... 1-7
  - 1.4.2 Comparison with Benchmark and Plant Specific Measurements ..... 1-8
  - 1.4.3 Overall Bias and Uncertainty ..... 1-8

2.0 REACTOR VESSEL SURVEILLANCE PROGRAM ..... 2-1
- 2.1 TEST MATERIAL SELECTION ..... 2-1
  - 2.1.1 Plate Material Selection ..... 2-2
  - 2.1.2 Weld Material Selection ..... 2-2
- 2.2 TEST SPECIMENS ..... 2-3
  - 2.2.1 Type and Quality ..... 2-3
  - 2.2.2 Unirradiated Specimens ..... 2-4
  - 2.2.3 Irradiated Specimens ..... 2-4
- 2.3 SPECIMEN IRRADIATION ..... 2-4
  - 2.3.1 Encapsulation of Specimens ..... 2-4
  - 2.3.2 Flux and Temperature Measurement ..... 2-6
  - 2.3.3 Irradiation Locations ..... 2-7
  - 2.3.4 Capsule Assembly Removal ..... 2-7

3.0 LTOP SYSTEM LIMITS ..... 3-1
- 3.1 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM ..... 3-1
- 3.2 BASIS FOR LTOP SYSTEM LIMITS ..... 3-1
  - 3.2.1 Peak Transient Pressures ..... 3-2
  - 3.2.2 Applicable P-T Limits ..... 3-5

## TABLE OF CONTENTS

4.0 BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURE ........ 4-1
4.1 BACKGROUND ........ 4-1

4.2 RESULTS ........ 4-1
4.2.1 Assumptions ........ 4-2
4.2.2 Calculation of Fluence Values for the EFPY of Interest ........ 4-2
4.2.3 Calculation of the Chemistry Factor Based on Surveillance Data ........ 4-2
4.2.4 Calculation of ART for the Limiting Plates at 1/4T and 3/4T ........ 4-3

4.3 ANALYSIS DETAILS ........ 4-3
4.3.1 Selection of Representative and Limiting Cases ........ 4-3
4.3.2 Fluence Calculation ........ 4-3

4.4 DETERMINATION OF THE CHEMISTRY FACTOR ........ 4-4

4.5 CALCULATION OF THE ADJUSTED REFERENCE TEMPERATURES AND SELECTION OF THE MAXIMA ........ 4-5

5.0 PRESSURE-TEMPERATURE LIMITS USING LIMITING ADJUSTED REFERENCE TEMPERATURE IN THE P-T CURVE CALCULATION ........ 5-1
5.1 RCS TEMPERATURE RATE-OF-CHANGE LIMITS ........ 5-1
5.2 RCS PRESSURE-TEMPERATURE LIMITS ........ 5-1

6.0 MINIMUM TEMPERATURE REQUIREMENTS IN THE PRESSURE-TEMPERATURE CURVES ........ 6-1

7.0 APPLICATION OF SURVEILLANCE DATA TO ADJUSTED REFERENCE TEMPERATURE CALCULATIONS ........ 7-1

8.0 REFERENCES ........ 8-1

APPENDIX A
TECHNICAL SPECIFICATION REFERENCES TO THE PTLR ........ A-1

## LIST OF TABLES


1-1 Bias Correction Factors ....... 1-4
1-2 Calculated Wetted Surface Neutron Fluence ....... 1-6
1-3 Fluence in $10^{19}$ n/cm$^2$ and Fluence Factors for 1/4T and 3/4T Locations ....... 1-6
1-4 Neutron Fluence Uncertainty ....... 1-9
2-1 Type and Quantity of Specimens for Irradiation Exposure and Irradiated Tests ....... 2-8
2-2 SONGS Unit 2 Capsule Assembly Removal Schedule ....... 2-8
3-1 Low Temperature RCS Overpressure Protection Range ....... 3-1
4-1 Calculated ART Values for 20 and 32 EFPY ....... 4-1
4-2 Calculated Wetted Surface Fluence ....... 4-4
4-3 Fluence and Fluence Factors for 1/4T and 3/4T Locations ....... 4-4
4-4 Calculation of Chemistry Factor Values for the Two Materials ....... 4-5
4-5 Credibility Test of the Calculated Chemistry Factors ....... 4-5
4-6 Calculated ART Values at 1/4T and 3/4T for 20 and 32 EFPY ....... 4-7
5-1 SONGS Unit 2 Heatup at 60°F/hr, RCS Pressure-Temperature Limits Valid until 32 EFPY ....... 5-2
5-2 SONGS Unit 2 Cooldown via Control Room Instrumentation, RCS Pressure-Temperature Limits Valid until 32 EFPY ....... 5-3
5-3 SONGS Unit 2 Cooldown via Remote Shutdown Panel Instrumentation, RCS Pressure-Temperature Limits Valid until 32 EFPY ....... 5-4
6-1 Minimum Temperature Requirements for SONGS Unit 2 until 32 EFPY ....... 6-1
6-2 Limiting Indicated Minimum Pressure Requirements for SONGS Unit 2 until 32 EFPY ....... 6-2
7-1 Calculation of Chemistry Factor Values for the Surveillance Plate and Weld Materials ....... 7-2
7-2 Credibility Test of the Calculated Chemistry Factors ....... 7-2
A-1 Low Temperature RCS Overpressure Protection Range ....... A-8


## LIST OF FIGURES


1-1 Fluence Analysis Methodologies for SONGS Unit 2 Surveillance Capsule ....... 1-10
5-1 SONGS Unit 2 RCS Heatup RCS Pressure-Temperature Limits until 32 EFPY – Normal Operation ....... 5-5
5-2 SONGS Unit 2 RCS Cooldown RCS Pressure-Temperature Limits until 32 EFPY – Normal Operation ....... 5-6
5-3 SONGS Unit 2 RCS Pressure-Temperature Cooldown Limits until 32 EFPY - Remote Shutdown Panel Operation ....... 5-7
A-1 SONGS Unit 2 RCS Heatup P-T Limits until 32 EFPY – Normal Operation ....... A-5
A-2 SONGS Unit 2 RCS Cooldown P-T Limits until 32 EFPY – Normal Operation ....... A-6
A-3 SONGS Unit 2 RCS Cooldown P-T Limits until 32 EFPY – Remote Shutdown Panel Operation ....... A-7

## LIST OF ACRONYMS

| | |
|---|---|
| ART | Adjusted Reference Temperature |
| CF | Chemistry Factor |
| CFR | Code of Federal Regulations |
| EFPY | Effective Full Power Years |
| EOC | End of Cycle |
| EOL | End of Life |
| ff | Fluence factor |
| GL | (NRC) Generic Letter |
| HAZ | Heat Affected Zone |
| HPSI | High Pressure Safety Injection |
| LCO | (Technical Specification) Limiting Condition for Operation |
| LCS | Licensee Controlled Specification |
| LTOP | Low-Temperature Overpressure Protection |
| MeV | Million electron volts |
| NDTT | Nil Ductility Transition Temperature |
| NRC | Nuclear Regulatory Commission |
| NSF | Non-saturation correction factor |
| P-T | Pressure-Temperature |
| PHTP | Preservice Hydrostatic Test Pressure |
| PTLR | Pressure Temperature Limits Report |
| PTS | Pressurized Thermal Shock |
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| RPD | Relative Power Density |
| $RT_{NDT}$ | Reference Temperature per ASME Code NB2300 |
| $RT_{PTS}$ | Pressurized Thermal Shock Reference Temperature |
| RV | Reactor Vessel |
| SCE | Southern California Edison |
| SDCS | Shutdown Cooling System |
| SER | (NRC) Safety Evaluation Report |
| SONGS | San Onofre Nuclear Generating Station |
| UFSAR | Updated Final Safety Analysis Report |

# ABSTRACT

During the development of the improved standard technical specifications, the NRC staff agreed with the industry that the Pressure Temperature (P-T) and Low Temperature Overpressure Protection (LTOP) system curves and setpoints may be voluntarily relocated outside the technical specifications to a licensee-controlled document. This change, promulgated in Generic Letter 96-03, *Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits*, permits the licensee to maintain these limits efficiently and at a lower cost, provided that the parameters for constructing the curves and setpoints are derived using a methodology approved by the NRC.

Pressure and temperature limits for heatup and cooldown of the San Onofre Nuclear Generating Station (SONGS) Unit 2 reactor coolant system corresponding to 32 Effective Full Power Years (EFPY) of operation are developed in this report. These heatup and cooldown limits are designed to prevent potential brittle fracture of the reactor pressure vessel during the most restrictive low temperature overpressure event. SONGS Unit 2 Technical Specifications affected by Pressure-Temperature or LTOP limits are discussed in Sections 3 and 5 of this report. A summary of the SONGS Unit 2 Technical Specifications changes is shown in Appendix A.

The methodology in this document is applicable to both SONGS Units 2 & 3. However, only the Pressure and Temperature Limits affecting Unit 2 and corresponding Technical Specification information is contained in this report. The SONGS LTOP methodology described in this report is conservative relative to the current methods approved by the NRC and encompass the expected operating conditions for SONGS-2 until 32 EFPY.

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# INTRODUCTION

This document contains the information needed to develop the P-T limit curves and LTOP setpoint values and curves for San Onofre Nuclear Generating Station (SONGS) Unit 2, extend these curves until 32 effective full-power years (EFPY) of operation, and support the relocation of these curves from the Technical Specifications into this PTLR. Methodology used in this report was approved by the NRC in CE NPSD-683-A, Rev 6, Reference 1, which in turn is based on the guidance contained in Generic Letter (GL) 96-03, Reference 2. Additional guidance on the information needed in the SONGS-specific PTLR in order to satisfy GL 96-03 criteria is contained in the Safety Evaluation issued for CE NPSD-683-A, Rev 6.

Consistent with Generic Letter 96-03, Sections 1 through 7 of this report develop the P-T limits, establish LTOP setpoints, calculate the Adjusted Reference Temperature, develop a reactor vessel surveillance program, and calculate the neutron fluence to support the SONGS Unit 2 PTLR. The methodology used is compatible with that currently approved by the NRC and is presently used in the design bases for SONGS Units 2 & 3. This PTLR incorporates the currently approved SONGS methods, clarifies the differences, and justifies the changes relative to the NRC-approved PTLR report, Reference 1.

The methodology in this document is applicable to both SONGS Units 2 & 3. However, only the P-T limits affecting Unit 2 and corresponding Technical Specification information is contained in this report. The peak adjusted reference temperatures and P-T limit curves contained in this report are valid for SONGS-2 until 32 EFPY.

## 1.0 NEUTRON FLUENCE VALUES

The reactor vessel beltline neutron fluence has been calculated for the critical locations in accordance with Reference 3. The following discussion gives the results of the fluence calculation followed by the details of the calculational analysis for SONGS Unit 2.

The peak value of neutron fluence (E>1 MeV) at the vessel wetted surface projected to 32 effective full power years (EFPY) is $4.37 \times 10^{19}$ neutrons per square centimeter (n/cm$^2$) and corresponds to the lower shell plates. This value is used as input to the adjusted reference temperature (ART) calculations for SONGS Unit 2. The peak fluence for the intermediate shell for 32 EFPY is $4.32 \times 10^{19}$ n/cm$^2$. The fluence values have an associated two-sigma ($2\sigma$) uncertainty of ±22.8%, Reference 4.

The SONGS Unit 2 capsule was located at 7.0 degrees (off the major axis) for Cycles 1 through 10, References 5, 6. The core power distribution during these ten irradiation cycles was symmetric in both azimuthal and axial direction. That is, the axial power shape was roughly the same for any azimuthal angle and the azimuthal power shape is the same for any height. This means that the neutron flux at some point (R, θ, Z) can be considered to be a separable function of (R, θ) and (R, Z). Therefore, irradiation for Cycles 1 through 10 was modeled using the standard synthesis procedures of Reference 7.

Figure 1-1, Reference 4, depicts the analytical process that is used to determine the fluence accumulated over each irradiation period. As shown in the figure, the analysis is divided into seven tasks: (1) generation of the neutron source, (2) development of the DORT geometry models, (3) calculation of the macroscopic material cross sections, (4) synthesis of the results, and (5-7) estimation of the calculational bias, the calculational uncertainty, and the final fluence. Additional detail for these tasks is provided in the following sections.

## 1.1 INPUT DATA

### 1.1.1 Materials and Geometry

The time-averaged space and energy-dependent neutron sources for Cycles 1-10 were calculated using the SORREL code, Reference 8. The effects of burnup on the spatial distribution of the neutron source were accounted for by calculating the cycle average fission spectrum for each fissile isotope on an assembly-by-assembly basis, and by determining the cycle-average specific neutron emission rate. This data was then used with the normalized time weighted average pin-by-pin relative power density (RPD) distribution to determine the space and energy-dependent neutron source. The azimuthally averaged, time averaged axial power shape in the peripheral assemblies was used with the fission spectrum of the peripheral assemblies to determine the neutron source for the axial DORT run. These two neutron source distributions were input to DORT as indicated in Figure 1-1. Three separate sources (Cycles 1-9a, 9b and 10) were developed in order to account for two changes in $T_{ave}$ that occurred between Cycles 9 and 10.

The system geometry models for the mid-plane (R, θ) DORT were developed using standard interval size and configuration guidelines. The (R, θ) model for the Cycles 1-9a, 9b, and 10 analysis extended radially from the center of the core to the outer surface of the pressure vessel, and azimuthally from the major axis to 45°. The axial model extended from 35 cm below the active core region to 35 cm above the active core region. This geometric model either meets or exceeds all guidance criteria concerning interval size that are provided in US Regulatory Guide 1.190, Reference 3. Cold dimensions were used in all cases. The geometry models were input to the DORT code as indicated in Figure 1-1. These models can be used in all subsequent fluence analyses for SONGS Unit 2.

### 1.1.2 Cross Sections

In accordance with Regulatory Guide 1.190, the BUGLE-93 cross section library, Reference 9, was used. The GIP code, Reference 10, was used to calculate the macroscopic energy-dependent cross sections for all materials used in the analysis, i.e., from the core out through the pressure vessel and from core plate to core plate. The ENDF/B-VI dosimeter reaction cross sections were used to generate the response functions that were used to calculate the DORT-calculated "saturated" specific activities.

## 1.2 CORE NEUTRON SOURCE

The primary tool used in the determination of the flux and fluence exposure to the surveillance capsule dosimeters is the two-dimensional discrete ordinates transport code DORT, Reference 11.

The cross sections, geometry, and appropriate source were combined to create a set of DORT models (r, θ and r, z) for the cycles 1-9a, 9b and 10 analyses. Each DORT run utilized a cross section Legendre expansion of three ($p_3$), seventy directions ($s_{10}$), with the appropriate boundary conditions. The r, z models used a cross section Legendre expansion of three ($p_3$), forty-eight directions ($s_8$), with the appropriate boundary conditions. A theta-weighted flux extrapolation model was used, and all other requirements of Regulatory Guide 1.190 that relate to the various DORT parameters, were either met or exceeded for all DORT runs.

### 1.2.1 Synthesized Three Dimensional Results

DORT analyses produce two sets of two-dimensional flux distributions; one for a vertical cylinder and one for the radial plane for each set of dosimetry. The vertical cylinder, referred to as the R, Z plane, is defined as the plane bounded 35 cm above and 35 cm below the active core region, and radially by the center of the core and the outside surface of the reactor pressure vessel. The horizontal plane, referred to as the R, θ plane, is defined as the radial plane bounded by the center of the core and the outside surface of the pressure vessel, and azimuthally by the major axis and the adjacent 45 degree radius. The vessel flux, however, varies significantly in all three cylindrical-coordinate directions (R, θ, Z). This means that if a point of interest is outside the boundaries of both R, Z DORT and R, θ DORT, then the true flux cannot be determined from either DORT run. Under the assumption that the three-dimensional flux is a separable function (Reference 7), both two-dimensional data sets were mathematically combined to estimate the flux at all three-dimensional points (R, θ, Z) of interest. The basis used for the flux-synthesis process is identical to the procedure outlined in Regulatory Guide 1.190.

### 1.2.2 Calculated Activities and Measured Activities

The calculated activities for each dosimeter type "d" for each irradiation period were determined using the following equation:

$$C_d = \sum_{g=1}^{G} \phi_g(\bar{r}_d)\, x\, RF_g^d\, x\, B_d\, x\, NSF \qquad (1\text{-}1)$$

where:

$C_d$ = calculated specific activity for dosimeter "d" in μCi of product isotope per gram of target isotope

$\phi_g(\bar{r}_d)$ = three dimensional flux for dosimeter "d" at position $\bar{r}_d$ for energy group "g"

$RF_g^d$ = dosimeter response function for dosimeter "d" and energy group "g"

$B_d$ = bias correction factors for dosimeter "d"

NSF = non-saturation correction factor

For this analysis, three separate sets of activities will be calculated, therefore a combination of the calculated activities must be performed. The end of cycle 10 (EOC 10) total calculated activity, $C_d^{10}$ will be accomplished using Equation 1-1 for dosimeter "$_{d,}$"

$$(C_{d(\text{cycles } 1\text{-}10)}) = (C_{d(\text{cycles } 1\text{-}9a)}) + (C_{d(\text{cycle } 9b)}) + (C_{d(\text{cycle } 10)}) \qquad (1\text{-}2)$$

Each activity in Equation 1-2, $C_{d(\text{cycle})}$, is calculated using Equation 1-1, however each cycle-specific set of data, i.e., 1-9a, 9b, and 10, is calculated using a cycle specific NSF(cycle) factor.

The bias correction factors ($B_d$) in the specific activity calculation above are listed in Table 1-1.

**Table 1-1**
**Bias Correction Factors**

| Dosimeter Type | Bias |
|---|---|
| Activation | Short Half Life |
| Fission | Photofission |
| | Impurities |

A photo-fission factor was applied to correct for the fact that some of the cesium-137 atoms present in the dosimeter were produced by *(γ, f)* reactions and were not accounted for in DORT analysis. Likewise, an impurity factor was included to account for U-235 content in the U-238 dosimetry. The short half life was insignificant and therefore was not applied.

### 1.2.3 Measurement / Calculation Ratios

The following explanations define the meanings of the terms "measurements" and "calculations" as used in this analysis, Reference 7.

- **Measurements:**

  The term "measurements" as used here means the measurement of the physical quantity of the dosimeter (specific activity) that responded to the neutron fluence, not to the "measured fluence." For example, reference to an iron dosimeter measurement means the specific activity of $^{54}Mn$ in μCi /g, which is the product isotope of the dosimeter reaction:

  $$^{54}Fe + {}^{1}_{0}n \rightarrow {}^{54}Mn + p^{+}$$

- **Calculations:**

  The calculational methodology produces two primary results, the calculated dosimeter activities and the neutron flux at all points of interest. The term "calculation" as used here means the calculated dosimeter activity. The calculated activities are determined in such a way that they are directly comparable to the measurement values, but without recourse to the measurements. That is, the calculated values determined by DORT are directly comparable to the measurement values. ENDF/B-VI based dosimeter reaction cross sections, Reference 12, and response functions were used in determining the calculated values for each individual dosimeter. In summary, it should be stressed that the calculation values in this approach, Reference 7, are independent of the measurement values.

## 1.3 FLUENCE CALCULATION

The following values were obtained from Reference 4:

- End of Cycle 10 (EOC 10) = 13.28 EFPY
- Wetted surface cumulative fluence, as shown in Table 1-2.

The fluence values for 20 EFPY shown in Table 1-2 were calculated by linear interpolation. Fluence values for any time between 13.28 and 32 EFPY can be calculated by linear interpolation.

**Table 1-2**
**Calculated Wetted Surface Neutron Fluence (in $10^{19}$ n/cm$^2$)**

| Location | 13.28 EFPY | 20 EFPY | 32 EFPY |
|---|---|---|---|
| Intermediate shell (Plates C-6404-1, 2, 3) | 1.77 | 2.69 | 4.32 |
| Lower shell (Plates C-6404-4, 5, 6) | 1.79 | 2.72 | 4.37 |

Reference 13 gives the following equation for the attenuation of fluence with distance into the plate:

$$f = f_{surf}\,(e^{-0.24x}) \tag{1-3}$$

Where: $f$ = fluence at the desired location,
$f_{surf}$ = fluence at the wetted surface of the vessel,
$x$ = distance, in inches, from the wetted surface of the vessel to the desired location.

For each location (intermediate and lower shells), the fluence at 1/4T and 3/4T are calculated by using Equation (1-3), with the following values of x from Reference 4:

$x = 2.375$ in. for 1/4T
$x = 6.6875$ in. for 3/4T

The fluence factors were calculated by using Equation (1-4) from Reference 13:

$$ff = f^{(0.28 - 0.10 \log f)} \tag{1-4}$$

**Table 1-3**
**Fluence in $10^{19}$n/cm$^2$ and Fluence Factors for 1/4T and 3/4T Locations**

| Location | EFPY | 1/4T f* | 1/4T ff** | 3/4T f* | 3/4T ff** |
|---|---|---|---|---|---|
| Intermediate Shell (Plates C-6404-1, 2, 3) | 20 | 1.520 | 1.116 | 0.540 | 0.828 |
| | 32 | 2.443 | 1.240 | 0.868 | 0.960 |
| Lower Shell (Plates C-6404-4, 5, 6) | 20 | 1.538 | 1.119 | 0.546 | 0.831 |
| | 32 | 2.471 | 1.243 | 0.878 | 0.963 |

*f – neutron fluence
**ff – fluence factor per Equation 1-4

## 1.4 METHODOLOGY QUALIFICATION AND UNCERTAINTY ESTIMATES

The SONGS Unit 2 Cycles 1 through 10 fluence predictions are based on the methodology described in the Framatome ANP "Fluence and Uncertainty Methodologies" topical report, Reference 7. Time-averaged fluxes and fluence values throughout the reactor and vessel are calculated with the DORT discrete ordinate computer code using three-dimensional synthesis methods. The basic theory for synthesis is described in Section 3.0 of Reference 7. DORT three-dimensional synthesis results are the bases for the fluence predictions using the Framatome ANP "Semi-Analytical" (calculational) methodology.

### 1.4.1 Analytic Uncertainty Analysis

The uncertainties in the SONGS Unit 2 fluence values have been evaluated to ensure that the greater than 1.0 MeV calculated fluence values are accurate with no discernible bias, and have a mean standard deviation that is consistent with the Framatome ANP benchmark database of uncertainties. Consistency between the fluence uncertainties in the updated calculations for SONGS Unit 2 Cycles 1-10 and those in the Framatome ANP benchmark database ensures that the vessel fluence predictions are consistent with the 10 CFR 50.61, Pressurized Thermal Shock (PTS) screening criteria and Regulatory Guide 1.99, Reference 13, embrittlement evaluations.

The verification of the fluence uncertainty for the SONGS Unit 2 reactor includes:

- Estimating the uncertainties in the Cycles 1 through 10 dosimetry measurements,
- Estimating the uncertainties in the Cycles 1 through 10 benchmark comparison of calculations to measurements,
- Estimating the uncertainties in the Cycles 1 through 10 pressure vessel fluence, and
- Determining if the specific measurement and benchmark uncertainties for Cycles 1-10 are consistent with the Framatome ANP database of generic uncertainties in the measurements and calculations.

The embrittlement evaluations in Regulatory Guide 1.99 and those in 10 CFR 50.61 for the PTS screening criteria apply a margin term to the reference temperatures. The margin term includes the product of a confidence factor of 2.0 and the mean embrittlement standard deviation. The factor of 2.0 implies a very high level of confidence in the fluence uncertainty as well as the uncertainty in the other variables contributing to the embrittlement. The dosimeter measurements from the SONGS Unit 2 analysis would not directly support this high level of confidence. However, the dosimeter measurement uncertainties are consistent with the Framatome ANP database. Therefore, the calculational uncertainties in the updated fluence predictions for SONGS Unit 2 are supported by 728 additional dosimeter measurements and thirty-nine benchmark comparisons of calculations to measurements as shown in Appendix A of Reference 7. The calculational uncertainties are also supported by the fluence sensitivity evaluation of the uncertainties in the physical and operational parameters, which are included in the vessel fluence uncertainty, Reference 7. The dosimetry measurements and benchmarks, as well as the fluence sensitivity analyses in the topical report, are sufficient to support a 95 percent confidence level with a confidence factor of ±2.0, in the fluence results from the "Semi-Analytical" methodology.

The Framatome ANP generic uncertainty in the dosimetry measurements has been determined to be unbiased and has an estimated standard deviation of 7.0 percent for the qualified set of dosimeters. The SONGS Unit 2 Cycles 1-10 dosimetry measurement uncertainties were evaluated to determine if any biases were evident and to estimate the standard deviation. The dosimetry measurements were found to be appropriately calibrated to standards traceable to the National Institute of Standards and Technology and are thereby unbiased by definition. The mean measurement uncertainty associated with Cycles 1-10 is as follows:

$$\sigma_M = 6.27\%$$

This value was determined from Equation 7.6 in Reference 7 and indicates that there is consistency with the Framatome ANP database. Consequently, when the database is updated, the SONGS Unit 2 Cycles 1-10 dosimetry measurement uncertainties may be combined with the other 728 dosimeter measurements. Since Cycle 1-10 measurements are consistent with the database, it is estimated that the SONGS Unit 2 dosimeter measurement uncertainty may be represented by the database standard deviation of 7.0 percent, per Appendix D of Reference 4. Based on the database, there appears to be a 95 percent level of confidence that 95 percent of the SONGS Unit 2 dosimetry measurements for fluence reactions above 1.0 MeV are within ±14.2 percent of the true values.

### 1.4.2 Comparison with Benchmark and Plant Specific Measurements

The Framatome ANP generic uncertainty for benchmark comparisons of dosimetry calculations relative to the measurements indicates that any benchmark bias in the greater than 1.0 MeV results is too small to be uniquely identified. The estimated standard deviation between the calculations and measurements is 9.9 percent. This implies that the root mean square deviation between the calculations of the SONGS Unit 2 dosimetry and the measurements should be approximately 9.9 percent in general and bounded by ±20.04 percent for a 95 percent confidence interval with thirty-nine independent benchmarks.

The weighted mean values of the ratio of calculated dosimeter activities to measurements (C/M) for Cycles 1-10 have been statistically evaluated using Equation 7.15 from Reference 7. The standard deviation in the benchmark comparisons is as follows:

$$\sigma_{C/M} = 0.8133\%$$

This standard deviation indicates that the benchmark comparisons are consistent with the Framatome ANP database. Consequently, when the database is updated, the Cycles 1-10 benchmark uncertainties may be included with the other thirty-nine benchmark uncertainties in Reference 7.

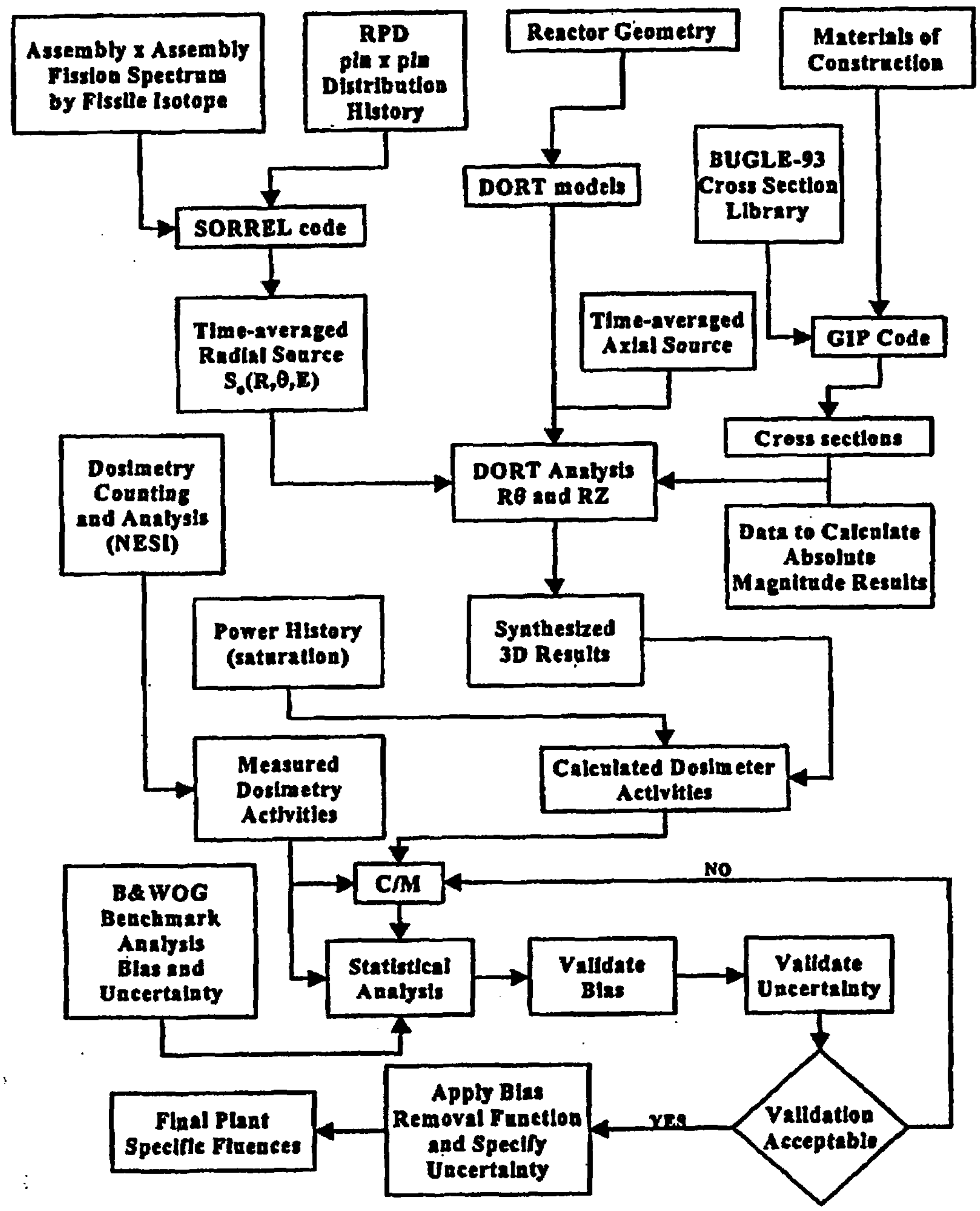
### 1.4.3 Overall Bias and Uncertainty

The consistency between the Cycles 1-10 benchmark uncertainties and those in the database indicates that the SONGS Unit 2 fluence calculations for Cycles 1-10 have no discernible bias for fluence values greater than 1.0 MeV. In addition, this consistency indicates that the fluence values can be represented by the Framatome ANP reference set which includes a calculational standard deviation of 7.0 percent at dosimetry locations. That is, the uncertainty in the calculated neutron fluence values is as shown in Table 1-4:

**Table 1-4**
**Neutron Fluence Uncertainty**

| Type of Calculation | Uncertainty (%) | |
|---|---|---|
| | Standard Deviation ($\sigma$) | 95% / 95% Confidence ($\approx \pm 2\sigma$) |
| Capsule | 7.0 | 14.2 |
| Pressure Vessel (maximum location) | 10.0 | 20.0 |
| Pressure Vessel (extrapolation) | 11.4 | 22.8 |

**Figure 1-1**
**Fluence Analysis Methodology for SONGS Unit 2 Surveillance Capsule**



# 2.0 REACTOR VESSEL SURVEILLANCE PROGRAM

The reactor vessel surveillance program for SONGS Units 2 and 3 is being conducted to monitor the neutron-irradiation induced changes in mechanical properties of the reactor vessel materials. The reactor vessel surveillance program and the surveillance capsule withdrawal schedule are described below and in References 5 and 6. The reports describing the pre-irradiation and post-irradiation evaluations of the surveillance materials are contained in References 4, 14 & 15.

The surveillance program for SONGS Units 2 and 3 was designed in accordance with ASTM E185-70, *"Standard Recommended Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Reactor Vessel Material."* ASTM E185-70 presents criteria for monitoring changes in the fracture toughness properties of reactor vessel beltline materials. The reactor vessel surveillance program for SONGS Units 2 and 3 adheres to all ASTM E185-73 requirements and to 10 CFR 50 Appendix H, with the exception of the method of attachment of the holders for the six surveillance capsules in each unit. At SONGS, the capsule holders are attached directly to the cladding on the inside of the vessel in the beltline region. The current requirements of 10 CFR 50 Appendix H (III.B.2) do not treat the method of attachment of the capsule holders as a compliance issue, since it states:

> *"...If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The design and location of the capsule holders must permit insertion of replacement capsules..."*

The capsule holder attachment method for SONGS Units 2 and 3 meets the design and inspection requirements of the ASME Code, Sections III and XI. Therefore, there are no deviations or exceptions needed from the current requirements of 10 CFR 50 Appendix H.

## 2.1 TEST MATERIAL SELECTION

Three metallurgically-different materials representative of the reactor vessel are used for test specimens in accordance with ASTM E185-73. These materials include base metal, weld metal, and heat affected zone (HAZ) materials. The guidelines for the selection of capsule specimen materials are presented in ASTM E185-73, Annex Al. Selection considerations are based on the equivalency of initial reference temperatures and residual element (copper and phosphorus) content. The working definition of equivalency is an initial reference temperature differing by no more than ±30°F, copper content differing by no more than ±0.03 weight-percent and phosphorus content differing by no more than ±0.003 weight percent.

### 2.1.1 Plate Material Selection

The plate materials for Unit 2 exhibited 30°F difference in initial $RT_{NDT}$. Therefore, the reference transition temperatures are considered to be equivalent, based on the guidelines of Annex Al of ASTM E185-73. (See Reference 16 for background information.) When the base metals have equivalent initial reference temperatures, an analysis of the copper and phosphorus contents is performed to determine equivalency of those elements. Since copper differences were ≤0.03 weight percent (considered to be equivalent) and phosphorus differences were >0.003 weight percent (considered non-equivalent), the plate material with the highest phosphorus content was selected. Based on the average chemistry values presented in Table 5.2-5 of the UFSAR, plate C-6404-2 has the highest phosphorus content and, therefore, was selected (Note 1) as the surveillance material for Unit 2.

Base metal test materials for Unit 2 were manufactured from sections of intermediate shell plates C-6404-2, Reference 6. The section of shell plate used was adjacent to the test material used for ASME Code Section III tests and was at a distance of at least one plate thickness from any water-quenched edge. This material was heat-treated to a metallurgical condition representative of the final metallurgical condition of the base metal in the completed reactor vessel.

In addition to the base materials from sections of the reactor vessel shell plate, material from a standard heat of ASTM A533 Bl steel, made available through the NRC-sponsored Heavy Section Steel Technology program, is also included. This reference material has been fully processed and has been characterized as to the sensitivity of its mechanical and fracture toughness properties to neutron radiation embrittlement. Correlation monitors provide an independent check on the measurement of the estimated irradiation conditions for the surveillance materials. Compilation of data generated from post-irradiation tests of the correlation monitors has been carried out in the Heavy Section Steel Technology Program.

A summary of the materials included in the six surveillance capsules is presented in Table 2-1 (from Reference 6).

### 2.1.2 Weld Material Selection

The weld material for the surveillance weld was selected to duplicate Weld Seam 9-203, Reference 6. (The equivalency approach was not used.)

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[1] Note that if the selection process were performed today using the newest version of ASTM E185 (E185-95), the plate material with the highest adjusted reference temperature, ART, at end-of-life would be selected. Plate C-6404-5 is predicted to have the highest ART, but only by 3°F from the originally selected plate C-6404-2. This difference is insignificant and plate C-6404-2 would be adequate under the latest criteria.

Weld metal and HAZ material specimens are produced by welding together sections from the base metal plate and another intermediate plate of the reactor vessel. The surveillance weldment for SONGS Unit 2 was fabricated using 3/16 inch diameter bare wire of Type Mil B-4, heat number 90130 and Linde Type 0091 flux, lot number 0842. This is the same heat of weld wire and flux lot as was used in the reactor vessel seam 9-203 for Unit 2 as specified in the fabrication records. The HAZ test material is manufactured from a section of the same shell plate used for the base metal test material. The section of shell plate used for weld metal and HAZ test material are adjacent to the test material used for ASME Code Section III tests and are at a distance of at least one plate thickness from any water-quenched edge. The procedure for making the intermediate-to-lower shell girth weld in the reactor vessel was repeated in manufacturing the weld metal and HAZ test materials. The heat-treatment of the surveillance weld materials was equivalent to the heat treatment accorded the reactor vessel. A summary of the weld and HAZ materials used in the surveillance capsules is presented in Table 2-1 (cf., Reference 6).

## 2.2 TEST SPECIMENS

### 2.2.1 Type and Quantity

The magnitude of the neutron-induced property changes of the reactor vessel materials is determined by comparing the results of tests using irradiated impact and tensile specimens to the results of similar tests using unirradiated specimens. The changes in $RT_{NDT}$ of the vessel materials are determined by adding to the reference temperature ($RT_{NDT}$) the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 30 ft-lb. The new values of reference temperature are known as adjusted reference temperature (ART).

Drop weight, Charpy impact, and tensile test specimens were provided for unirradiated tests. Drop weight tests were conducted in accordance with ASTM E208. Charpy impact tests were conducted in accordance with ASTM E23. Tensile tests were conducted in accordance with ASTM E8 and E21. Correlation of drop weight and Charpy impact tests to establish reference temperature were made in accordance with NB-2300 of the ASME Code, Section III. Charpy impact and tensile test specimens are provided for post-irradiation tests.

The total quantity of specimens furnished for carrying out the overall requirements of this program is presented in Reference 6. A sufficient amount of base metal, weld metal, and HAZ test material to provide two additional sets of test specimens has been obtained with full documentation and identification for future evaluation should the need arise. Each of the test materials was chemically analyzed for approximately 21 elements, including all those listed in Paragraph 4.1.3 of ASTM E185-73.

### 2.2.2 Unirradiated Specimens

The type and quantity of test specimens provided for establishing the properties of the unirradiated reactor vessel materials are presented in Reference 6. The data from tests of these specimens provide the basis for determining the neutron-induced property changes of the reactor vessel materials.

**Drop Weight Test Specimens:** Twelve drop weight test specimens, each of the base metal (longitudinal and transverse), weld metal, and HAZ material are provided for establishing the nil ductility transition temperature (NDTT) of the unirradiated surveillance materials. These data form the basis for $RT_{NDT}$ determination. $RT_{NDT}$ is the reference temperature from which subsequent neutron-induced changes are determined.

**Charpy Impact Test Specimens:** Thirty test specimens, each of base metal (longitudinal and transverse), weld metal, and HAZ material are provided for impact testing. This quantity exceeds the minimum number of test specimens recommended by ASTM E185 for developing a Charpy impact energy transition curve and is intended to provide a sufficient number of data points for establishing accurate Charpy impact energy transition temperatures for these materials. These data, together with the drop weight NDTT, are used to establish an $RT_{NDT}$ for each material.

**Uniaxial Tension Test Specimens:** Eighteen tensile test specimens, each of base metal (longitudinal and transverse), weld metal, and HAZ materials are provided for tension testing. This quantity also exceeds the minimum number of test specimens recommended by ASTM E185 and is intended to permit a sufficient number of tests to accurately establish the tensile properties for these materials at a minimum of three test temperatures; e.g., ambient, operating, and design.

### 2.2.3 Irradiated Specimens

Both tensile and impact test specimens are used for determining changes in the static and dynamic properties of the materials due to neutron irradiation. A total of 288 Charpy impact and 54 tensile test specimens are provided. The type and quantity of test specimens provided for establishing the properties of the irradiated materials over the lifetime of the vessel are presented in Table 2-1 (cf., Reference 6).

## 2.3 SPECIMEN IRRADIATION

### 2.3.1 Encapsulation of Specimens

The test specimens are housed within corrosion-resistant capsule assemblies in order to:

- Prevent corrosion of the carbon steel test specimens by the primary coolant during irradiation,
- Physically locate the test specimens in selected locations within the reactor, and

- Facilitate the removal of a desired quantity of test specimens from the reactor when a specified fluence has been attained.

A typical capsule assembly (cf., References 5 and 6) consists of a series of seven specimen compartments, connected by wedge couplings, and a lock assembly. Each compartment enclosure of the capsule assembly is internally supported by the surveillance specimens and is externally pressure tested during final fabrication. The wedge couplings also serve as end caps for the specimen compartments and position the compartments within the capsule holders, which are attached to the reactor vessel. The lock assemblies fix the locations of the capsules within the holders by exerting axial forces on the wedge coupling assemblies which cause these assemblies to exert horizontal forces against the sides of the holders preventing relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor.

Each capsule assembly is made up of four Charpy impact test specimens (Charpy impact) and three tensile test specimen-flux/temperature monitor (tensile-monitor) compartments. Each capsule compartment is assigned a unique identification so that a complete record of test specimen location within each compartment can be maintained.

#### 2.3.1.1 Charpy Impact Compartments

Each Charpy impact compartment contains 12 Charpy impact specimens. This quantity of specimens provides an adequate number of data points for establishing an impact energy transition curve for a given irradiated material. Comparison of the unirradiated and irradiated Charpy impact energy transition curves permits determination of the $RT_{NDT}$ changes due to irradiation for the various materials.

The specimens are arranged vertically in four 1 x 3 arrays and are oriented with the notch toward the core. The temperature differential between the specimens and the reactor coolant is minimized by using spacers between the specimens and the compartment and by sealing the entire assembly in an atmosphere of helium.

#### 2.3.1.2 Tensile-Monitor Compartments

Each tensile-monitor compartment contains three tensile test specimens, a set of neutron flux monitors, and a set of temperature monitors for estimating the maximum temperature to which the specimens have been exposed. The entire tensile-monitor compartment is sealed within an atmosphere of helium. The tensile specimens are placed in a housing machined to fit inside the compartment. Split spacers are placed around the gage length of the specimens to minimize the temperature differential between the specimen gage length and the coolant.

### 2.3.2 Flux and Temperature Measurement

The changes in the $RT_{NDT}$ of the reactor vessel materials are derived from specimens irradiated to various fluence levels and in different neutron energy spectra. In order to permit accurate predictions of the $RT_{NDT}$ of the vessel materials, complete information on the neutron flux, neutron energy spectra, and the irradiation temperature of the surveillance specimens must be available.

#### 2.3.2.1 Flux Measurements

Neutron flux measurements are obtained from detectors located in each of the six irradiation capsules. Such detectors are particularly suited for the proposed application because their effective threshold energies lie in the low MeV range. (See References 5 and 6 for a list of detectors used.) Selection of threshold detectors is based on the recommendations of ASTM E261, *"Method of Measuring Neutron Flux by Radioactive Techniques."*

Neutron threshold detectors can be used to monitor the thermal and fast neutron spectra incident on the test specimens. These detectors possess reasonably long half-lives and activation cross-sections covering the desired neutron energy range. One set of neutron flux spectrum monitors is included in each tensile-monitor compartment. Each detector is placed inside a sheath which identifies the material and facilitates handling. Cadmium covers are used for those materials; e.g., uranium, nickel, copper and cobalt, which have competing neutron capture activities. The flux monitors are placed in holes drilled in stainless steel housings at three axial locations in each capsule assembly to provide an axial fluence profile for each set of test specimens.

In addition to these detectors, the program also includes correlation monitors (Charpy impact test specimens made from a reference heat ASTM A533 B1) which are irradiated along with the specimens made from reactor vessel materials. The changes in impact properties of the reference material provide a cross-check on the dosimetry in any given surveillance program. These changes also provide data for correlating the result from this surveillance program with the results from experimental irradiations and other reactor surveillance programs using the same reference material.

#### 2.3.2.2 Temperature Estimates

Because the changes in mechanical and impact properties of irradiated specimens are highly dependent on the irradiation temperature, it is necessary to have knowledge of the temperature of the specimens as well as that of the pressure vessel. During irradiation, instrumented capsules are not practical for a surveillance program extending over the design lifetime of a power reactor. The maximum temperature of the irradiated specimens can be estimated with reasonable accuracy by including in the capsule assemblies small pieces of low melting point alloys or pure metals. The compositions of candidate materials with melting points in the operating range of power reactors are listed in References 5 and 6. The monitors are selected to bracket the operating temperature of the reactor vessel.

The temperature monitors consist of a helix of low melting alloy wire inside a sealed quartz tube. A stainless steel weight is provided to destroy the integrity of the wire when the melting point of the alloy is reached. The compositions and therefore the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes which contain the alloy wires.

A set of temperature monitors is included in each tensile-monitor compartment. The temperature monitors are placed in holes drilled in stainless steel housings and are also placed at three axial locations in each capsule assembly to provide an axial profile of the maximum temperature to which the specimens were exposed.

### 2.3.3 Irradiation Locations

The encapsulated test specimens are irradiated at approximately identical radial positions about the midplane of the core. The test specimens are enclosed within six capsule assemblies at axial positions that are bisected by the midplane of the core.

The test specimens contained in the capsule assemblies are used to monitor the irradiation induced property changes of the reactor vessel materials. These capsules, therefore, are positioned near the inside wall of the reactor vessel so that the irradiation conditions (fluence, flux spectrum, temperature) of the test specimens resemble, as closely as possible, the irradiation condition of the reactor vessel. The neutron fluence of the test specimens is expected to be within 15% of that seen by the adjacent vessel wall. The $RT_{NDT}$ changes resulting from the irradiation of these specimens closely approximate the $RT_{NDT}$ changes in the materials of the reactor vessel.

The capsule assemblies are placed in capsule holders positioned circumferentially about the inside of the reactor vessel. Table 2-2 presents the exposure locations for the capsule assemblies. All capsule assemblies were inserted into their respective capsule holders during the final reactor assembly operation.

### 2.3.4 Capsule Assembly Removal

Surveillance capsule assemblies are withdrawn during an appropriate refueling outage when the test specimens have attained the desired fluence. The target or actual neutron fluence for removal of each capsule assembly is presented in Table 2-2.

The target fluence levels for the surveillance capsules were determined for each azimuthal location and for the time intervals indicated in the withdrawal schedule in 10 CFR 50 Appendix H (II.B.3). The Unit 2 capsule assembly located in the 97-degree position was withdrawn as described in Reference 15. The Unit 2 capsule assembly located in the 263-degree position was withdrawn as described in Reference 4. Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

**Table 2-1**
**Type and Quantity of Specimens for Irradiation Exposure and Irradiated Tests**

| | | Quantity of Specimens | | | | |
|---|---|---|---|---|---|---|
| **Type of Specimen** | **Orientation** | **Base Metal** | **Weld Metal** | **HAZ** | **SRM[a]** | **Total** |
| Charpy Impact | Longitudinal | 48 | -- | -- | 24 | 72 |
| | Transverse | 72 | 72 | 72 | -- | 216 |
| Uniaxial Tensile | Longitudinal | -- | -- | -- | -- | -- |
| | Transverse | 18 | 18 | 18 | -- | 54 |
| Total | | 138 | 90 | 90 | 24 | 342 |

(a) Standard Reference Material (SRM) characterized by Heavy Section Steel Technology Program; specimens are provided only for correlation with characterization tests.

**Table 2-2**
**SONGS Unit 2 Capsule Assembly Removal Schedule**

| Capsule Number | Azimuthal Location (degrees) | Removal Time (Effective Full Power Years) | Fluence ($n/cm^2$) |
|---|---|---|---|
| 1* | 83 | 24 | $3.80 \times 10^{19}$ $n/cm^2$ |
| 2 | 97 | 2.85 | $5.07 \times 10^{18}$ $n/cm^2$ |
| 3 | 104 | Standby | -- |
| 4 | 284 | Standby | -- |
| 5 | 263 | 13.28 | $1.64 \times 10^{19}$ $n/cm^2$ |
| 6 | 277 | Standby | -- |

* Anticipated

# 3.0 LTOP SYSTEM LIMITS

## 3.1 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

The Low Temperature Overpressure Protection (LTOP) system protects the reactor coolant system (RCS) pressure boundary integrity by ensuring that the RCS pressure remains below the applicable P-T limits of 10 CFR 50, Appendix G, particularly at low temperatures when the RCS is water-solid.

SONGS Unit 2 Tech Spec LCO 3.4.12.1 protects the design basis assumptions for the LTOP system that no more than two HPSI pumps can be operable and that the safety injection tanks must be isolated or depressurized to less than the limit specified in LCO 3.4.3. In the event that the RCS is below the enable temperature, the Shutdown Cooling System (SDCS) relief valve shall be operable or the RCS vented with an area greater than 5.6 sq. in.

LCO 3.4.12.2 specifies that the SDCS relief valve or the pressurizer code safety valves must be operable above the enable temperature specified in LCO 3.4.3. The following sections describe the process for developing the LTOP system limits and demonstrating adequate performance at SONGS-2.

## 3.2 BASIS FOR LTOP SYSTEM LIMITS

The design basis for the LTOP System for SONGS 2 & 3 is described in Reference 17. The LTOP system limitations consist of a SDCS relief valve setpoint aligned whenever the RCS temperature is below an enable temperature along with controls on the RCS heatup and cooldown rates. The relief valve setpoint and capacity have been selected such that the peak transient pressures in the postulated overpressure events do not exceed the applicable RCS P-T limits presented in LCO 3.4.3. The development of the setpoint follows the plant-specific methods described in Reference 17 and the results bound the NRC approved methodology contained in Section 3.0 of Reference 1. Limiting temperatures for LTOP heatup and cooldown protection are given in Table 3-1.

**Table 3-1**
**Low Temperature RCS Overpressure Protection Range**

| Operating Period | Indicated Cold Leg Temperature | |
|---|---|---|
| | During Heatup | During Cooldown |
| Until 32 EFPY (Normal and Remote Shutdown Panel Operation) | ≤ 218.5°F | ≤ 190.3°F |

### 3.2.1 Peak Transient Pressures

Analyses of the worst-case energy addition (RCP start) and mass addition (HPSI start) overpressure events were performed to calculate the peak transient pressures. The design basis for the SONGS plant is contained in Reference 17 that describes the models and assumptions used to produce a conservative LTOP model and analysis for these overpressure transients. These analytic models present a bounding design for LTOP that allows the plant to operate with discretionary conservatism relative to the NRC-approved analytic models identified in Reference 1. It should be noted that the SDCS relief valve installed at SONGS Unit 2 has excess capacity relative to relieving requirements, thus any LTOP pressure transient is quickly terminated upon valve actuation. The design basis transients displayed in UFSAR Figures 5.2-6 and 5.2-7 clearly show the relative severity for several potential overpressure transients and the rapid termination of such transients.

These design basis peak pressure analyses incorporate the following assumptions:

- The pressurizer is water-solid, i.e. no steam space,
- The RCS pressure boundary is rigid, i.e. no expansion due to pressure or thermal effects,
- The RCS metal mass is adiabatic, i.e. no heat absorption from the coolant or heat losses to the outside,
- The RCS letdown flow is isolated,
- All pumps attain rated speed instantaneously,
- Only one relief valve is used in the transient mitigation,
- No operator action is required, and
- Conservative energy addition sources are used for both energy addition and mass addition transient scenarios, including:
  - Full heat output from all pressurizer heaters (1500 Kw) is assumed for the duration of the transient in order to maximize the energy input into the RCS.
  - Decay heat, increased by 10% for conservatism, is assumed constant throughout the transient at a value consistent with the earliest time after shutdown that the transient can occur.

The following additional assumptions are made to assure a conservative analysis:

- The SDCS is assumed isolated at the start of the transient in order to minimize the total volume absorbing the heat/mass addition and to isolate any heat removal from the RCS,
- The SDCS relief valve opening profile is conservative relative to the ASME model described in Section 3.2.1.1. This results in a delayed response to the relief valve lift and a resulting delay in providing the relief capability,
- No RCP seal leakage or controlled bleed-off is assumed,
- The RCS is isothermal and is not cooled or heated by the mass addition, and

- The initial conditions are chosen to maximize the pressure transients in order to develop the greatest rate of pressure rise. The RCS pressure at the initiation of the transients is selected to be 376 psia, which is the highest pressure for Shutdown Cooling System operation and is also the highest pressure for the SDCS relief valve being in service.

The following sections discuss the details for the transient analysis used in the determination of the SONGS LTOP design basis.

#### 3.2.1.1 Relief Valve Overpressure Protection

LTOP system overpressure protection at SONGS is performed by a spring actuated relief valve located in the SDCS suction line. This valve is placed in service at the LTOP enable temperature in order to protect the reactor vessel from brittle fracture in the event of a low-temperature overpressure event. The SDCS relief valve passes subcooled water due to its location in the SDCS piping. Valve opening and discharge characteristics are consistent and conservative relative to the ASME Code requirements for spring loaded safety valves and/or the manufacturer's recommendations, whichever is more conservative. This SDCS valve has a relieving capacity substantially greater than that needed to mitigate the design basis mass addition and energy addition transients affecting the reactor coolant system. The valve setpoint is sufficiently below the limiting reactor coolant system pressure established by 10 CFR 50 Appendix G, thus LTOP protection of the reactor vessel is assured.

The SDCS relief valve design parameters are a lift pressure of 417 psia allowing 3089 gpm to pass when fully open at 10% pressure accumulation; the setpoint for this valve (PSV 9349) is 406 ±10 psig. These relief valve parameters were selected for protection of the SDCS and as shown in the LTOP analyses are quite conservative and require no changes to provide the necessary LTOP function. The analytical model for the valve opening and its associated capacity prior to the 10% accumulation is an important characteristic for the limiting pressure transient scenario.

The NRC-accepted model in Reference 1 follows the ASME Code model with an initial opening at 3% accumulation, while the SONGS design basis evaluation was conducted with a model that delayed opening until 7% accumulation. This conservatism resulted in the valve opening at a 4% accumulation delay (7% versus 3%) that increased the peak pressure transient before the pressure excursion was terminated. It should be noted that the pressure rise is terminated in both limiting transients before the relief valve reaches full open.

#### 3.2.1.2 Mass Addition Overpressure Events

The design basis mass addition transient was identified as an inadvertent actuation of two HPSI pumps while all three charging pumps are operating at their design flowrate. This event was analyzed by determining incremental inputs for HPSI mass additions, charging pump mass addition, and the equivalent-mass additions that result from energy additions. The RCS, assumed to be a single node at a uniform temperature and pressure, remains at the initial temperature and volume while the mass addition (Reference 17) results in a pressure rise. The additional mass is added to the original mass and divided by the system volume to calculate an updated specific volume. A new RCS system pressure is then determined using the initial temperature and the updated specific volume. Finally, the updated pressure was assumed to be the "back pressure" for the HPSI pump delivery curve in order to determine the revised HPSI delivery for the next time step.

The HPSI mass addition was obtained from curves developed for ECCS calculations using two-pump delivery with no losses, which is the maximum volumetric delivery at any pressure difference between the reactor coolant system and the refueling water tank. A conservative low temperature is assumed for the supply water to establish the largest mass addition rate.

The overpressure transient results given in SONGS-2&3 UFSAR Figure 5.2-6 show that the relative values for the HPSI mass addition are significantly greater than the pressurizer heater effects. The incremental effect for relief valve discharge flow was combined within each time increment to develop the transient curves shown in UFSAR Figures 5.2-7. The time steps were small since the RCS was water solid and the pressure rise was rapid. The transient was quickly mitigated due to the large capacity discharge through the relief valve, making the cumulative effect of the decay heat and the pressurizer heaters inconsequential. The peak pressure is less than 450 psia as shown in UFSAR Figure 5.2.7.

#### 3.2.1.3 Energy Addition Overpressure Events

The SONGS LTOP design basis results for transient energy addition events were determined with OVERP, a computer code that simulates the pressure increase to a solid RCS due to reverse heat transfer from relatively hot steam generators when an idle RCP is started. A detailed description of the OVERP computer code is provided in Reference 18. OVERP, used extensively in LTOP analyses performed by Westinghouse, simulates the discharge from a relief device and determines pressure during the relieving action. An earlier (mainframe) version of the OVERP computer code was used in the original design basis analysis for SONGS; the current model runs on a personal computer platform. The SDCS relief valve opening characteristics in the current SONGS design basis analysis assumes valve opening at 7% accumulation rather than 3% accumulation listed in Reference 1. Although this results in the relief valve opening later and the RCS pressure transient peaking at a slightly higher value, the relieving capacity is sufficient to protect the RCS. The OVERP model used in the SONGS design bases evaluations complies with the NRC requirements in a conservative manner.

The following paragraphs discuss the energy addition model and input parameters that further illustrate the conservative nature of the earlier design basis calculation contained in the design report, Reference 17.

The following assumptions are included in the initiation of this design basis energy addition transient:

- The steam generator secondary temperature is assumed to be 100°F hotter than the primary coolant temperature.
- One RCP is assumed to start and instantaneously reach rated speed to initiate the transient. The model assumes a constant heat input for the duration of the analysis.

The analytical model assumption of constant specific volume results from the RCS fluid mass remaining constant by assuming zero RCP seal leakage and no charging flow.

With the relatively large capacity SDCS relief valve, the energy addition pressure transient is mitigated immediately upon the valve opening at 7% accumulation (447 psia). The peak pressure is less than 450 psia as shown in UFSAR Figure 5.2.7.

### 3.2.2 Applicable P-T Limits

The P-T limits for the SONGS-2 LTOP system setpoints were developed using the methodology described in Section 5.0. These heatup and cooldown P-T limits are listed in Tables 5-1 through 5-3 and are shown in Figures 5-1 through 5-3 to permit comparison of the P-T values with the peak transient pressures given in Section 3.2.1. Applicable P-T limits are established based on the method described in Reference 1, which performs a comparative evaluation of the P-T limitations developed per Section 5.0 and the peak pressurizer transient evaluation of Section 3.2.1.

Applicable limiting heatup and cooldown rates for SONGS Unit 2 are presented in Appendix A. Figures A-1 through A-3 presented in Appendix A are identical to Figures 5-1 through 5-3, respectively, but simplified for clarity and ease of use by plant operators.

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# 4.0 BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURE

The calculation of the adjusted reference temperature (ART), Reference 19, for the beltline region was performed using the NRC-accepted methodologies as described below. Application of surveillance data was used to refine the chemistry factor and the margin term in accordance with Regulatory Guide 1.99, Reference 13. The limiting ART values in the beltline region for the SONGS Unit 2 reactor vessel corresponding to 32 Effective Full Power Years (EFPY) for the 1/4T and 3/4T locations are as follows:

| Location | ART(°F) | Limiting Material |
|---|---|---|
| 1/4T | 121.8°F | Lower Shell Plate C-6404-4 |
| 3/4T | 102.7°F | Lower Shell Plate C-6404-4 |

The $RT_{PTS}$ value for the SONGS Unit 2 reactor vessel was calculated in accordance with 10 CFR 50.61. The highest predicted value is 130.8°F and corresponds to Lower Shell Plate C-6404-4. Applicable surveillance data were used to refine the chemistry factor and the margin term. The $RT_{PTS}$ determination is defined in Section 4.5.

## 4.1 BACKGROUND

Given below is the determination of the adjusted reference temperature (ART) for the SONGS Unit 2 reactor vessel beltline materials for 20 and 32 effective full power years (EFPY). These results are based on an analysis of the 263-degree surveillance capsule described in Reference 4, and the 97-degree surveillance capsule described in Reference 15.

## 4.2 RESULTS

The results of the adjusted reference temperature calculations are summarized in Table 4-1:

**Table 4-1**
**Calculated ART Values for 20 and 32 EFPY**

| EFPY | 1/4T ART | 3/4T ART | Location |
|---|---|---|---|
| 20 | 113.3°F | 93.7°F | Plate C-6404-4 |
| 32 | 121.8°F | 102.7°F | Plate C-6404-4 |

The calculated ART values reflect the added confidence from using the measured properties of the vessel, for example the Charpy transition temperature shifts as a result of measured surveillance capsule evaluations. These data were used to justify reducing the standard deviation for transition temperature shift from 34°F to 17°F in conjunction with a chemistry factor of 68.21°F. (See Section 7.0.)

### 4.2.1 Assumptions

The input and assumptions from Reference 19 were used when calculating the adjusted reference temperatures applicable to beltline materials and limiting plates and welds.

### 4.2.2 Calculation of Fluence Values for the EFPY of Interest

Fluence is a function of time and location. Reference 4 provides the fluence values at the vessel clad interface for the end of Cycle 10 (13.28 EFPY) and for EOL (32 EFPY). For any time in between, fluence can be calculated by linear interpolation.

For the attenuation of fluence with distance into the plate, Reference 13 gives the following equation:

$$f = f_{surf}\,(e^{-0.24x}) \qquad (4\text{-}1)$$

Where:

$f$ = fluence at the desired location,
$f_{surf}$ = fluence at the wetted surface of the vessel,
$x$ = distance, in inches, from the wetted surface of the vessel to the desired location.

### 4.2.3 Calculation of the Chemistry Factor Based on Surveillance Data

When surveillance data are available, according to Reference 13, the data are fitted using the following equation:

$$\Delta RT_{NDT} = (CF) * ff \qquad (4\text{-}2)$$

where:

$$ff = f^{(0.28 - 0.10 \log f)} \qquad (4\text{-}3)$$

and:

$$CF = \Sigma(\Delta RT_{NDT} * ff)_i / \Sigma_i^2 \qquad (4\text{-}4)$$

A test of the validity of the estimated Chemistry Factor (CF) consists of calculating $\Delta RT_{NDT}$ for a given fluence and comparing it with the measured $\Delta RT_{NDT}$ for that fluence. The measured $\Delta RT_{NDT}$ must fall within $\pm 1\sigma$ of the calculated $\Delta RT_{NDT}$ where $\sigma = 17F$ for base metal and $\sigma = 28F$ for welds (Reference 13).

### 4.2.4 Calculation of ART for the Limiting Plates at 1/4T and 3/4T

Adjusted reference temperatures are calculated using the following equation (Reference 13):

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{4-5}$$

where all temperatures are in degrees F.

## 4.3 ANALYSIS DETAILS

### 4.3.1 Selection of Representative and Limiting Cases

For the purpose of developing a chemistry factor that gives the best fit to the data, the materials examined were base metal plate C-6404-2 (intermediate shell) and weld metal heat 90130 (vessel weld 9-203). For each of these materials, the following data are available:

- Initial $RT_{NDT}$,
- Charpy impact data for $5.07 \times 10^{18}$ n/cm$^2$ irradiation (97-degree surveillance capsule, Reference 15), and
- Charpy impact data for $1.637 \times 10^{19}$ n/cm$^2$ irradiation (263-degree surveillance capsule, Reference 4).

For the purpose of determining the maximum ART values, the lower shell plate C-6404-4 (because of its higher fluence), an intermediate shell plate (C-6404-2), and the weld metal (9-203) will be considered.

### 4.3.2 Fluence Calculation

The following values were obtained from Reference 4:

- End of Cycle 10 (EOC 10) = 13.28 EFPY
- Wetted surface cumulative fluence, as shown in Table 4-2.

The fluence values for 20 EFPY, shown in Table 4-2, were calculated by linear interpolation.

**Table 4-2**
**Calculated Wetted Surface Fluence ($10^{19}$ n/cm$^2$)**

| Location | 13.28 EFPY | 20 EFPY | 32 EFPY |
|---|---|---|---|
| Intermediate shell (Plates C-6404-1, 2, 3) | 1.77 | 2.69 | 4.32 |
| Lower shell (Plates C-6404-4, 5, 6) | 1.79 | 2.72 | 4.37 |

For each location (intermediate and lower shells), the fluence at 1/4T and 3/4T are calculated by using Equation (4-1), with the following values of depth (x) from Reference 4:

- x = 2.375 inches for 1/4T, and
- x = 6.6875 inches for 3/4T.

The fluence factors listed in Table 4-3 were calculated by using Equation (4-3).

**Table 4-3**
**Fluence ($10^{19}$ n/cm$^2$) and Fluence Factors for 1/4T and 3/4T Locations**

| Location | EFPY | 1/4T f* | 1/4T ff** | 3/4T f* | 3/4T ff** |
|---|---|---|---|---|---|
| Intermediate Shell (Plates C-6404-1, 2, 3) | 20 | 1.520 | 1.116 | 0.540 | 0.828 |
| | 32 | 2.443 | 1.240 | 0.868 | 0.960 |
| Lower Shell (Plates C-6404-4, 5, 6) | 20 | 1.538 | 1.119 | 0.546 | 0.831 |
| | 32 | 2.471 | 1.243 | 0.878 | 0.963 |

* f = neutron fluence
** ff = fluence factor per Equation 4-3

## 4.4 DETERMINATION OF THE CHEMISTRY FACTOR

The $RT_{NDT}$ and fluence data shown in Table 4-4 were obtained from Reference 4. The fluence factors were calculated from Equation (4-3) and the Chemistry Factors from Equation (4-4). The data of Table 4-4 were used to develop Table 4-5 and to validate the calculated Chemistry Factors.

Based on the Chemistry Factors shown in Table 4-5 and described in Section 7.0, the predicted $\Delta RT_{NDT}$ values fall within $\pm 1\sigma_\Delta$ of the measured values of shift. It is therefore justifiable to use the Regulatory Guide 2.1 Chemistry Factors (CF) with the reduced margin ($1\sigma_\Delta$) as described in Reference 13.

**Table 4-4**
**Calculation of Chemistry Factor Values for the Two Materials**

| Location | 97 degree Capsule | 263 degree Capsule | Unirradiated $RT_{NDT}$ | Sum | Chemistry Factor (°F) |
|---|---|---|---|---|---|
| Fluence ($x10^{19}$ n/cm$^{2}$) | 0.507 | 1.637 | | | |
| Fluence factor (ff) | 0.810 | 1.136 | | | |
| $(ff)^2$ | 0.6561 | 1.2905 | | 1.947 | |
| **PLATE C-6404-2** | | | | | |
| $RT_{NDT}$ | 68.6 | 115.1 | 27.5 | | |
| $\Delta RT_{NDT}$ | 41.1 | 87.6 | | | |
| $ff^*\Delta RT_{NDT}$ | 33.31 | 99.51 | | 132.82 | 68.21 |
| **WELD 9-203** | | | | | |
| $RT_{NDT}$ | -49.2 | -29.9 | -53.2 | | |
| $\Delta RT_{NDT}$ | 4.0 | 23.3 | | | |
| $ff^*\Delta RT_{NDT}$ | 3.24 | 26.47 | | 29.71 | 15.26 |

**Table 4-5**
**Credibility Test of the Calculated Chemistry Factors**

| Material | $\sigma_\Delta$ | CF (°F) | Fluence (x $10^{19}$ n/cm$^2$) | ff | CF*ff (°F) ($\Delta RT_{NDT}$) | $\Delta RT_{NDT}+\sigma_\Delta$ (°F) | $\Delta RT_{NDT}-\sigma_\Delta$ (°F) | Measured $\Delta RT_{NDT}$ (°F) |
|---|---|---|---|---|---|---|---|---|
| Plate C-6404-2 | 17 | 68.21 | 0.507 | 0.810 | 55.28 | 72.28 | 38.28 | 41.1 |
| | | | 1.637 | 1.136 | 77.48 | 94.48 | 60.48 | 87.6 |
| Weld 9-203 | 28 | 15.26 | 0.507 | 0.810 | 12.37 | 40.37 | -15.63 | 4.0 |
| | | | 1.637 | 1.136 | 17.33 | 45.33 | -10.67 | 23.3 |

## 4.5 CALCULATION OF THE ADJUSTED REFERENCE TEMPERATURES AND SELECTION OF THE MAXIMA

The values of CF, f, $\sigma_\Delta$ and the initial $RT_{NDT}$ are known for all the materials of interest and are provided in Reference 19. The adjusted reference temperatures are calculated using Equation (4-5). According to Reference 13:

$$\text{Margin} = 2\,(\sigma_i^2 + (\sigma_\Delta/2)^2)^{1/2} \qquad (4\text{-}6)$$

However, $\sigma_\Delta$ need not exceed 1/2 the predicted shift (Reference 13). According to Reference 16, the standard deviation of the initial $RT_{NDT}$, $\sigma_i$, is = 0, and Margin = $\sigma_\Delta$. The calculated values of ART appear in Table 4-6, with the highest values listed in Table 4-1. The adjusted reference temperature values developed here are used to define P-T limits for SONGS Unit 2 for 32 EFPY (EOL).

Similar calculations were performed for the limiting plate and weld at end of life following 10 CFR 50.61. $RT_{PTS}$ for the plate and weld was calculated for the peak fluence at the wetted surface after 32 EFPY, $4.37 \times 10^{19}$ n/cm$^2$. The chemistry factors derived in Table 4-4, 68.21°F and 15.26°F , and the initial $RT_{NDT}$ values from Table 4-6, 20°F and -53.2°F, were used. The predicted $RT_{PTS}$ for the plate and weld, using $1\sigma_{\Delta}$ = 17°F and 28°F, is 130.8°F and -4.2°F, respectively.

**Table 4-6**
**Calculated Values for ART for 1/4T and 3/4T and for 20 and 32 EFPY**

| Plate | EFPY | Initial $RT_{NDT}$ (°F) | CF | Margin (°F) | Wet Surf. Fluence ($n/cm^2$) | 1/4T Position | | | | 3/4T Position | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | Fluence ($n/cm^2$) | ff | $\Delta RT_{NDT}$ (°F) | ART (°F) | Fluence ($n/cm^2$) | ff | $\Delta RT_{NDT}$ (°F) | ART (°F) |
| C-6404-2 | 32 | 20 | 68.21 | 17.0 | 4.320 | 2.443 | 1.240 | 84.612 | 121.6 | 0.868 | 0.960 | 65.501 | 102.5 |
| | 20 | 20 | 68.21 | 17.0 | 2.690 | 1.521 | 1.116 | 76.131 | 113.1 | 0.540 | 0.828 | 56.478 | 93.5 |
| C-6404-4 | 32 | 20 | 68.21 | 17.0 | 4.370 | 2.471 | 1.243 | 84.809 | 121.8 | 0.878 | 0.963 | 65.721 | 102.7 |
| | 20 | 20 | 68.21 | 17.0 | 2.720 | 1.538 | 1.119 | 76.336 | 113.3 | 0.546 | 0.831 | 56.686 | 93.7 |
| Weld 9-203 | 32 | -53.2 | 15.26 | 28.0 | 4.320 | 2.443 | 1.240 | 18.927 | -6.3 | 0.868 | 0.960 | 14.652 | -10.5 |
| | 20 | -53.2 | 15.26 | 28.0 | 2.690 | 1.521 | 1.116 | 17.029 | -8.2 | 0.540 | 0.828 | 12.633 | -12.6 |

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# 5.0 PRESSURE-TEMPERATURE LIMITS USING LIMITING ADJUSTED REFERENCE TEMPERATURE IN THE P-T CURVE CALCULATION

Analytical methodology approved by the NRC and described in Reference 1 is used to develop the beltline RCS pressure-temperature limits for SONGS Unit 2. These pressure-temperature limits will be removed from Technical Specification LCO 3.4.3 and updated limits presented in Appendix A of this PTLR. The PTLR methodology is also documented in the SONGS Unit 2 Technical Specifications Bases.

RCS pressure-temperature limits established for non-beltline locations do not change significantly due to the lower exposure to neutron flux unless they are updated through regulation or more recent advances in technology. However, it is appropriate to consider these non-beltline locations, as necessary, in the updating of pressure-temperature limits throughout plant life, as they are currently part of the plant design basis. Therefore, the pressure-temperature limits for the beltline region are combined with the non-beltline regions, as appropriate, to develop the set of composite curves for specific modes of operation in this PTLR. The lower bound of these composite curves defines the pressure-temperature limit for a plant at a specific mode of operation. The pressure-temperature limits for the non-beltline regions are relocated from the SONGS-2 Technical Specifications and incorporated into this PTLR.

## 5.1 RCS TEMPERATURE RATE-OF-CHANGE LIMITS

Information describing the rate-of-change limits for SONGS Unit 2 will be removed from the current Technical Specifications and updated limits valid until 32 EFPY relocated into Appendix A of this PTLR. The specific heatup and cooldown rate limits specified in LCO 3.4.3, SR 3.4.3.1, and SR 3.4.3.2 are replaced with text describing that the allowable limits are located in the PTLR. Technical Specification Figures 3.4.3-1 through 3.4.3-5 and Table 3.4.3-1 are removed in their entirety.

## 5.2 RCS PRESSURE-TEMPERATURE LIMITS

Tables 5-1 through 5-3, shown plotted in Figures 5-1 through 5-3, provide heatup, cooldown, in-service hydrostatic and leak testing, and criticality pressure and temperature limits for SONGS-2 until 32 EFPY.

**Table 5-1**
**SONGS Unit 2 Heatup at 60°F/hr**
**RCS Pressure-Temperature Limits valid until 32 EFPY**

| **Control Room Instrumentation** | |
|---|---|
| **Indicated RCS Temp (°F)** | **Indicated RCS Pressure (psig)** |
| 58.5 | 593.0 |
| 68.5 | 587.0 |
| 78.5 | 577.0 |
| 88.5 | 602.0 |
| 98.5 | 597.0 |
| 108.5 | 599.0 |
| 118.5 | 610.0 |
| 125.2 | 625.0 |
| 125.2 | 572.0 |
| 128.5 | 579.0 |
| 138.5 | 611.0 |
| 148.5 | 654.0 |
| 158.5 | 711.0 |
| 168.5 | 783.0 |
| 178.5 | 873.0 |
| 188.5 | 986.0 |
| 198.5 | 1124.0 |
| 208.5 | 1293.0 |
| 218.5 | 1503.0 |
| 228.5 | 1759.0 |
| 238.5 | 2069.0 |
| 248.5 | 2455.0 |
| 258.5 | 2924.0 |
| 268.5 | 3642.0 |
| 278.5 | 3713.0 |
| 568.5 | 3713.0 |

Note:
Pressure and temperature values shown are adjusted for instrument uncertainty, and RCS pressure and elevation effects. The pressure shift at 125.2°F results from the change in pressure correction factors applied to the low-range vice the wide-range pressure instrumentation.

**Table 5-2**
**SONGS Unit 2 Cooldown via Control Room Instrumentation**
**RCS Pressure-Temperature Limits Valid until 32 EFPY**

| RCS Temp (°F) | PSIG at 100°F /hr Cooldown | PSIG at 80°F/hr Cooldown | PSIG 60°F/hr Cooldown | PSIG 40°F/hr Cooldown | PSIG 30°F/hr Cooldown | PSIG 20°F/hr Cooldown |
|---|---|---|---|---|---|---|
| 58.5 | 281.0 | 336.0 | 395.0 | 459.0 | 491.0 | 525.0 |
| 68.5 | 320.0 | 369.0 | 423.0 | 482.0 | 513.0 | 545.0 |
| 78.5 | 367.0 | 409.0 | 457.0 | 511.0 | 539.0 | 569.0 |
| 88.5 | 425.0 | 458.0 | 498.0 | 545.0 | 571.0 | 598.0 |
| 95.9 | -- | -- | -- | -- | -- | 625.0 |
| 95.9 | -- | -- | -- | -- | -- | 572.0 |
| 98.5 | 496.0 | 518.0 | 549.0 | 588.0 | 610.0 | 580.0 |
| 101.6 | -- | -- | -- | -- | 625.0 | -- |
| 101.6 | -- | -- | -- | -- | 572.0 | -- |
| 105.7 | -- | -- | -- | 625.0 | -- | -- |
| 105.7 | -- | -- | -- | 572.0 | -- | -- |
| 108.5 | 582.0 | 590.0 | 610.0 | 586.0 | 604.0 | 624.0 |
| 112.5 | 625.0 | | | | | |
| 112.5 | 572.0 | | | | | |
| 118.5 | 634.0 | | | | | |
| 128.5 | 763.0 | | | | | |
| 128.5 | 763.0 | | | | | |
| 138.5 | 836.0 | | | | | |
| 148.5 | 919.0 | | | | | |
| 158.5 | 1019.0 | | | | | |
| 168.5 | 1142.0 | | | | | |

| RCS Temp (°F) | PSIG at 100°F /hr Cooldown | RCS Temp (°F) | PSIG at 100°F /hr Cooldown |
|---|---|---|---|
| 178.5 | 1291.0 | 368.5 | 3151.0 |
| 188.5 | 1474.0 | 378.5 | 3154.0 |
| 198.5 | 1698.0 | 388.5 | 3158.0 |
| 208.5 | 1971.0 | 398.5 | 3163.0 |
| 218.5 | 2304.0 | 408.5 | 3169.0 |
| 228.5 | 2711.0 | 418.5 | 3175.0 |
| 238.5 | 3152.0 | 428.5 | 3183.0 |
| 248.5 | 3152.0 | 438.5 | 3191.0 |
| 258.5 | 3153.0 | 448.5 | 3201.0 |
| 268.5 | 3154.0 | 458.5 | 3213.0 |
| 278.5 | 3154.0 | 468.5 | 3227.0 |
| 288.5 | 3155.0 | 478.5 | 3243.0 |
| 298.5 | 3156.0 | 488.5 | 3262.0 |
| 308.5 | 3158.0 | 498.5 | 3283.0 |
| 318.5 | 3159.0 | 508.5 | 3308.0 |
| 328.5 | 3161.0 | 518.5 | 3509.0 |
| 338.5 | 3163.0 | 528.5 | 3543.0 |
| 340.0 | 3163.0 | 538.5 | 3582.0 |
| 340.1 | 3143.0 | 548.5 | 3627.0 |
| 348.5 | 3145.0 | 558.5 | 3675.0 |
| 358.5 | 3148.0 | 568.5 | 3713.0 |

Note:
Pressure and temperature values shown are adjusted for instrument uncertainty, and RCS pressure and elevation effects. The pressure shift at 112.5°F (at 100°F /hr cooldown) results from the change in pressure correction factors applied to the low-range vice the wide-range pressure instrumentation.

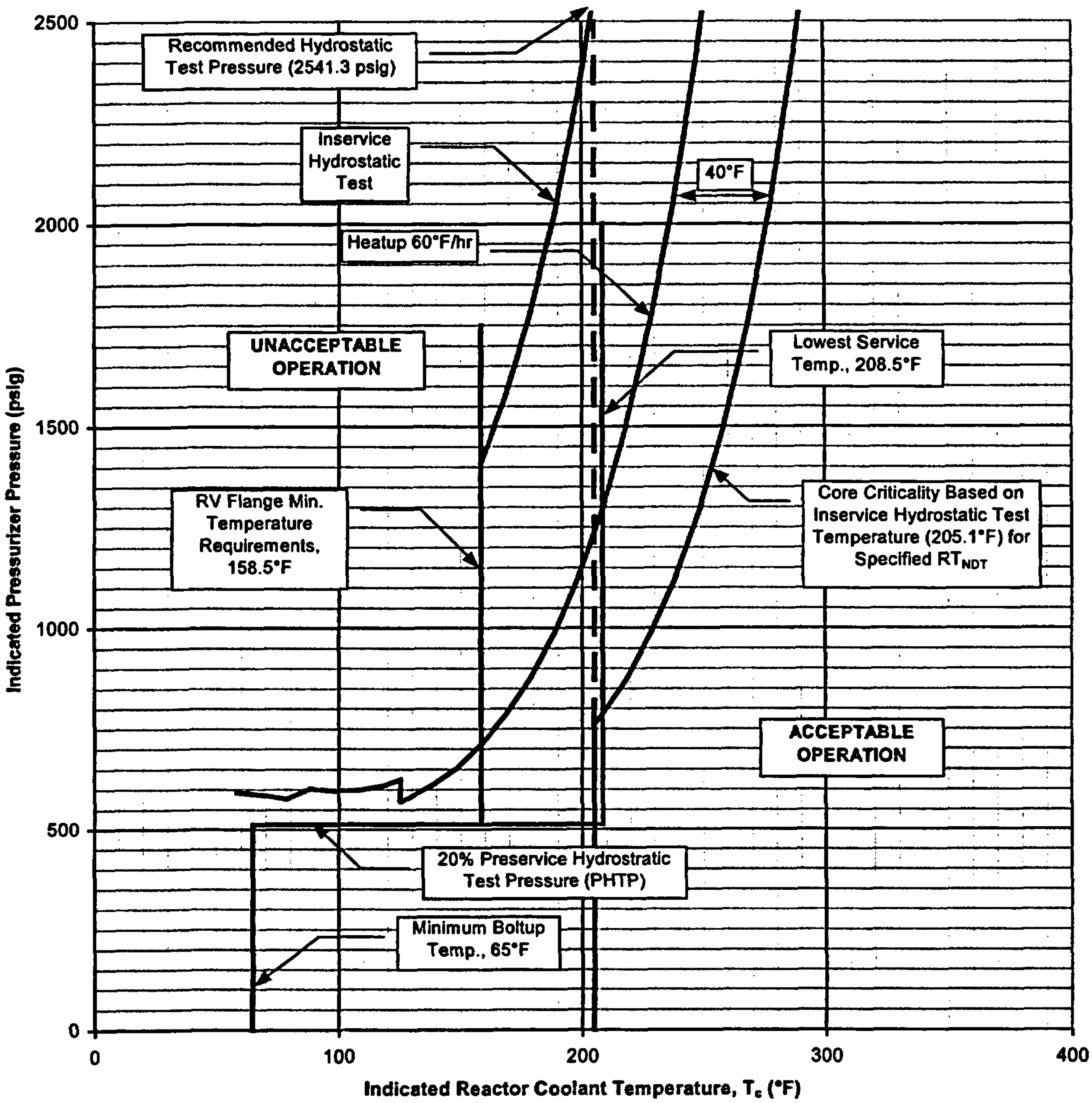
**Table 5-3**
**SONGS Unit 2 Cooldown via Remote Shutdown Panel Instrumentation**
**RCS Pressure-Temperature Limits Valid until 32 EFPY**

| RCS Temp (°F) | PSIG at 100°F /hr Cooldown | PSIG at 80°F/hr Cooldown | PSIG 60°F/hr Cooldown | PSIG 40°F/hr Cooldown | PSIG 30°F/hr Cooldown | PSIG 20°F/hr Cooldown |
|---|---|---|---|---|---|---|
| 58.5 | 0.233 | 288.0 | 347.0 | 411.0 | 443.0 | 477.0 |
| 68.5 | 0.272 | 320.0 | 375.0 | 434.0 | 464.0 | 496.0 |
| 78.5 | 0.319 | 360.0 | 409.0 | 462.0 | 491.0 | 520.0 |
| 88.5 | 0.377 | 409.0 | 450.0 | 497.0 | 522.0 | 549.0 |
| 98.5 | 0.447 | 469.0 | 500.0 | 539.0 | 561.0 | 585.0 |
| 108.5 | 0.534 | 542.0 | 562.0 | 591.0 | 609.0 | 629.0 |
| 118.5 | 0.639 | | | | | |
| 128.5 | 0.768 | | | | | |
| 138.5 | 0.841 | | | | | |
| 148.5 | 0.924 | | | | | |
| 158.5 | 1.024 | | | | | |
| 168.5 | 1.147 | | | | | |
| 178.5 | 1.296 | | | | | |
| 188.5 | 1.479 | | | | | |
| 190.5 | 1.525 | | | | | |
| 190.5 | 1.520 | | | | | |
| 198.5 | 1.698 | | | | | |
| 208.5 | 1.971 | | | | | |
| 218.5 | 2.304 | | | | | |

| RCS Temp (°F) | PSIG at 100°F /hr Cooldown | RCS Temp (°F) | PSIG at 100°F /hr Cooldown |
|---|---|---|---|
| 228.5 | 2.711 | 398.5 | 3.163 |
| 238.5 | 3.152 | 408.5 | 3.169 |
| 248.5 | 3.152 | 418.5 | 3.175 |
| 258.5 | 3.153 | 428.5 | 3.183 |
| 268.5 | 3.154 | 438.5 | 3.191 |
| 278.5 | 3.154 | 448.5 | 3.201 |
| 288.5 | 3.155 | 458.5 | 3.213 |
| 298.5 | 3.156 | 468.5 | 3.227 |
| 308.5 | 3.158 | 478.5 | 3.243 |
| 318.5 | 3.159 | 488.5 | 3.262 |
| 328.5 | 3.161 | 498.5 | 3.283 |
| 338.5 | 3.163 | 508.5 | 3.308 |
| 340 | 3.163 | 518.5 | 3.509 |
| 340.1 | 3.143 | 528.5 | 3.543 |
| 348.5 | 3.145 | 538.5 | 3.582 |
| 358.5 | 3.148 | 548.5 | 3.627 |
| 368.5 | 3.151 | 558.5 | 3.675 |
| 378.5 | 3.154 | 568.5 | 3.713 |
| 388.5 | 3.158 | | |

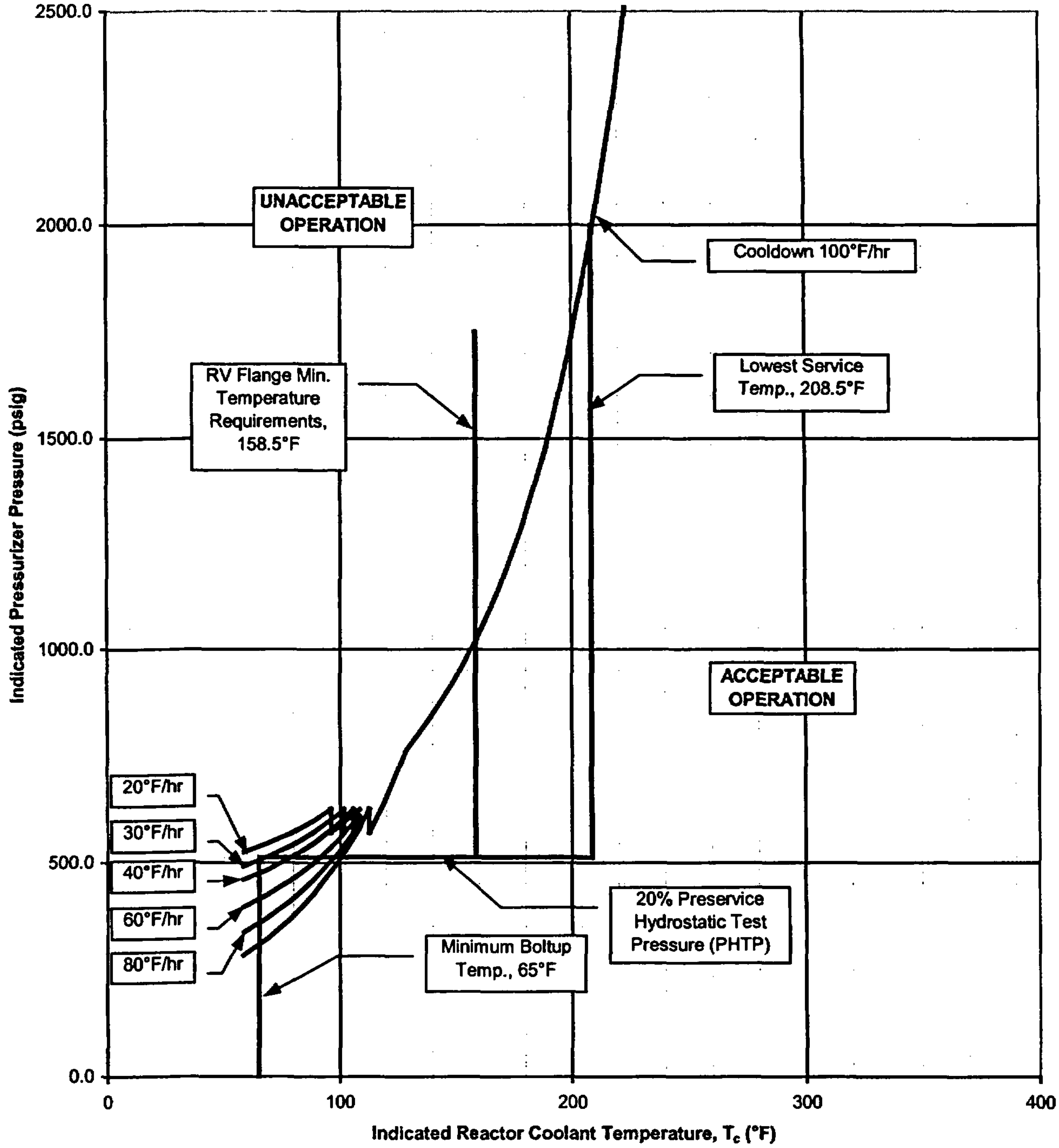
Note:
Pressure and temperature values shown are adjusted for instrument uncertainty, and RCS pressure and elevation effects. The pressure shift at 190.5°F (at 100°F /hr cooldown) results from the change in pressure correction factors applied to the low-range vice the wide-range pressure instrumentation.

**Figure 5-1**
**SONGS Unit 2 RCS Heatup Pressure-Temperature Limits Until 32 EFPY – Normal Operation**



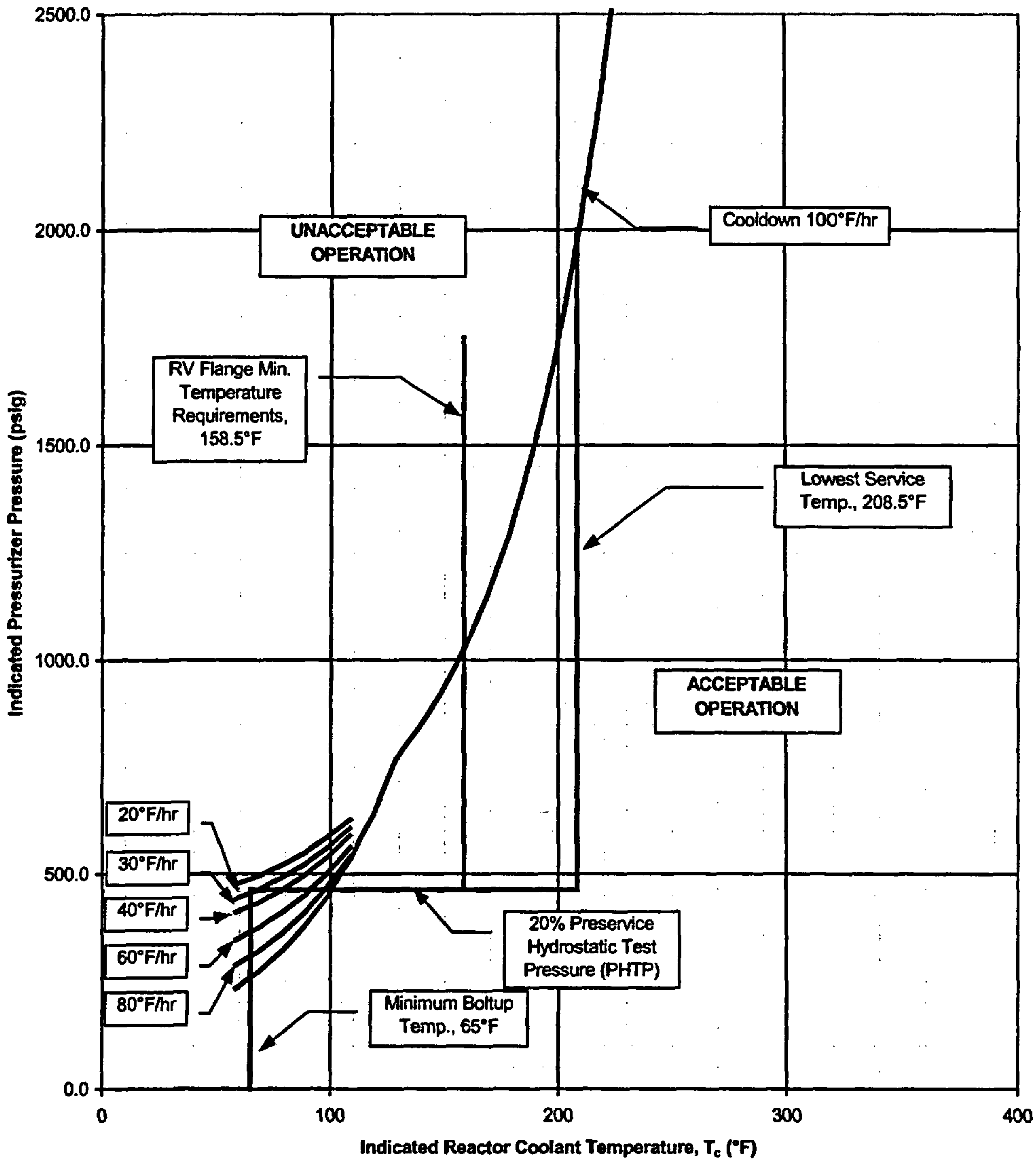

* The more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP should be used in the development of plant P-T limits.

**Figure 5-2**
**SONGS Unit 2 RCS Cooldown Pressure-Temperature Limits Until 32 EFPY – Normal Operation**




* The more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP should be used in the development of plant P-T limits.

**Figure 5-3**
**SONGS Unit 2 RCS Cooldown Pressure-Temperature Limits Until 32 EFPY – Remote Shutdown Panel Operation**




* The more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP should be used in the development of plant P-T limits.

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# 6.0 MINIMUM TEMPERATURE REQUIREMENTS IN THE PRESSURE-TEMPERATURE CURVES

The minimum temperature requirements specified in Appendix G to 10 CFR 50 are applied to the pressure-temperature curves using the NRC-reviewed methodologies as described in Section 6.0 of Reference 1.

The minimum temperature values applied to the pressure-temperature curves of SONG Unit 2 corresponding to 32 Effective Full Power Years (EFPY) are:

**Table 6-1**
**Minimum Temperature Requirements for SONGS Unit 2 at 32 EFPY**

| Requirement | Minimum Temperature |
|---|---|
| Minimum Bolt-Up Temperature | 65°F |
| Minimum Hydrotest Temperature | 205.1°F |
| Lowest Service Temperature | 208.5°F |
| Minimum Flange Limit (NOP) | 158.5°F |
| Minimum Flange Limit (Hydrotest) | 128.5°F |

The lowest service temperature is established for CE NSSSs based on the limiting $RT_{NDT}$ for the reactor coolant pumps.

In the development of pressure-temperature limits for CE NSSSs, the intent is to utilize the more conservative of either the Lowest Service Temperature or the other minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of the preservice hydrostatic test pressure.

The "minimum pressure criteria" specified in 10 CFR 50 Appendix G serves as a regulatory breakpoint in the development of pressure-temperature limits and is defined as twenty percent of pre-service hydrostatic test pressure. For CE NSSS plants, the preservice hydrostatic test pressure is defined as 1.25 times the design pressure. The function of minimum pressure in the development of pressure-temperature limits is to provide a transition between the various temperature only based pressure-temperature limits, such as minimum bolt up and the lowest service temperature or flange limits.

For SONGS Unit 2, the minimum pressure is calculated as follows:

Minimum Pressure, Uncorrected:

$= (1.25 \times \text{Design Pressure}) \times 0.20 = (1.25 \times 2500 \text{ psia}) \times 0.20 = 625 \text{ psia}$

With pressure corrections due to flow, elevation, and instrument uncertainties, the limiting minimum pressure for SONGS-2 until 32 EFPY becomes:

**Table 6-2**
**Limiting Indicated Minimum Pressure Requirements for SONGS Unit 2 until 32 EFPY**

| **For the Control Room:** | **Minimum Pressure Requirement*** |
|---|---|
| $T_{RCS} = 340.0°F$ | $P_{RCS} = 527.2$ psia |
| $T_{RCS} > 340.0°F$ | $P_{RCS} = 507.2$ psia |
| **For the Remote Shutdown Panel** | **Minimum Pressure Requirement*** |
| $T_{RCS} = 340.0°F$ | $P_{RCS} = 478.7$ psia |
| $T_{RCS} > 340.0°F$ | $P_{RCS} = 458.7$ psia |

* Note: The limiting temperature of 340°F results from the assumed transition from two-reactor coolant pump operation to three-reactor coolant pump operation.

# 7.0 APPLICATION OF SURVEILLANCE DATA TO ADJUSTED REFERENCE TEMPERATURE CALCULATIONS

Post-irradiation surveillance capsule test results for SONGS Unit 2 are given in References 4 and 15. The test results meet the credibility criteria of Regulatory Guide 1.99, Revision 2. The criteria were met as follows:

- The surveillance program plate or weld duplicates the controlling reactor vessel beltline material in terms of ART,
- Charpy data scatter does not cause ambiguity in the determination of the 30 ft-lb shift,
- The measured shifts are consistent with the predicted shifts,
- The capsule irradiation temperature is comparable to that of the vessel, and
- Correlation monitor data are available and are consistent with the known data for that material.

The data supporting the credibility analysis are presented in Reference 4.

Credible surveillance data were used to refine the chemistry factor and the margin term in accordance with the methodology prescribed in Position 2.1 of Regulatory Guide 1.99, Revision 2. The chemistry factor is calculated as shown in Table 7-1 (reproduced from Section 4, Table 4-4). The credibility test for the surveillance capsule measurement is shown in Table 7-2 (reproduced from Section 4, Table 4-5). In addition, from Table 7.4 of Reference 4, the measured shift for the correlation monitor material from the 263-degree capsule was 151.6°F versus the predicted shift of 149.6°F. Therefore, the correlation monitor material meet the credibility test to be (well) within the scatter band of the database for that material.

The derived chemistry factor for the plate from Table 7-1 is 68.21°F, and that for the weld is 15.26°F. The margin on the shift, $\sigma_?$, is 17°F and 28°F for the plate and the weld, respectively. The value of the margin on initial $RT_{NDT}$, $\sigma_\Delta$, is zero for the plate and the weld because there are measured values for both the plate and the weld. The total margin is then taken as $1\sigma_\Delta$ as prescribed in Position 2.1 of Regulatory Guide 1.99, Revision 2, when the surveillance data have been shown to be credible, and the margin on the shift can be halved, i.e., margin = $2\,(\sigma_i^2 + (\sigma_\Delta/2)^2\,)^{1/2} = \sigma_\Delta$ when $\sigma_i$ is 0.

The calculation of adjusted reference temperature, ART, using the credible surveillance results from SONGS Unit 2 is described in Section 4.

**Table 7-1**
**Calculation of Chemistry Factor Values for the Surveillance Plate and Weld Materials**

| Location | 97-Degree Capsule | 263-Degree Capsule | Unirradiated $RT_{NDT}$ | Sum | Chemistry Factor (°F) |
|---|---|---|---|---|---|
| Fluence ($x10^{19}$ n/cm$^{2)}$ | 0.507 | 1.637 | | | |
| Fluence factor (ff) | 0.810 | 1.136 | | | |
| $(ff)^2$ | 0.6561 | 1.2905 | | 1.947 | |
| **PLATE C-6404-2** | | | | | |
| $RT_{NDT}$ | 68.6 | 115.1 | 27.5 | | |
| $\Delta RT_{NDT}$ | 41.1 | 87.6 | | | |
| ff*$\Delta RT_{NDT}$ | 33.31 | 99.51 | | 132.82 | 68.21 |
| **WELD 9-203** | | | | | |
| $RT_{NDT}$ | -49.2 | -29.9 | -53.2 | | |
| $\Delta RT_{NDT}$ | 4.0 | 23.3 | | | |
| ff*$\Delta RT_{NDT}$ | 3.24 | 26.47 | | 29.71 | 15.26 |

**Table 7-2**
**Credibility Test of the Calculated Chemistry Factors**

| Material | $\sigma_\Delta$ | Chemistry Factor(°F) | Fluence ($x10^{19}$ n/cm$^2$) | ff | CF*ff (°F) ($\Delta RT_{NDT}$) | $\Delta RT_{NDT}+\sigma_\Delta$ (°F) | $\Delta RT_{NDT}-\sigma_\Delta$ (°F) | Measured $\Delta RT_{NDT}$ (°F) |
|---|---|---|---|---|---|---|---|---|
| Plate | 17 | 68.21 | 0.507 | 0.810 | 55.28 | 72.28 | 38.28 | 41.1 |
| C-6404-2 | | | 1.637 | 1.136 | 77.48 | 94.48 | 60.48 | 87.6 |
| Weld | 28 | 15.26 | 0.507 | 0.810 | 12.37 | 40.37 | -15.63 | 4.0 |
| 9-203 | | | 1.637 | 1.136 | 17.33 | 45.33 | -10.67 | 23.3 |

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# APPENDIX A

## Technical Specification
## References to the PTLR
## (Provided by SCE)

The P-T limits information contained in Appendix A is extracted from the PTLR and displayed in a format similar to SCE's existing Technical Specification. This Appendix provides a convenient centralized location for information relocated from the Technical Specification to the PTLR in the format familiar to SCE's Operation Group.

This Appendix is currently a sample representation. SCE will replace this sample Appendix with final information.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.3 RCS Pressure and Temperature (P/T) Limits

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

With the reactor vessel head bolts tensioned*, the Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on *Figures A-1 through A-3 and Table A-1* during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than or equal to 65 $^{\circ}$F.

b. *A maximum cooldown of 100 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 90 $^{\circ}$F for normal operation. A maximum cooldown of 100 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 97 $^{\circ}$F for remote shutdown operation.*

c. *A maximum cooldown of 80 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 84 $^{\circ}$F for normal operation. A maximum cooldown of 80 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 93 $^{\circ}$F for remote shutdown operation.*

d. *A maximum cooldown of 60 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 75 $^{\circ}$F for normal operation. A maximum cooldown of 60 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 85 $^{\circ}$F for remote shutdown operation.*

e. *A maximum cooldown of 40 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than or equal to 65 $^{\circ}$F for normal operation. A maximum cooldown of 40 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater than 70 $^{\circ}$F for remote shutdown operation.*

f. *A maximum cooldown of 30 $^{\circ}$F in any 1-hour period with RCS cold leg temperature greater the 65 $^{\circ}$F for remote shutdown operation.*

g. *A maximum temperature change of 10 $^{\circ}$F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.*

h. A minimum temperature of 65 $^{\circ}$F to tension reactor vessel head bolts.

With the reactor vessel head bolts detensioned, the Reactor Coolant System (except the pressurizer) temperature shall be limited to a maximum heatup or cooldown of 60 $^{\circ}$F in any 1-hour period.

* With the reactor vessel head bolts detensioned, RCS cold leg temperature may be less than 65 $^{\circ}$F.

3/27/03 3.4.3

APPLICABILITY: At all times.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A.<br>--------NOTE--------<br>Required Action A.2 shall be completed whenever this Condition is entered.<br>----------------<br>Requirements of LCO not met in MODE 1, 2, 3, or 4. | A.1<br>Restore parameter(s) to within limits.<br><br><u>AND</u><br>A.2<br>Determine RCS is acceptable for continued operation. | 30 minutes<br><br>72 hours |
| B.<br>Required Action and associated Completion Time of Condition A not met. | B.1<br>Be in MODE 3.<br><br><u>AND</u><br>B.2<br>Be in MODE 5 with RCS pressure < 500 psia. | 6 hours<br><br>36 hours |
| C.<br>--------NOTE--------<br>Required Action C.2 shall be completed whenever this Condition is entered.<br>----------------<br>Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4. | C.1<br>Initiate action to restore parameter(s) to within limits.<br><br><u>AND</u><br>C.2<br>Determine RCS is acceptable for continued operation. | Immediately<br><br>Prior to entering MODE 4 |

3/27/03 3.4.3 Continued

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ----------------NOTE------------------<br>Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing.<br>------------------------------------------<br><br>Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in *Figures A-1 through A-3.* | 30 minutes |
| The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update *the PTLR.* | In accordance with requirements of 10CFR 50 Appendix H |

Sample

3/27/03 3.4.3 Continued

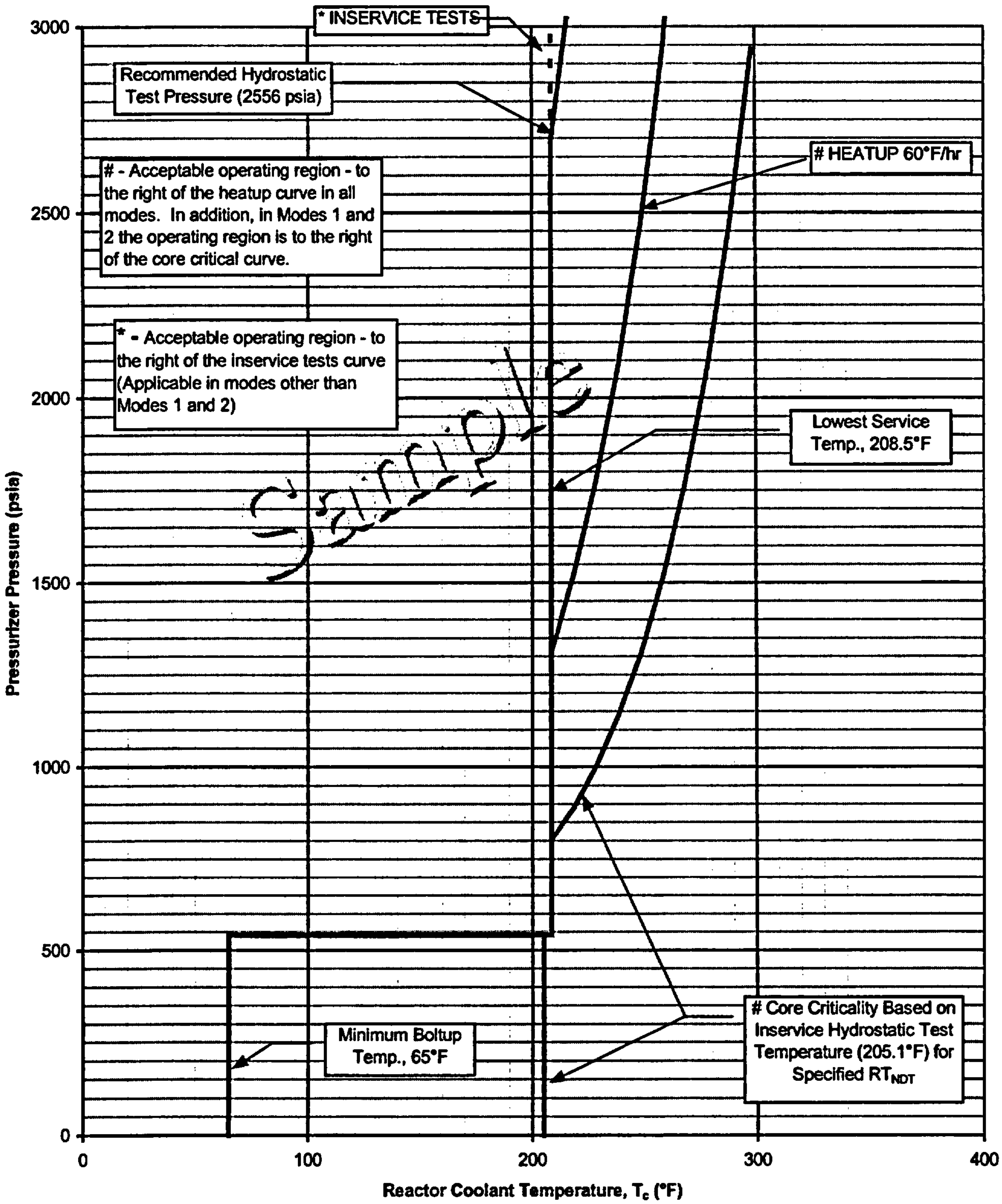


**FIGURE A-1**
**SONGS Unit 2 RCS Heatup P-T Limits until 32 EFPY**
**Normal Operation**

3/27/03 3.4.3 Continued

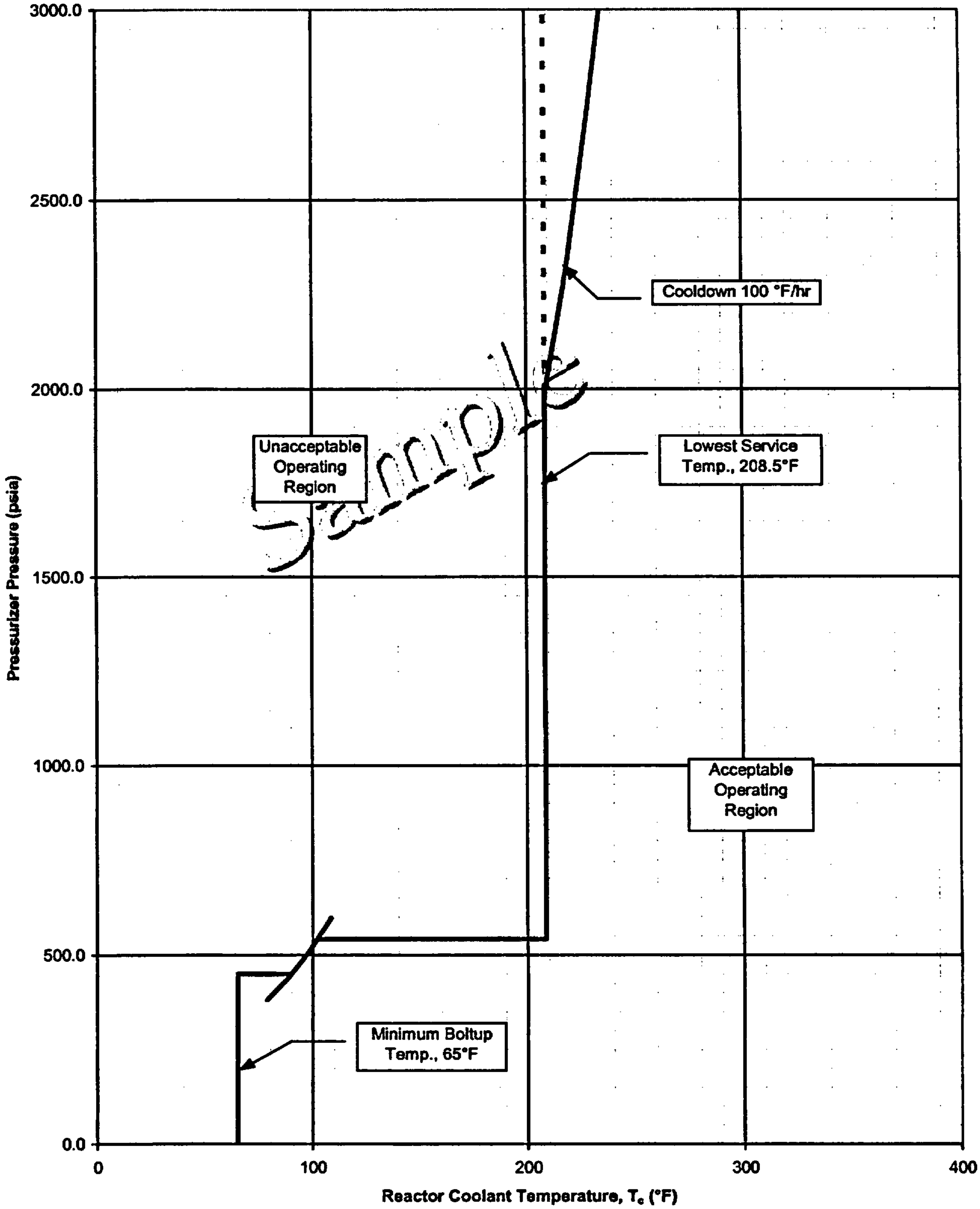


**FIGURE A-2**
**SONGS Unit 2 RCS Cooldown P-T Limits until 32 EFPY**
**Normal Operation**

3/27/03

3.4.3 Continued

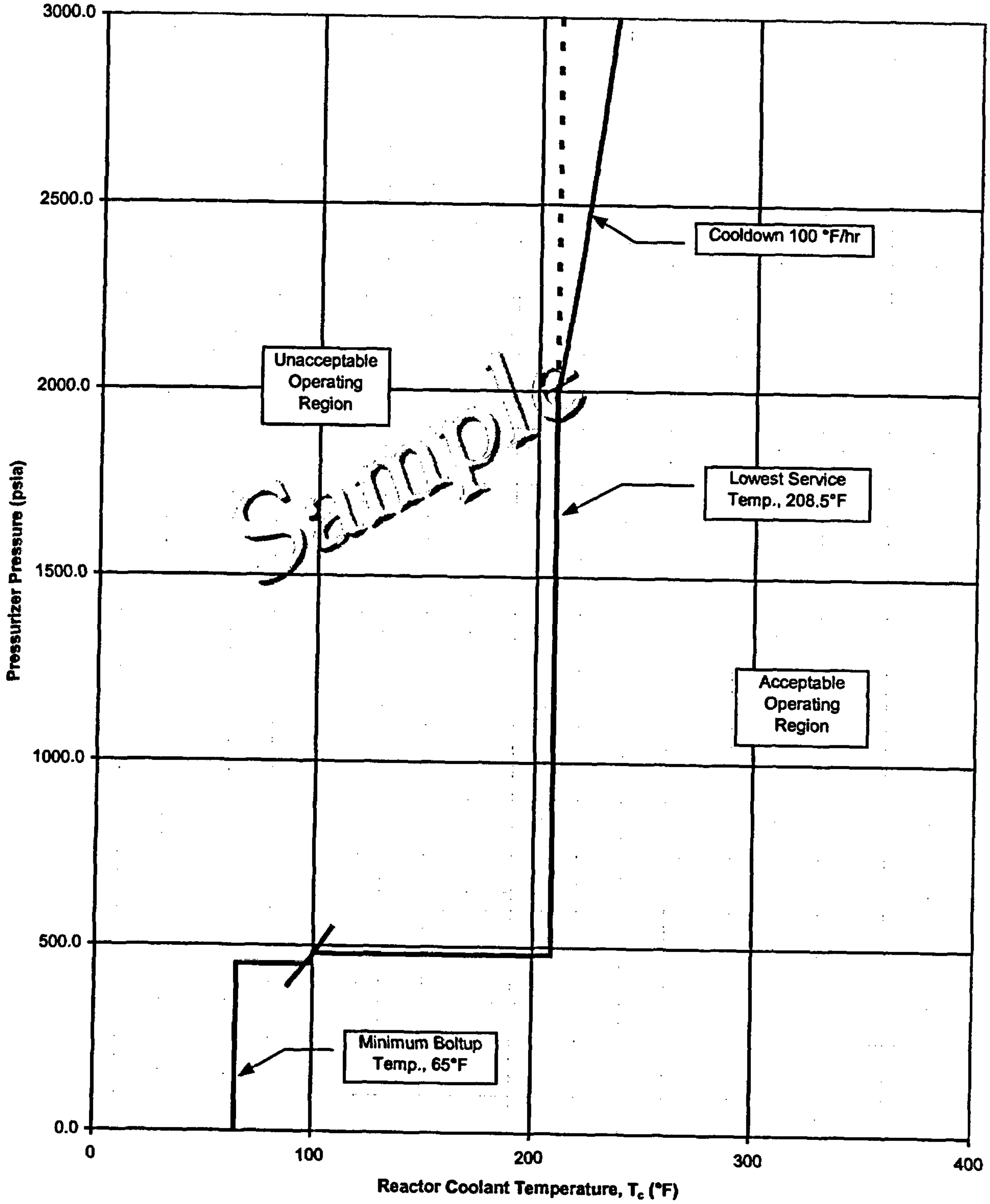


**FIGURE A-3**
**SONGS Unit 2 RCS Cooldown P-T Limits until 32 EFPY**
**Remote Shutdown Panel Operation**

3/27/03 3.4.3 Continued

TABLE A-1

Low Temperature RCS Overpressure Protection Range

| Operating Period, EFPY | Cold Leg Temperature, °F | |
|---|---|---|
| | During Heatup | During Cooldown |
| Until *32* (Normal Operation) | ≤ *219* | ≤ *191* |
| Until *32* (Remote shutdown Operation) | * | ≤ *191* |



* Heatup operations are not normally performed from the Remote Shutdown Panel

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.12.1 Low Temperature Overpressure Protection (LTOP) System

RCS Temperature ≤ *219°F*

---

LCO 3.4.12.1 No more than two high pressure safety injection pumps shall be OPERABLE, the safety injection tanks shall be isolated or depressurized to less than the limit specified in *Figure A-2* and at least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (Unit 2, PSV9349) with:

1) A lift setting of 406 ± 10 psig,

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

or,

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



APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to the enable temperatures specified in Table *A-1*,

MODE 5, and

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

------------------NOTE------------------

SIT isolation or depressurization to less than the Figure *A-2* limit is only required when SIT pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in Figure *A-1* and Figure *A-2*.

---

3/27/03 3.4.12.1 Continued

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A.<br>With more than two HPSI pumps capable of injecting into the RCS. | A.1<br>Initiate action to verify a maximum of two HPSI pumps capable of injecting into the RCS. | Immediately |
| B.<br>SIT pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in Figure *A-1* and Figure *A-2*. | B.1<br>Isolate affected SIT. | 1 hour |
| C.<br>Required Action and associated Completion Time of Condition B not met. | C.1<br>Depressurize affected SIT to less than the maximum RCS pressure for existing cold leg temperature allowed in Figure *A-1* and Figure *A-2*. | 12 hours |
| D.<br>With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (Unit 2 valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed. | D.1<br>Open the closed valve(s).<br><br><u>OR</u><br>D.2<br>Power-lock open the OPERABLE SDCS Relief Valve isolation valve pair. | 24 hours<br><br><br><br>24 hours |

(continued)

3/27/03

3.4.12.1 Continued

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| E.<br>SDCS Relief Valve inoperable.<br><br><u>OR</u><br><br>Required Action and associated Completion Time of Condition A, C, or D not met.<br><br><u>OR</u><br><br>LTOP System inoperable for any reason other than Condition A, C, or D. | E.1<br>Reduce $T_{avg}$ to less than 200 °F, depressurize RCS and establish RCS vent of ≥ 5.6 square inches. | 8 hours |

3/27/03 3.4.12.1 Continued

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ---------------NOTE--------------- A HPSI pump is secured by verifying that its motor circuit breaker is not racked-in, or its discharge valve is locked closed. The requirement to rack out the HPSI pump breaker is satisfied with the pump breaker racked out to its disconnected or test position. ------------------------------------ Verify a maximum of two HPSI pumps are capable of injecting into the RCS. | 12 hours |
| ---------------NOTE--------------- Required to be performed when complying with the LCO 3.4.12.1 Note. ------------------------------------ Verify each SIT is isolated or depressurized less than the limit specified in Figure *A-2*. | 12 hours |
| Verify RCS vent ≥ 5.6 square inches is open when in use for overpressure protection. | 12 hours for unlocked open vent valve(s)<br><br>AND<br><br>31 days for locked, sealed, or otherwise secured open vent valve(s), or open flanged RCS penetrations |



(continued)

3/27/03 3.4.12.1 Continued

| SURVEILLANCE | FREQUENCY |
|---|---|
| ---------------NOTES--------------<br><br>1. Only required to be performed when the SDCS Relief Valve isolation valve pair is inoperable.<br><br>2. The power-lock open requirement is satisfied either with the AC breakers open for valve pair 2HV9337 and 2HV9339 or the inverter input and output breakers open for valve pair 2HV9377 and 2HV9378, whichever valve pair is OPERABLE.<br>------------------------------<br>Verify the OPERABLE SDCS Relief Valve isolation valve pair (Unit 2 valve pair 2HV9337 and 2HV9339, or valve pair 2HV9377 and 2HV9378) is in the power-lock open condition. | 12 hours |
| Verify that SDCS Relief Valve isolation valves (Unit 2) 2HV9337, 2HV9339, 2HV9377, and 2HV9378 are open when the SDCS Relief Valve is used for overpressure protection. | 72 hours |
| Verify SDCS Relief Valve Setpoint. | In accordance with the Inservice Testing Program |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.12.2 Low Temperature Overpressure Protection (LTOP) System

RCS Temperature > ***219°F***

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

a. The Shutdown Cooling System Relief Valve (Unit 2, PSV9349) with:

1) A lift setting of 406 ± 10 psig,

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

or,

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



***APPLICABILITY:MODE 4 when the temperature of all RCS cold legs are greater than the enable temperatures specified in Table* A-1.**

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

1. The lift setting pressure of the pressurizer code safety valve shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.
2. The SDCS Relief Valve lift setting assumes valve temperatures less than or equal to 130 °F.

3/27/03 3.4.12.2 Continued

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A.<br>No pressurizer code safety valves OPERABLE.<br><br>AND<br><br>The SDCS Relief Valve INOPERABLE. | A.1<br>Be in MODE 5 and vent the RCS through a greater than or equal to 5.6 square inch vent. | 8 hours |
| B.<br>With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed. | B.1<br>Open the closed valve(s).<br><br>OR<br>B.2<br>Power-Lock open the OPERABLE SDCS Relief Valve isolation valve pair. | 24 hours<br><br><br><br>24 hours |



SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ---------------Note---------------<br>Only required when the SDCS Relief Valve is being used for overpressure protection.<br>--------------------------------<br><br>Verify that the SDCS Relief Valve isolation valves (Unit 2) 2HV9337, 2HV9339, 2HV9377, and 2HV9378 are open. | 72 hours |
| Verify relief valve setpoint. | In accordance with the Inservice Testing Program |

WCAP-16005-NP, Rev. 0

Westinghouse Non-Proprietary Class 3

Westinghouse

Westinghouse Electric Company, LLC
2000 Day Hill Road
Windsor, Connecticut 06095-0500

## ENCLOSURE 4

## JUSTIFICATION FOR EXEMPTION FOR ALTERNATE METHOD OF CALCULATING THE THERMAL STRESS INTENSITY FACTOR ($K_{IT}$)

## SAN ONOFRE UNIT 2

# JUSTIFICATION FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50.60 ALTERNATE METHOD FOR CALCULATING $K_{IT}$

In accordance with 10 CFR 50.12(a), SCE requests an exemption from the regulations of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation." The exemption request would allow SONGS Unit 2 to use an alternate methodology to calculate the thermal stress intensity factor ($K_{IT}$) used in the development of pressure-temperature curves, in lieu of the methodology cited in ASME Boiler and Pressure Vessel Code, Appendix G.

SONGS Unit 2 uses a methodology provided by Westinghouse to generate the pressure-temperature curves and limits. This Westinghouse methodology uses an alternate approach to ASME Appendix G to calculate the thermal stress intensity factor. The methodology compared $K_{IT}$ results obtained from the Westinghouse methodology to those obtained from the ASME Appendix G methodology, using the details of a sample reactor vessel. The Non-Proprietary Version of the Westinghouse methodology "The Technical Methodology Paper Comparing ABB Combustion Engineering Pressure Temperature Curve to ASME Section III, Appendix G" was presented in the submittal by Indian Point 3 Nuclear Power Plant, Docket No. 50-286, Proposed Exemption from Requirements of 10 CFR 50.60 to Utilize Alternate Methodology to Determine $K_{IT}$ and accepted by the NRC (TAC NO. M99928).

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.

## Justification for Exemption

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10 CFR 50 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 satisfied, as described below.

1. The requested exemption is authorized by law.

   The NRC is authorized by law to grant this exemption. 10 CFR 50.60 states 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 thermal stress intensity factor. Specifically, this methodology uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate $K_{IT}$. The results of this methodology are comparable to the results obtained using the ASME Appendix G methodology. 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, the 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.

<u>Conclusion</u>

SCE concludes that the use of the Westinghouse alternate methodology to calculate the thermal stress intensity factor ($K_{IT}$) provides greater operational flexibility while providing adequate protection of the reactor vessel against non-ductile failure.

**ENCLOSURE 5**

**JUSTIFICATION FOR EXEMPTION FOR ALTERNATE METHOD FOR THE USE OF ASME CODE CASE N-640**

**SAN ONOFRE UNIT 2**

## JUSTIFICATION FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50.60 USE OF ASME CODE CASE N – 640

In accordance with 10 CFR 50.12(a), SCE requests an exemption from the requirements of 10 CFR 50, Appendix G, to use American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1" to support the methodology used in the PTLR for determining pressure-temperature limits. Paragraph (IV)(A)(2)(b) of 10 CFR 50, Appendix G, requires that pressure-temperature limits be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code. Pressure-temperature limits obtained using ASME Code Case N-640 do not meet the requirements of paragraph (IV)(A)(2)(b). Use of ASME Code Case N-640 was presented in the submittal by Fort Calhoun Station and accepted by the NRC (TAC NO. MB3606).

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.

SCE believes that the exemption requirements of 10 CFR 50.12 are satisfied. The exemption is requested for the life of the plant or until incorporation into 10 CFR 50 Appendix G.

<u>Justification for Exemption</u>

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10 CFR 50, 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 satisfied, as described below.

1. The requested exemption is authorized by law.

   The NRC is authorized by law to grant this exemption. 10 CFR 50.60 states that the use of alternate 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.

Code Case N-640 permits use of $K_{IC}$, fracture toughness curve shown on ASME XI, Appendix A, Figure A-2200-1, in lieu of the $K_{IA}$, fracture toughness curve from ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The exemption request involves only a change of the fracture toughness curve used for development of the P-T curves from $K_{IA}$ to $K_{IC}$. The other margins involved with the ASME XI, Appendix G, process of determining P-T limit curves remain unchanged.

Use of the $K_{IC}$ curve in determining the lower bound fracture toughness in the development of a P-T operating limits curve is more technically correct than the $K_{IA}$ curve. The $K_{IC}$ curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The use of the initial conservatism of the $K_{IA}$ curve when the curve was codified in 1974 was necessary due to the limited knowledge for reactor pressure vessel materials. Since 1974, additional knowledge has been gained about reactor pressure vessel materials, which demonstrates that the lower bound on fracture toughness provided by the $K_{IA}$ curve is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure. In addition, P-T curves based on the $K_{IC}$ curve will enhance overall plant safety by opening the P-T operating window, especially in the region of low temperature operations. 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) and (iii).

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.

As described above, the existing approach for determining the P-T limits was conservatively developed based on the level of knowledge existing in 1974. Since 1974, the level of knowledge in this area has been greatly expanded. This increased knowledge permits relaxation of the current ASME XI, Appendix G, requirements as provided by ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

10 CFR 50.12(a)(2)(iii) – Compliance would result in undue hardships or costs that are significantly in excess of those contemplated when the regulation was developed.

The reactor coolant system pressure-temperature operating window is defined by the pressure-temperature operating and test limit curves developed in accordance with the ASME Code, Section XI, Appendix G. Continued operation of SONGS Unit 2, without the use of ASME Code Case N-640, could unnecessarily require plant operators to maintain a high reactor pressure vessel temperature in a limited pressure-temperature operating window during pressure tests. The restriction could also subject personnel conducting inspections in primary containment to potential steam vapor hazards. Implementation of the proposed pressure-temperature limits, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety and could potentially reduce the personnel risks associated with performing inspections at higher temperatures by allowing inspections in primary containment to be conducted at lower coolant temperatures. Based on the above, maintaining pressure-temperature limits based on the $K_{IA}$ fracture toughness curve from ASME Code, Section XI, Appendix G, Figure G-2200-1, as the lower bound for fracture toughness, constitutes an unnecessary burden that can be alleviated by allowing the use of ASME Code Case N-640.

Conclusion

The exemption requirements of 10 CFR 50.12 are satisfied because special circumstances are present, as described in 10 CFR 50.12(a)(2)(ii) and (iii), to warrant granting the exemption. The specified requirements of 10 CFR 50, Appendix G, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. American Society of Mechanical Engineers Section XI, Appendix G, requirements were conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. SCE believes this increased knowledge permits relaxation of the ASME Section XI, Appendix G, requirements by application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and the NRC regulations for ensuring an acceptable margin of safety.