ENCLOSURE 1, ATTACHMENT 5

WCAP-16835-NP, Palo Verde Nuclear Generating Station Units 1, 2, and 3; Basis for RCS Pressure and Temperature Limits Report, June 2008

WCAP-16835-NP
Revision 0

June 2008

Palo Verde Nuclear Generating Station Units 1, 2 and 3; Basis for RCS Pressure and Temperature Limits Report

Westinghouse

WCAP-16835, Revision 0

Palo Verde Nuclear Generating Station Units 1, 2 and 3; Basis for RCS Pressure and Temperature Limits Report

June 2008

S. T. Byrne

F. P. Ferraraccio

B. R. Ganta

V. A. Paggen

Electronically Approved Records are authenticated in the Westinghouse Electronic Document Management System.

© 2008 Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

All Rights Reserved

LEGAL NOTICE

This report was prepared as an account of work performed by Westinghouse Electric Company LLC. Neither Westinghouse Electric Company LLC, nor any person acting on its behalf:

A. Makes any warranty or representation, express or implied including the warranties of fitness for a particular purpose or merchantability, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this report.

COPYRIGHT NOTICE

This report has been prepared by Westinghouse Electric Company LLC and bears a Westinghouse Electric Company copyright notice. Information in this report is the property of and contains copyright material owned by Westinghouse Electric Company LLC and/or its subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document and the material contained therein in strict accordance with the terms and conditions of the agreement under which it was provided to you.

As a sponsor of this task, you are permitted to make the number of copies of the information contained in this report that are necessary for your internal use in connection with your implementation of the report results for your plant(s) in your normal conduct of business. Should implementation of this report involve a third party, you are permitted to make the number of copies of the information contained in this report that are necessary for the third party's use in supporting your implementation at your plant(s) in your normal conduct of business if you have received the prior, written consent of Westinghouse Electric Company LLC to transmit this information to a third party or parties. All copies made by you must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

The NRC is permitted to make the number of copies beyond those necessary for its internal use that are necessary in order to have one copy available for public viewing in the NRC public document room if the number of copies submitted is insufficient for this purpose, subject to the applicable federal regulations regarding restrictions on public disclosure to the extent such information has been identified as proprietary. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

TABLE OF CONTENTS

LIST OF TABLES v
LIST OF FIGURES vi
ABSTRACT vii

1.0 VESSEL NEUTRON FLUENCE 1-1
1.1 OVERVIEW 1-1
1.2 DISCRETE ORDINATES ANALYSIS 1-1
1.3 VESSEL FLUENCE ANALYSIS 1-3
1.4 NEUTRON DOSIMETRY 1-3
1.5 CALCULATIONAL UNCERTAINTIES 1-5

2.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM 2-1
2.1 TEST MATERIAL SELECTION 2-1
2.1.1 Plate Material Selection 2-1
2.1.2 Weld Material Selection 2-1
2.2 TEST SPECIMENS 2-2
2.2.1 Type and Quantity 2-2
2.2.2 Unirradiated Specimens 2-2
2.2.3 Irradiated Specimens 2-3
2.3 SURVEILLANCE SPECIMEN IRRADIATION 2-3
2.3.1 Specimen Encapsulation 2-3
2.3.2 Flux and Temperature Measurement 2-4
2.3.3 Irradiation Specimen Location 2-5
2.3.4 Surveillance Capsule Withdrawal Schedule 2-5

3.0 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM LIMITS 3-1
3.1 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM 3-1
3.1.1 Introduction 3-1
3.1.2 LTOP Technical Specifications 3-1
3.2 BASIS FOR LTOP SYSTEM LIMITS 3-1
3.2.1 Peak Transient Pressures 3-2
3.2.2 Applicable Pressure-Temperature Limits 3-4

4.0 ADJUSTED REFERENCE TEMPERATURE 4-1
4.1 BACKGROUND 4-1
4.2 RESULTS 4-1
4.2.1 Fluence Calculation 4-1
4.2.2 Chemistry Factor Calculation 4-2
4.2.3 Calculation of ART for the Limiting Plates at 1/4T and 3/4T 4-2
4.3 ANALYSIS DETAILS 4-2
4.3.1 Selection of Representative and Limiting Cases 4-2
4.3.2 Calculation of Fluence at 1/4T and 3/4T 4-3
4.4 LIMITING ADJUSTED REFERENCE TEMPERATURES 4-3

TABLE OF CONTENTS, CONTINUED

5.0 RCS PRESSURE-TEMPERATURE LIMITS 5-1

5.1 STRESS INTENSITY FACTOR 5-1

5.1.1 General 5-1

5.1.2 Pressure-Temperature Limits Calculation 5-1

5.2 FRACTURE TOUGHNESS CRITERIA 5-2

5.3 TRANSIENT PRESSURE-TEMPERATURE LIMITS 5-3

5.4 CRACK TIP PRESSURE-TEMPERATURE LIMITS 5-3

5.5 HYDROSTATIC AND LEAK TEST PRESSURE-TEMPERATURE LIMITS 5-4

5.6 MINIMUM BOLTUP TEMPERATURE 5-4

5.7 MINIMUM PRESSURE REQUIREMENT 5-4

5.8 LOWEST SERVICE TEMPERATURE 5-4

5.9 FLANGE LIMITS 5-5

5.10 MINIMUM TEMPERATURE REQUIREMENTS 5-5

5.11 TEMPERATURE REQUIREMENTS FOR NORMAL OPERATION 5-6

5.12 LTOP ENABLE TEMPERATURE LIMITS 5-6

5.13 PRESSURE AND TEMPERATURE CORRECTON FACTORS 5-7

5.14 SUMMARY OF RCS PRESSURE-TEMPERATURE LIMITS 5-8

6.0 MINIMUM TEMPERATURE REQUIREMENTS 6-1

7.0 APPLICATION OF SURVEILLANCE CAPSULE DATA 7-1

8.0 REFERENCES 8-1

LIST OF TABLES

1-1 Calculated Neutron Fluence at Core Midplane through 32 EFPY 1-8
1-2 Calculated Azimuthal Variation of Neutron Fluence for Unit 1 1-8
1-3 Calculated Azimuthal Variation of Neutron Fluence for Unit 2 1-9
1-4 Calculated Azimuthal Variation of Neutron Fluence for Unit 3 1-10
1-5 Comparison of Measured, Calculated and Best Estimate Reaction Rates for Unit 2 1-10
1-6 Comparison of Calculated and Best Estimate Neutron Flux for Unit 2 1-11
1-7 Comparison of Measured to Calculated Reaction Rate Ratios for Unit 2 1-11
1-8 Comparison of Best Estimated to Calculated Neutron Flux Ratios for Unit 2 1-11

2-1 Base Metal Materials Selected for Surveillance Program 2-6
2-2 Weld Metal Materials Selected for Surveillance Program 2-6
2-3 Type and Quantity of Specimens for Irradiation Exposure 2-6
2-4 Surveillance Capsule Assembly Removal Schedule through 32 EFPY 2-6

3-1 RCS Heatup and Cooldown Rate Limits through 32 EFPY 3-4

4-1 Summary of Limiting ART and RT_{PTS} through 32 EFPY 4-4
4-2 ART Input Values for Unit 1 Beltline Materials 4-4
4-3 ART Input Values for Unit 2 Beltline Materials 4-4
4-4 ART Input Values for Unit 3 Beltline Materials 4-5
4-5 Predicted ART Values for Unit 1 Beltline Materials through 32 EFPY 4-5
4-6 Predicted ART Values for Unit 2 Beltline Materials through 32 EFPY 4-5
4-7 Predicted ART Values for Unit 3 Beltline Materials through 32 EFPY 4-6
4-8 Fluence and Fluence Factors at 1/4T and 3/4T through 32 EFPY 4-6

5-1 RCS Pressure and Temperature Heatup Limits through 32 EFPY 5-8
5-2 RCS Pressure and Temperature Cooldown Limits through 32 EFPY 5-9
5-3 LTOP Enable Temperature Limits through 32 EFPY 5-9
5-4 Allowable In-service Hydrostatic Test Pressure 5-10
5-5 Core Critical Limits for Heatup at 75°F/hr 5-10
5-6 Core Critical Limits for Cooldown at 100°F/hr 5-10

6-1 Minimum Indicated RCS Pressure through 32 EFPY 6-1
6-2 Minimum Indicated RCS Temperature through 32 EFPY 6-1

7-1 Chemistry Factors for Unit 1 Surveillance Plates and Weld Materials 7-2
7-2 Chemistry Factors for Unit 2 Surveillance Plates and Weld Materials 7-2
7-3 Chemistry Factors for Unit 3 Surveillance Plates and Weld Materials 7-3
7-4 Credibility of Surveillance Measurements for Unit 1 7-3
7-5 Credibility of Surveillance Measurements for Unit 2 7-4
7-6 Credibility of Surveillance Measurements for Unit 3 7-4

LIST OF FIGURES

1-1 Reactor Vessel Geometry at Core Midplane 1-12

5-1 RCS Heatup Pressure-Temperature Limits through 32 EFPY 5-11
5-2 RCS Cooldown Pressure-Temperature Limits through 32 EFPY 5-12
5-3 RCS Composite Pressure-Temperature Heatup Limits through 32 EFPY 5-13
5-4 RCS Composite Pressure-Temperature Cooldown Limits through 32 EFPY 5-14
5-5 RCS Through-wall Thermal Gradients at Crack Tips for Heatup of 75°F/hr 5-15
5-6 RCS Heatup Thermal Stress Intensity Factors at 3/4T Crack Tip 5-16
5-7 RCS Through-wall Thermal Gradients at Crack Tips for Cooldown of 100°F/hr 5-17
5-8 RCS Cooldown Thermal Stress Intensity Factors at 1/4T Crack Tip 5-18

ABSTRACT

Methodology used to establish the pressure-temperature (P-T) curves and low temperature overpressure protection (LTOP) system limits for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 and 3 is documented in topical report CE NPSD-683-A, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," Reference 1. Report CE NPSD-683-A has been reviewed and approved by the NRC staff for compliance with the specific requirements of Appendices G and H to 10 CFR Part 50.

The basis for the PVNGS Units 1, 2 and 3 reactor coolant system pressure-temperature limit curves, heatup and cooldown rates, low temperature overpressure protection setpoints, vessel adjusted reference temperatures and the projected reactor vessel neutron fluence are developed in this report. These heatup and cooldown limits and LTOP controls, effective through 32 effective full-power years of operation, are designed to prevent potential brittle fracture of the reactor pressure vessel during the most restrictive low temperature overpressure event.

The organization of this report follows that presented in Generic Letter 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," Reference 2. Low temperature overpressure protection setpoints applicable to PVNGS Units 1, 2 and 3 are developed in Section 3.0. RCS heatup and cooldown pressure-temperature limits are developed in Section 5.0.

This report supports the relocation of specific PVNGS Units 1, 2 and 3 reactor coolant system pressure-temperature curves, heatup and cooldown rate limits and low temperature overpressure protection setpoints from the PVNGS Technical Specifications into a separate Pressure-Temperature Limits Report (PTLR) that is controlled by Arizona Public Service (APS).

This page is intentionally blank.

1.0 VESSEL NEUTRON FLUENCE

1.1 OVERVIEW

The neutron fluence for the Palo Verde Nuclear Generation Station (PVNGS) Units 1, 2 and 3 reactor vessel beltline locations has been calculated in accordance with Regulatory Guide 1.190, Reference 3. The following discussions provide the results of the fluence calculation and the details of the calculational analysis for PVNGS.

A three-dimensional discrete ordinates transport analysis was performed for the PVNGS Units 1, 2 and 3 reactors to determine the neutron radiation environment within the reactor pressure vessel and surveillance capsules. In this analysis, fast neutron fluence with energy levels greater than 1.0 MeV and iron atom displacements (dpa) are established on a plant- and fuel cycle-specific basis. These calculations form the basis for projecting the reactor pressure vessel neutron exposure to operating periods through 32 effective full-power years (EFPY).

Current neutron exposure data for the PVNGS reactor vessels, drawn from the sensor sets for each withdrawn capsule and analyzed using dosimetry evaluation methodology, are summarized in Section 1.4. The comparison of these dosimetry evaluation results to the analytical predictions is used to validate the plant-specific neutron transport calculations.

All PVNGS dosimetry evaluations are based on the latest available nuclear cross-section data derived from ENDF/B-VI, Reference 4. These neutron transport and dosimetry evaluation methodologies follow the guidance issued by the staff in Regulatory Guide 1.190.

The peak reactor vessel neutron fluence values at 32 EFPY for PVNGS Units 1, 2 and 3 are given in Table 1-1. These fluence values, calculated at the core midplane inner radius, 1/4T and 3/4T locations, are based on those reported in the post-irradiation analyses of PVNGS surveillance specimen capsules, References 5, 6 and 7. Neutron fluence values at the 1/4T and 3/4T locations are determined using the attenuation formula from Regulatory Guide 1.99, Reference 8.

1.2 DISCRETE ORDINATES ANALYSIS

A plan view of the PVNGS reactor geometry at the core midplane is shown in Figure 1-1. Six irradiation capsules attached to the reactor pressure vessel cladding are included in the reactor vessel surveillance program. These capsules are located at azimuthal angles of 38° and 142° (38° from the core cardinal axes), 230° and 310° (40° from the core cardinal axes), and 43° and 137° (43° from the core cardinal axes). From a neutronic standpoint, the surveillance capsules and associated support structures are significant because the presence of these materials affects both the spatial distribution of neutron flux and the neutron energy spectrum in the water annulus between the core barrel and the reactor vessel. Therefore, the capsules and capsule holders must be included in the analytical model to accurately establish the neutron environment at the test specimen location.

A series of fuel cycle-specific forward transport calculations were performed for the PVNGS reactor vessels and surveillance capsules using the following three-dimensional flux synthesis technique when performing the fast neutron exposure evaluations:

$$\phi(r,\theta,z) = \phi(r,\theta) * \frac{\phi(r,z)}{\phi(r)} \qquad \text{(Eqn. 1-1)}$$

In Equation 1-1:

$\phi(r,\theta,z)$ = synthesized three-dimensional neutron flux distribution,
$\phi(r,\theta)$ = transport solution in r,θ geometry,
$\phi(r,z)$ = two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and
$\phi(r)$ = one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the (r,θ) two-dimensional calculation.

This synthesis procedure is performed for each operating cycle.

Two sets of transport calculations are used when performing the fast neutron exposure evaluations for PVNGS. The first set of calculations is based on the (r,θ) model, shown in Figure 1-1, that includes surveillance capsules at 38°, 40°, and 43°. The second set of calculations is based on the (r,θ) model having no surveillance capsules present. The first set of calculations is used to perform surveillance capsule dosimetry evaluations for comparison with calculated results. The second set of calculations is used to determine the maximum neutron exposure levels at the pressure vessel wall. UFSAR Figures 5.3-1 through 5.3-4 provide additional details regarding the arrangement and location of surveillance specimens at PVNGS.

Nominal design dimensions are employed for the various structural components when developing these analytical models, with two exceptions. The first exception is that the radius to the center of the surveillance capsule holder and the pressure vessel inner radius are taken from the as-built drawings, Reference 9, for each PVNGS reactor. The second exception is that the coolant temperature (i.e., coolant density) is treated on a fuel cycle-specific basis, with water temperatures in the reactor vessel core and bypass regions based on full power operating conditions. The reactor core is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures (such as fuel assembly grids and guide tubes). The geometric mesh description of the (r,θ) reactor models typically consist of 151 radial by 78 azimuthal intervals. Mesh sizes are chosen to ensure that proper convergence of the inner iterations is achieved on a point-wise basis. The point-wise inner iteration flux convergence criterion used in the (r,θ) calculations is set at a value of 0.001.

The (r,z) model used for the PVNGS calculations extends both radially from the centerline of the reactor core out to a location interior to the primary biological shield and axially from an elevation two feet below the active fuel to two feet above the active fuel. As in the case of the (r,θ) models, nominal design dimensions (except for the pressure vessel inner radius as-built dimension) and full power coolant densities are employed in the calculations. In this case, the homogenous core region is treated as an equivalent cylinder with a volume equal to that of the active core zone. The stainless steel core shroud assembly girth rings located between the core shroud and core barrel regions are explicitly included in the model. The (r,z) geometric mesh description of these reactor models typically consists of 141 radial by 79 axial intervals. Consistent with the case of the (r,θ) calculations, mesh sizes are chosen to ensure that proper convergence of the inner iterations is achieved on a point-wise basis. The point-wise inner iteration flux convergence criterion used in the (r,z) calculations is set at a value of 0.001.

One-dimensional radial models used in the synthesis procedure consist of the same 141 radial mesh intervals included in the (r,z) models. Radial synthesis factors can be determined on a mesh-wise basis throughout the entire geometry. Further details of the analysis are provided in References 5, 6 and 7.

Data used in the transport analyses represents cycle-dependent fuel assembly enrichments, burnups, axial power distributions, and radial pin powers. This information is used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport

calculations provide data in terms of fuel cycle averaged neutron flux. This, when multiplied by the appropriate fuel cycle length, generates the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on the initial enrichments and burnup histories of individual fuel assemblies. Composite values of energy release per fission, neutron yield per fission, and fission spectrum are determined from the assembly-dependent fission splits.

All transport calculations supporting this analysis are performed using the DORT discrete ordinates code Version 3.2, Reference 10, and the BUGLE-96 cross-section library, Reference 11. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering is treated with a P_5 Legendre expansion, and angular discretization is modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, are treated on a fuel cycle-specific basis.

1.3 VESSEL FLUENCE ANALYSIS

Fast neutron fluence results from References 5, 6 and 7, projected at a core power level of 3990 MWt for future cycles, are provided in Tables 1-2 through 1-4 for PVNGS Units 1, 2 and 3, respectively. This calculated fast fluence with energy levels greater than 1.0 MeV is given for the reactor vessel inner radius at four azimuthal locations. Data given in Tables 1-2 through 1-4 are determined for the clad/base metal interface; therefore, the data represent the maximum calculated fluence on the respective vessels.

Data tabulations include both plant- and fuel cycle-specific calculated fluence at the end of the most recent operating fuel cycle and future projections through 54 EFPY. These projections are based on a core thermal power of 3990 MWt and the assumption that the core power distributions and associated plant operating characteristics from the most recent operating fuel cycle are representative of future plant operation.

1.4 NEUTRON DOSIMETRY

The validity of the calculated neutron fluence reported in Section 1.3 is demonstrated by a direct comparison against the measured sensor reaction rates and by a least squares evaluation performed for each of the capsule dosimetry sets. For completeness, an assessment of all measured dosimetry removed through 15 EFPY is documented in the post-irradiation surveillance capsule evaluation reports for each PVNGS unit. Comparison of measured dosimetry results to both the calculated and least squares adjusted values is summarized herein.

The sensor sets from the capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190. Sensor set results demonstrate that the overall measurements agree with the calculated and least squares adjusted values to within ±20%, as specified by Regulatory Guide 1.190. This agreement within ±20% validates the calculated fluence reported in Section 1.3.

A least squares evaluation of the PVNGS surveillance capsule dosimetry was performed using the FERRET code, Reference 12. FERRET was employed to combine the results of the plant-specific neutron transport calculations and the sensor set reaction rate measurements to determine best-estimate values of exposure parameters and associated uncertainties for the surveillance capsules withdrawn through 15 EFPY. The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location;
2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set; and

3. The energy-dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

The calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations with sensor reaction rates derived from the measured specific activities. Dosimetry reaction cross-sections and uncertainties were obtained from the SNLRML dosimetry cross-section library, Reference 13. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E 1018-01, Reference 14.

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrums are input to the least squares procedure in the form of variances and covariances. Input uncertainties are assigned following the guidance provided in ASTM Standard E 944-02, Reference 15. The following discussion of the reaction rate uncertainties is associated with the least squares evaluation of the PVNGS Unit 2 surveillance capsule sensor sets, Reference 6. (See References 5 and 7 for the corresponding results for Units 1 and 3.)

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is ensured by using laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type. After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures are assigned the following net uncertainties for input to the least squares evaluation:

Reaction	Unit 2 Uncertainty[(1)]
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{46}Ti(n,p)^{46}Sc$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)^{137}Cs$	10%
(1) Uncertainties are given at the one-sigma (1σ) level.	

The following summary provides a comparison of measured versus calculated (M/C) fast neutron threshold reaction rates for the sensors from Capsule W230 withdrawn from PVNGS Unit 2 at the end of the twelfth fuel cycle.

Reaction	Unit 2 Reaction Rates (rps/atom)		Unit 2 M/C Ratio
	Measured	Calculated	
$^{46}Ti(n,p)^{46}Sc$	5.23E-16	5.28E-16	0.99
$^{54}Fe(n,p)^{54}Mn$	2.88E-15	2.93E-15	0.98
$^{58}Ni(n,p)^{58}Co$ (Cd)	3.59E-15	3.82E-15	0.94
		Average:	0.97
		Standard Deviation:	2.8%

The reaction rate cross-sections used in the least squares evaluations are taken from the SNLRML library. This data library provides reaction cross-sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross-sections and uncertainties are provided in a fine multigroup

structure for use in least squares adjustment applications. These cross-sections are compiled from the most recent cross-section evaluations and have been tested with respect to their accuracy and consistency for least squares evaluations. Furthermore, the library has been empirically tested for use in fission spectra determination, as well as in the fluence and energy characterizations of 14 MeV neutron sources.

For sensors included in the PVNGS Unit 2 surveillance program, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package:

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	4.08 - 4.16%
$^{46}Ti(n,p)^{46}Sc$	4.51 - 4.87%
$^{54}Fe(n,p)^{54}Mn$	3.05 - 3.11%
$^{58}Ni(n,p)^{58}Co$	4.49 - 4.56%
$^{238}U(n,f)^{137}Cs$	0.54 - 0.64%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

The measured-to-calculated reaction rate ratios for the Capsule W230 threshold reactions range from 0.94 to 0.99; the average M/C ratio is 0.97 ±2.8% at the one-sigma level. This direct comparison falls well within the ±20% criterion specified in Regulatory Guide 1.190.

1.5 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the PVNGS surveillance capsule and reactor pressure vessel is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology was carried out in the following four stages:

1. Comparison of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL).
2. Comparisons of calculations to surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment.
3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments.
4. Comparisons of the plant-specific calculations to all available dosimetry results from the PVNGS surveillance program.

The first phase of the methods qualification, PCA comparisons, addressed the adequacy of basic transport calculation, as well as dosimetry evaluation techniques and associated cross-sections. However, this phase did not test the accuracy of commercial core neutron source calculations, nor did it address uncertainties in operational or geometric variables that impact power reactor calculations. The second phase of the qualification, H. B. Robinson comparisons, addressed uncertainties in those additional areas that are primarily methods-related, and would tend to apply generically to all fast neutron exposure evaluations. The third phase of the qualification, analytical sensitivity studies, identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant-specific input parameters. The overall calculational uncertainty applicable to the PVNGS analysis is established from results of these three phases of the methods qualification.

The uncertainties developed from the first three phases of the methodology qualification are summarized as follows:

Basis	Capsule	Vessel IR[1]
PCA Comparisons	3%	3%
H. B. Robinson Comparisons	3%	3%
Analytical Sensitivity Studies	10%	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%	5%
Net Calculational Uncertainty	12%	13%
[1] Uncertainty at IR applies to the vessel inside radius.		

The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results.

The fourth phase of the uncertainty assessment, comparisons to PVNGS measurements, is used to demonstrate the validity of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison is used only as a check, not to modify the calculated surveillance capsule and pressure vessel neutron exposures. Results of the least squares evaluations of the dosimetry from the PVNGS Unit 2 surveillance capsules withdrawn to date are provided in Tables 1-5 and 1-6. In Table 1-5, measured, calculated, and best-estimate values for sensor reaction rates are given for each Unit 2 capsule. Also provided in this tabulation are ratios of the measured reaction rates to both the calculated and least squares adjusted reaction rates. These ratios of measured-to-calculated (M/C) and measured-to-best-estimate (M/BE) illustrate the consistency of the calculated neutron energy spectra fit to the measured reaction rates both before and after adjustment. In Table 1-6, comparison of the calculated and best estimate values of fast neutron flux (E > 1.0 MeV) and iron atom displacement rate are tabulated along with the best-estimate-to-calculated (BE/C) ratios observed for each of the capsules.

The data comparisons provided in Tables 1-5 and 1-6 show that the adjustments to the calculated spectra are relatively small and well within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections. Furthermore, these results indicate that the use of the least squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. For the neutron exposures given in Section 1.3, the uncertainty associated with the unadjusted calculation of fast neutron fluence and iron atom displacements at the surveillance capsule locations is specified as 12% at the one-sigma level. The corresponding uncertainties associated with the least squares adjusted exposure parameters, Table 1-6, are 7% for fast neutron flux and 6% for iron atom displacement rate. Again, these uncertainties from the least squares evaluation are at the one-sigma level.

Further comparisons of the measurement results to calculations are provided in Tables 1-7 and 1-8. These comparisons are provided on two levels. In Table 1-7, calculations of individual threshold sensor reaction rates are compared directly to the corresponding measurements. These threshold reaction rate comparisons provide a good evaluation of the accuracy of the fast neutron portion of the calculated energy spectra. In Table 1-8, calculations of fast neutron exposure rates in terms of neutron flux and the iron atom displacement rate (dpa/s) are compared to the best estimate results obtained from the least squares evaluation of the capsule dosimetry results. These two levels of comparison yield consistent and similar results. All measurement-to-calculation comparisons fall well within the 20% limits specified as the acceptance criteria in Regulatory Guide 1.190.

In the case of the direct comparison of measured and calculated sensor reaction rates, the M/C comparisons for fast neutron reactions range from 0.94 to 1.16 for the eight samples included in the data set. The overall average M/C ratio for the entire set of PVNGS Unit 2 data is 1.03, with an associated standard deviation of 7.6%.

In the comparisons of best estimate and calculated fast neutron exposure parameters for Unit 2, the corresponding BE/C comparisons for the capsule data sets range from 0.96 to 1.01 for fast neutron flux, and from 0.96 to 1.00 for iron atom displacement rate. The overall average BE/C ratios for fast neutron flux and iron atom displacement rate are 0.98 with a standard deviation of 3.0%, and 0.99 with a standard deviation of 3.4%, respectively. Results from the other two units are comparable. For Unit 1, the corresponding BE/C comparisons for the capsule data sets range from 0.94 to 1.03 for fast neutron flux, and from 0.94 to 1.03 for iron atom displacement rate. The overall average BE/C ratios for fast neutron flux and iron atom displacement rate are 0.98 with a standard deviation of 4.7%, and 0.98 with a standard deviation of 4.5%, respectively. For Unit 3, the corresponding BE/C comparisons for the capsule data sets range from 0.95 to 1.01 for fast neutron flux, and from 0.96 to 1.01 for iron atom displacement rate. The overall average BE/C ratios for fast neutron flux and iron atom displacement rate are 0.98 with a standard deviation of 4.7%, and 0.98 with a standard deviation of 4.4%, respectively.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 1.3 of this report are validated for use in the assessment of the condition of the materials comprising the beltline region of the PVNGS Unit 1, 2 and 3 reactor pressure vessels.

Table 1-1
Calculated Neutron Fluence at Core Midplane through 32 EFPY

Unit	PVNGS Unit 1[3]	PVNGS Unit 2[3]	PVNGS Unit 3[3]
Vessel Inner Radius[1]	1.51E+19 n/cm^2	1.66E+19 n/cm^2	1.74E+19 n/cm^2
Vessel 1/4 Thickness[2]	8.77E+18 n/cm^2	9.63E+18 n/cm^2	1.01E+19 n/cm^2
Vessel 3/4 Thickness[2]	2.96E+18 n/cm^2	3.25E+18 n/cm^2	3.40E+18 n/cm^2

(1) Clad-to-base metal interface.
(2) Calculated as $f_x = f_{surface}\ e^{(-0.24x)}$ where x = 1/4 or 3/4 thickness dimension for vessel lower shell.
(3) Projected at a core power of 3990 MWt.

Table 1-2
Calculated Azimuthal Variation of Neutron Fluence for Unit 1

Cycle	Cumulative Irradiation Time (EFPY)	Neutron Fluence[2], E > 1.0 MeV (n/cm^2)			
		0°	15°	30°	45°
1	1.22	5.74E+17	8.18E+17	8.32E+17	9.62E+17
2	2.00	9.14E+17	1.19E+18	1.24E+18	1.40E+18
3	3.37	1.57E+18	1.93E+18	1.89E+18	2.08E+18
4	4.57	2.03E+18	2.57E+18	2.49E+18	2.72E+18
5	5.79	2.31E+18	2.92E+18	2.90E+18	3.28E+18
6	6.99	2.57E+18	3.26E+18	3.31E+18	3.84E+18
7	8.32	2.83E+18	3.61E+18	3.69E+18	4.28E+18
8	9.76	3.14E+18	3.97E+18	4.05E+18	4.71E+18
9	11.13	3.48E+18	4.38E+18	4.54E+18	5.37E+18
10	12.49	3.81E+18	4.80E+18	4.97E+18	5.85E+18
11	13.83	4.16E+18	5.23E+18	5.44E+18	6.47E+18
Future[1]	16.00	4.75E+18	5.94E+18	6.22E+18	7.50E+18
Future[1]	32.00	9.06E+18	1.12E+19	1.20E+19	1.51E+19
Future[1]	48.00	1.34E+19	1.64E+19	1.78E+19	2.28E+19
Future[1]	54.00	1.50E+19	1.84E+19	1.99E+19	2.56E+19

(1) Future projections for Unit 1 are based on power level of 3990 MWt.
(2) Calculated at reactor vessel clad-to-metal interface.

Table 1-3
Calculated Azimuthal Variation of Neutron Fluence for Unit 2

Cycle	Cumulative Irradiation Time (EFPY)	Neutron Fluence[2], E > 1.0 MeV (n/cm^2)			
		0°	15°	30°	45°
1	1.15	5.49E+17	7.78E+17	7.96E+17	9.20E+17
2	2.23	9.10E+17	1.28E+18	1.34E+18	1.51E+18
3	3.41	1.34E+18	1.86E+18	1.92E+18	2.18E+18
4	4.54	1.77E+18	2.41E+18	2.50E+18	2.79E+18
5	5.52	1.98E+18	2.73E+18	2.97E+18	3.32E+18
6	6.39	2.21E+18	3.03E+18	3.29E+18	3.70E+18
7	7.70	2.48E+18	3.43E+18	3.84E+18	4.32E+18
8	9.12	2.79E+18	3.84E+18	4.23E+18	4.72E+18
9	10.50	3.10E+18	4.25E+18	4.67E+18	5.32E+18
10	11.73	3.38E+18	4.64E+18	5.14E+18	5.92E+18
11	13.14	3.92E+18	5.20E+18	5.64E+18	6.50E+18
12	14.35	4.30E+18	5.66E+18	6.12E+18	7.14E+18
Future[1]	16.00	4.83E+18	6.30E+18	6.79E+18	8.03E+18
Future[1]	32.00	9.96E+18	1.24E+19	1.33E+19	1.66E+19
Future[1]	48.00	1.51E+19	1.85E+19	1.97E+19	2.51E+19
Future[1]	54.00	1.70E+19	2.08E+19	2.21E+19	2.83E+19

(1) Future projections for Unit 2 are based on power level of 3990 MWt.
(2) Calculated at reactor vessel clad-to-metal interface.

Table 1-4
Calculated Azimuthal Variation of Neutron Fluence for Unit 3

Cycle	Cumulative Irradiation Time (EFPY)	Neutron Fluence[2], E > 1.0 MeV (n/cm^2)			
		0°	15°	30°	45°
1	1.07	5.31E+17	7.52E+17	7.71E+17	8.94E+17
2	2.14	8.89E+17	1.26E+18	1.30E+18	1.47E+18
3	3.31	1.32E+18	1.76E+18	1.81E+18	2.09E+18
4	4.44	1.62E+18	2.17E+18	2.28E+18	2.68E+18
5	5.67	1.97E+18	2.59E+18	2.77E+18	3.26E+18
6	6.89	2.22E+18	2.94E+18	3.20E+18	3.78E+18
7	8.33	2.53E+18	3.38E+18	3.61E+18	4.22E+18
8	9.75	2.86E+18	3.84E+18	4.12E+18	4.86E+18
9	11.07	3.14E+18	4.23E+18	4.63E+18	5.49E+18
10	12.44	3.46E+18	4.68E+18	5.24E+18	6.20E+18
11	13.75	3.77E+18	5.15E+18	5.84E+18	6.95E+18
Future[1]	16.00	4.32E+18	5.96E+18	6.87E+18	8.23E+18
Future[1]	32.00	8.25E+18	1.17E+19	1.42E+19	1.74E+19
Future[1]	48.00	1.22E+19	1.75E+19	2.15E+19	2.65E+19
Future[1]	54.00	1.37E+19	1.97E+19	2.43E+19	2.99E+19

[1] Future projections for Unit 3 are based on power level of 3990 MWt.
[2] Calculated at reactor vessel clad-to-metal interface.

Table 1-5
Comparison of Measured, Calculated and Best Estimate Reaction Rates for Unit 2

	Capsule W137 Reaction Rate[1]			Capsule W137 Ratios	
Reaction	Measured	Calculated	Best Estimate	M/C	M/BE
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	4.84E-17	4.18E-17	4.72E-17	1.16	1.03
$^{46}Ti(n,p)^{46}Sc$	7.24E-16	6.44E-16	7.10E-16	1.12	1.02
$^{54}Fe(n,p)^{54}Mn$	3.75E-15	3.60E-15	3.80E-15	1.04	0.99
$^{58}Ni(n,p)^{58}Co$ (Cd)	4.85E-15	4.69E-15	4.92E-15	1.03	0.99
$^{238}U(n,f)^{137}Cs$ (Cd)	1.16E-14	1.21E-14	1.24E-14	0.96	0.94
	Capsule W230 Reaction Rate[1]			**Capsule W230 Ratios**	
Reaction	**Measured**	**Calculated**	**Best Estimate**	**M/C**	**M/BE**
$^{46}Ti(n,p)^{46}Sc$	5.23E-16	5.28E-16	5.17E-16	0.99	1.01
$^{54}Fe(n,p)^{54}Mn$	2.88E-15	2.93E-15	2.85E-15	0.98	1.01
$^{58}Ni(n,p)^{58}Co$ (Cd)	3.59E-15	3.82E-15	3.67E-15	0.94	0.98

[1] Reaction rate in reactions per second per atom.

Table 1-6
Comparison of Calculated and Best Estimate Neutron Flux for Unit 2

Capsule ID[2]	Neutron Flux, E > 1.0 MeV (n/cm^2-s)			
	Calculated[1]	Best Estimate	Uncertainty (1σ)	BE/C
W137	2.70E+10	2.72E+10	6%	1.007
W230	2.19E+10	2.11E+10	7%	0.965
Capsule ID[2]	**Iron Atom Displacement Rate (dpa/s)**			
	Calculated[1]	**Best Estimate**	**Uncertainty (1σ)**	**BE/C**
W137	3.92E-11	3.93E-11	6%	1.002
W230	3.19E-11	3.04E-11	6%	0.955

(1) Calculated results are based on the synthesized transport calculations taken at the core midplane following the completion of each respective capsules irradiation period.
(2) Capsule identification is based on azimuthal position.

Table 1-7
Comparison of Measured to Calculated Reaction Rate Ratios for Unit 2

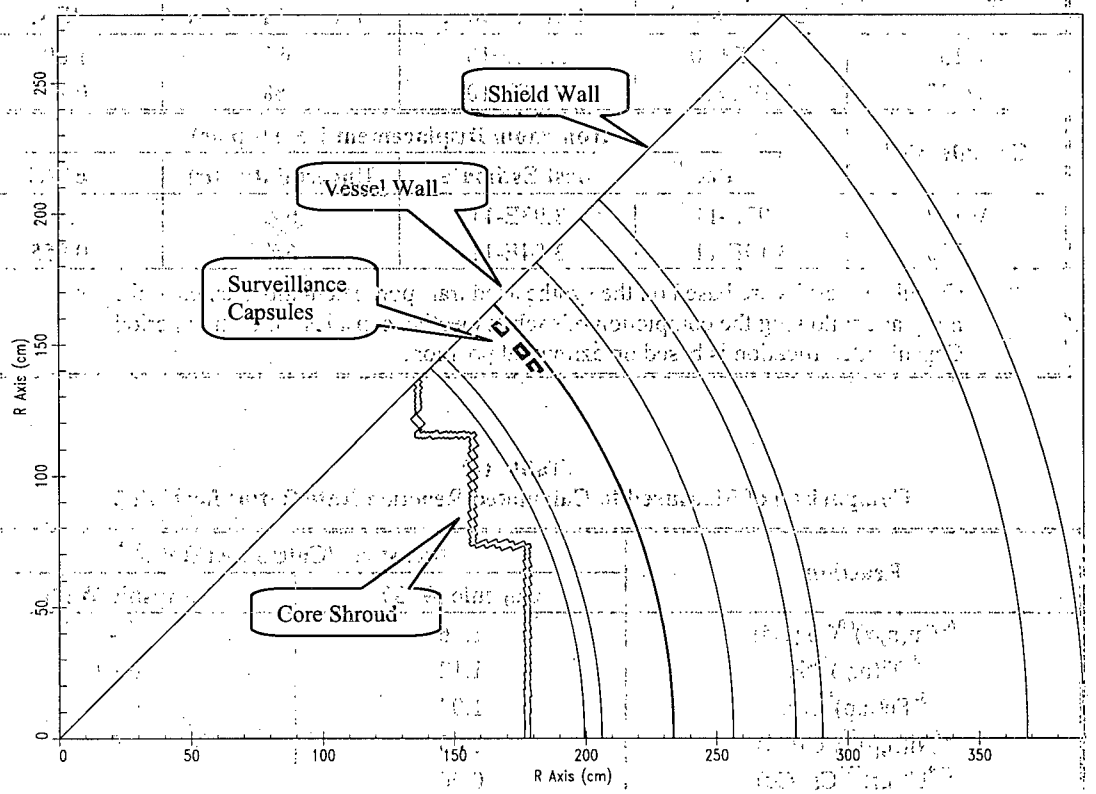
Reaction	Measured/Calculated Ratio[1]	
	Capsule W137	Capsule W230
$^{63}Cu(n,\alpha)^{60}Co$ (Cd)	1.16	--
$^{46}Ti(n,p)^{46}Sc$	1.12	0.99
$^{54}Fe(n,p)^{54}Mn$	1.04	0.98
$^{58}Ni(n,p)^{58}Co$ (Cd)	1.03	0.94
$^{238}U(n,f)^{137}Cs$ (Cd)	0.96	--
Average	1.06	0.97
Standard Deviation	7.4%	2.8%

(1) The overall average M/C ratio for the set of eight sensor measurements is 1.03 with an associated standard deviation of 7.6%.

Table 1-8
Comparison of Best Estimate to Calculated Neutron Flux Ratios for Unit 2

Capsule ID	Best Estimate/Calculated Ratio	
	Neutron Flux, E > 1.0 MeV	dpa/s
W137	1.01	1.00
W230	0.96	0.96
Average	0.99	0.98
Standard Deviation	3.0%	3.4%

Figure 1-1
Reactor Vessel Geometry at Core Midplane



2.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The surveillance program for PVNGS Units 1, 2 and 3 is based on ASTM E185-79, Reference 16, which presents criteria for monitoring changes in the fracture toughness properties of reactor vessel beltline materials. This program and the surveillance capsule withdrawal schedule for PVNGS are described in this section. Reports describing the pre-irradiation evaluations of the surveillance materials are contained in References 17, 18 and 19. References 5, 6 and 7 are the most recent reports that describe the post-irradiation evaluations of the PVNGS surveillance materials.

The PVNGS surveillance program adheres to all ASTM E185-79 guidelines and to 10 CFR 50, Appendix H, Reference 20. All reactor vessel surveillance specimen capsule holders are attached to the inside vessel wall cladding in the beltline region at PVNGS. This capsule holder attachment method meets the design and inspection requirements of the ASME Code, Sections III and XI.

2.1 TEST MATERIAL SELECTION

Three metallurgically different materials representative of the reactor vessel are used for test specimens in accordance with the general guidelines of ASTM E185-79. These materials include base metal, weld metal, and heat affected zone (HAZ) materials.

2.1.1 Plate Material Selection

The lower shell and a portion of the intermediate shell plate materials are nearest to the reactor core; hence they will sustain the greatest neutron exposure. Each of the six plates which make up the intermediate and lower shell courses were evaluated (References 21, 22, and 23) in terms of initial RT_{NDT}, residual copper and phosphorus content, and the cumulative effect of irradiation on RT_{NDT} shift. The base materials selected for the PVNGS surveillance programs, including two plates selected for Unit 1, are listed in Table 2-1.

Base metal test materials for PVNGS Units 1, 2 and 3 are manufactured from sections of the shell plates listed in Table 2-1. The section of the 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.

2.1.2 Weld Material Selection

The weld materials for the PVNGS Units 1, 2 and 3 surveillance programs are selected to duplicate the materials in the lower shell axial weld seams. Table 2-2 lists the weld materials selected for the PVNGS surveillance programs.

Weld metal and HAZ specimens for PVNGS are produced by welding together sections from the beltline plates (i.e., the lower shell plates listed in Table 2-1). The sections of the 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. These specimens are heat-treated to a condition representative of the final metallurgical condition of the weld metal in the completed reactor vessel.

A summary of the materials and types of specimens included in the six PVNGS surveillance capsules is presented in Table 2-3. Table 2-3 also defines specimen quantities for each surveillance capsule assembly by type (i.e., pre-cracked Charpy or compact tension), as described below.

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 specimens to the results of similar tests using unirradiated specimens. For example, changes in toughness of the vessel materials are determined using the amount of the temperature shift in the Charpy impact test curves between the unirradiated material and the irradiated material, measured at 30 ft-lbs impact energy. Drop weight, Charpy impact, pre-cracked Charpy, compact tension, and tensile test specimens are 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. Use of drop weight and Charpy impact tests to establish initial reference temperature was done in accordance with NB-2300 of the ASME Code, Section III. Charpy impact, pre-cracked Charpy, compact tension, 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 References 21, 22 and 23. An amount of base metal, weld metal, and HAZ test material sufficient to provide two additional sets of test specimens has been retained with full documentation and identification for future use. Each of the test materials was chemically analyzed for approximately 21 elements, including all those listed in Paragraph 5.7 of ASTM E185-79.

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 References 21, 22, and 23. 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 (transverse orientation), 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 the reference temperature, RT_{NDT}, from which subsequent neutron-induced changes for the PVNGS plate and weld materials are determined.

Charpy Impact Test Specimens: Eighteen to twenty-four test specimens each of base metal (longitudinal and transverse orientation), weld metal, and HAZ material are provided for impact testing. This quantity exceeds the minimum number of test specimens recommended by ASTM E185-79 for developing a Charpy impact energy transition temperature curve for these materials. These data together with the drop weight NDTT are used to establish the reference temperature, RT_{NDT}, for each material.

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

Pre-cracked Charpy Impact Test Specimens: Twelve pre-cracked Charpy test specimens each of base metal (longitudinal and transverse orientation) and weld metal materials are provided for fracture toughness testing. These test specimens are provided for supplemental toughness property determination, and are in addition to the drop weight, standard Charpy, and uniaxial tension specimens required by ASTM E185-79.

Compact Tension Test Specimens: Four 1/2T and eight 1T compact tension test specimens each of base metal (transverse) and weld metal material are provided for fracture toughness testing. Similar to the pre-cracked Charpy specimens, these test specimens are provided for supplemental toughness property determination, and are in addition to those required by ASTM E185-79.

2.2.3 Irradiated Specimens

Charpy impact, pre-cracked Charpy, compact tension, and uniaxial tension test specimens are used to determine changes in the static and dynamic properties of the PVNGS reactor vessel materials due to irradiation. The type and quantity of test specimens provided for establishing the properties of the irradiated materials over the lifetime of the vessel, References 21, 22, and 23, are presented in Table 2-3.

2.3 SURVEILLANCE SPECIMEN IRRADIATION

2.3.1 Specimen Encapsulation

The test specimens are housed within corrosion-resistant capsule assemblies in an inert environment 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 neutron fluence has been attained.

A typical PVNGS capsule assembly (References 21, 22, and 23) consists of a series of three 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 that are attached to the reactor vessel cladding. The lock assemblies fix the locations of the capsules within the holders and prevent relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor.

Each surveillance specimen compartment consists of two sections connected by a spacer. Capsule compartments are assigned a unique identification so that a complete record of test specimen location within each compartment can be maintained. Each PVNGS reactor vessel contains six surveillance capsules, including both pre-cracked Charpy assemblies and compact tension assemblies. PVNGS Unit 1 has two pre-cracked Charpy assemblies and one compact tension assembly for lower shell plate M-4311-1 specimens, and a second set for intermediate shell plate M-6701-2 specimens. The two sets of assemblies are necessary to accommodate the two different Unit 1 base metal surveillance materials. PVNGS Units 2 and 3 have three each of the pre-cracked Charpy assemblies and compact tension assemblies. These assemblies are further described below.

2.3.1.1 Pre-cracked Charpy Assemblies

The pre-cracked Charpy capsule assemblies consist of three separate compartments. Two compartments, designated as Charpy and flux compartments, contain Charpy impact (standard and pre-cracked) specimens and neutron flux spectrum monitors. Charpy specimens are provided to establish an impact energy transition curve for a given irradiated material. These specimens are arranged vertically in one-by-three 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.

One compartment (designated the temperature, flux, tension, and Charpy compartment) contains the uniaxial tension test specimens, neutron flux spectrum monitors, temperature monitors, and standard reference material (SRM) Charpy impact specimens. Tensile specimens are placed in a housing machined to fit inside the compartment. Split spacers are placed around the gage length of the tensile specimens to minimize the temperature differential between the specimen gage length and the coolant. The entire compartment is sealed within an atmosphere of helium.

2.3.1.2 Compact Tension Assemblies

The compact tension capsule assemblies consist of three separate compartments. Two compartments (designated as Charpy, flux, and compact tension compartments) contain Charpy impact (standard and pre-cracked) specimens and neutron flux spectrum monitors. Standard Charpy specimens are provided to establish an impact energy transition curve for a given irradiated material. The Charpy specimens are arranged vertically in one-by-three arrays, and are oriented with the notch toward the core. The 1/2T compact tension specimens are oriented with the opening of the crack starter notch facing the top of the compartment (i.e., to obtain a uniform fluence gradient across the crack front). The temperature differential between the specimens and the reactor coolant is minimized by using spacers between the specimens and the compartment, as well as by sealing the entire assembly in an atmosphere of helium.

One compartment is designated as a temperature, flux, tension, and Charpy compartment. This compartment is the same as described for the pre-cracked Charpy assemblies.

2.3.2 Flux and Temperature Measurement

The changes in the reactor vessel materials toughness are derived from specimens irradiated to various fluence levels and in different neutron energy spectra. Complete information on the neutron flux, neutron energy spectra, and the irradiation temperature of the surveillance specimens is obtained from the flux and temperature monitors in order to permit accurate interpretation of the surveillance material test results.

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. (References 21, 22 and 23 provide a list of neutron detectors installed at PVNGS.)

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 that identifies the material and facilitates handling. Cadmium covers are used for those materials (e.g., uranium, nickel, copper, and cobalt) that 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 PVNGS surveillance program also includes standard reference material (SRM) specimens. These are Charpy impact test specimens made from a reference heat of ASTM A533

B1. Those specimens are being irradiated along with the specimens made from the PVNGS reactor vessel materials to serve as a correlation monitor material. The changes in impact properties of the reference material provide a cross-check on the neutron irradiation environment 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

Changes in mechanical and impact properties of irradiated specimens are highly dependent on the irradiation temperature. Therefore, it is necessary to have knowledge of the temperatures of both the specimens and 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 small pieces of low melting point alloys or pure metals in the capsule assemblies. The compositions of monitor materials and melting points in the operating range of power reactors are listed in References 21, 22 and 23. The monitors are selected to bracket the operating temperature of the reactor vessel.

The temperature monitors consist of a helix of low melting point alloy wire inside a sealed quartz tube. A stainless steel weight supported by the wire ensures that monitor wires having reached their melting point are clearly indicated. The compositions and, therefore, the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes that contain the alloy wires.

A set of temperature monitors is included in each temperature, flux, tension, and Charpy compartment. The temperature monitors are placed in holes drilled in stainless steel housings to provide the maximum temperature to which the specimens are exposed.

2.3.3 Irradiation Specimen Location

The test specimens are enclosed within capsule assemblies. They are irradiated at six radial positions around the active core and at axial positions that are bisected by the midplane of the core. Test specimens contained in the capsule assemblies are used to monitor the irradiation-induced property changes of the reactor vessel materials. Therefore, these capsules 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 within approximately 15% of that seen by the adjacent vessel wall. Therefore, the RT_{NDT} changes resulting from the irradiation of these specimens will closely approximate the RT_{NDT} changes in the materials of the reactor vessel.

2.3.4 Surveillance Capsule Withdrawal Schedule

Surveillance capsule assemblies are withdrawn during an appropriate refueling outage when the test specimens have attained the desired fluence. Table 2-4 presents the azimuthal locations for the capsule assemblies and the actual time of capsule removal in terms of effective full power years.

The capsule assemblies located in the 137-degree and 230-degree position have been withdrawn from each of the PVNGS units. Additionally, the 38-degree capsule has been withdrawn from Unit 1. The actual time of capsule removal may be modified to coincide with the refueling outage or plant shutdown most closely approaching the scheduled time for withdrawal.

Table 2-1
Base Metal Materials Selected for Surveillance Program

PVNGS Unit	Plate ID Number	Plate Location
Unit 1	M-4311-1	Lower Shell
Unit 1	M-6701-2	Intermediate Shell
Unit 2	F-773-1	Lower Shell
Unit 3	F-6411-2	Lower Shell

Table 2-2
Weld Metal Materials Selected for Surveillance Program

PVNGS	Wire Heat Number	Flux Type	Flux Lot Number
Unit 1	90071	Linde 0091	1054
Unit 2	3P7317	Linde 124	0662
Unit 3	4P7869	Linde 124	0281

Table 2-3
Type and Quantity of Specimens for Irradiation Exposure

	Pre-cracked Cv Assembly			Compact Tension Assembly		
Material	**Std. Cv**	**Tension**	**PCv**	**Std. Cv**	**Tension**	**1/2T CT**
Base Metal (Long.)	9	-	9	-	-	-
Base Metal (Trans.)	15	3	9	15	3	10
Weld Metal	15	3	9	15	3	10
HAZ	12	-	-	12	-	-
SRM[1]	9	-	-	9	-	-
Total in Assembly	60	6	27	51	6	20

(1) Standard Reference Material (SRM) characterized by heavy section steel technology program. Specimens are provided only for correlation with characterization tests.

Table 2-4
Surveillance Capsule Assembly Removal Schedule through 32 EFPY

Azimuthal Location (degrees)	Unit 1		Unit 2		Unit 3	
	Capsule Number	Removal Time (EFPY)[1]	Capsule Number	Removal Time (EFPY)[1]	Capsule Number	Removal Time (EFPY)[1]
38	1	9.76	1	Standby	1	Standby
43	2	Standby	2	Standby	2	Standby
137	3	4.57	3	4.54	4	4.44
142	4	Standby	4	Standby	3	Standby
230	5	13.83	5	14.35	5	13.75
310	6	18 - 24	6	18 - 24	6	18 - 24

(1) Removal time may be adjusted to coincide with the refueling outage or scheduled shutdown most closely approximating the withdrawal schedule.

3.0 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM LIMITS

3.1 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

3.1.1 Introduction

The low temperature overpressure protection (LTOP) system protects the PVNGS reactor coolant system (RCS) pressure boundary integrity by ensuring that the pressure remains below the applicable P-T limits of 10 CFR 50, Appendix G, at low temperature conditions when the RCS is water-solid.

Technical Specifications require two shutdown cooling system (SCS) relief valves ("LTOP valves") to be operable or the RCS vented with a vent area greater than or equal to 16 square inches when any RCS cold leg temperature is less than the specified LTOP enable temperature. Once enabled, the LTOP system automatically maintains the RCS pressure below the brittle fracture threshold without requiring action or intervention by the plant operator.

This Section describes the process for developing the LTOP system limits and demonstrating adequate LTOP performance at PVNGS Units 1, 2 and 3.

3.1.2 LTOP Technical Specifications

Protection of the PVNGS reactor coolant pressure boundary against brittle fracture is provided during Mode 4 heatup or cooldown, when all cold leg temperatures are above the LTOP enable temperature, by at least one operable pressurizer safety valve with a lift pressure setting as specified in Technical Specification LCO 3.4.11.

Similarly, RCS brittle fracture protection is provided when any RCS cold leg temperature is less than or equal to the specified LTOP enable temperature by two operable SCS relief valves or, if the RCS is depressurized, by providing an RCS vent area equal to or greater than sixteen square inches. This limit on plant operations applies in Modes 4, 5 and 6 as specified in TS LCO 3.4.13. TS LCO 3.4.13 also prohibits startup of a reactor coolant pump in Mode 4 if the secondary side water temperature in either steam generator is more than 100°F above any RCS cold leg temperature.

3.2 BASIS FOR LTOP SYSTEM LIMITS

PVNGS LTOP system setpoints and controls are developed using the methodology approved by the staff in CE NPSD-683-A, Reference 1, and the plant-specific methodology presented in this report.

Technical Specifications require two SCS suction line ("LTOP") relief valves to be operable, although event mitigation only credits one valve, along with controls on the heatup and cooldown rate whenever any RCS cold leg temperature is below an enable setpoint. The SCS relief valve setpoint and capacity, and the LTOP heatup and cooldown enable setpoint temperatures support analyses that ensure the peak RCS pressure resulting from postulated overpressure events does not exceed the allowable RCS P-T limits.

Calculated limiting temperatures for LTOP heatup and cooldown rate protection are given in Table 3-1. Between the minimum boltup temperature determined in accordance with Section 5.6 and the LTOP enable temperature determined in accordance with Section 5.12, the peak RCS pressures resulting from postulated

overpressure events does not exceed either the minimum pressure value determined in accordance with Section 5.7 or the corrected allowable pressures determined in Section 5.14.

3.2.1 Peak Transient Pressures

Analyses of the limiting case mass addition (HPSI start) and energy addition (RCP start) events are performed using conservative LTOP assumptions to establish the peak transient RCS pressures at PVNGS. Calculated results confirm that the shutdown cooling system relief valves (SI-179, SI-189, also termed "LTOP valves") installed at PVNGS Units 1, 2 and 3 have excess capacity relative to relieving requirements. Therefore, the limiting mass and energy addition overpressure events are quickly terminated upon valve actuation.

These design basis peak pressure analyses incorporate the following LTOP assumptions:

- The pressurizer is initially water-solid with no steam space and no credit for the presence of a cover gas (e.g., nitrogen),
- The RCS pressure boundary is rigid, i.e., no expansion due to pressure or thermal effects,
- No heat is transferred to or from the RCS,
- The RCS letdown flow is isolated,
- All pumps attain rated speed instantaneously,
- Only one SCS relief valve actuates to mitigate the transient,
- No operator action is required, and
- Conservative energy addition sources are used to evaluate both energy addition and mass addition transients, including:
  - Full heat output from all pressurizer heaters (1800 kW) is assumed for the duration of the transient in order to maximize the energy input into the RCS, and
  - Decay heat, with two-sigma uncertainties, 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 ensure a conservative analysis:

- The SCS 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 SCS relief valve opening profile is consistent with the ASME model described in Section 3.2.1.1; this model results in a delayed response to the relief valve lift and a resulting delay in providing the relief capacity.
- No RCP seal leakage or controlled bleed-off is assumed,
- The RCS is isothermal and is not cooled or heated by any mass addition, and
- The initial conditions are chosen to maximize the pressure transients in order to develop the greatest rate of pressure rise.

The transient analysis methodology used to establish results for the design basis low temperature overpressure events at PVNGS Units 1, 2 and 3 is discussed in the following sections.

3.2.1.1 Relief Valve Overpressure Protection

Low temperature overpressure protection at PVNGS Units 1, 2 and 3 is provided by spring actuated relief valves ("LTOP" valves) installed in the SCS suction lines. These relief valves are placed in service during RCS cooldown at or above the LTOP enable temperature, and at or below the enable temperature during heatup, to ensure the reactor vessel is protected from brittle fracture in the event of a low temperature overpressure event.

The SCS relief valve opening and discharge characteristics are consistent and conservative relative to the ASME Code requirements for spring loaded safety valves. These valves follow the ASME Code model with an initial SCS relief valve opening of 30% at 3% accumulation and full opening at 10% accumulation at which point each valve has a relieving capacity of 5635 gpm. This relieving capacity is substantially greater than that needed to mitigate the design basis RCS mass addition and energy addition transients.

The setpoint for SCS relief valve actuation, a lift pressure of 467 psig as defined in TS LCO 3.4.13, is established by the limiting pressure to which any SCS component (e.g., LPSI pump seals) may be exposed. This setpoint is sufficiently below the limiting reactor coolant system pressure established by 10 CFR 50 Appendix G. Therefore, low temperature overpressure protection of the reactor vessel is assured since the calculated RCS pressure rise is terminated before the relief valve reaches full open in both the limiting energy addition and mass addition LTOP transients.

3.2.1.2 Mass Addition Overpressure Event

The limiting design basis mass addition transient is an inadvertent actuation of two HPSI pumps while all three charging pumps are operating at their design flowrate. Mass addition from safety injection tanks is not considered in this transient since Technical Specification 3.5.2 allows the tanks to be isolated during Mode 4 operation with the pressurizer pressure below 430 psia.

This event is analyzed using the methodology of CE NPSD-683-A. That method determines inputs for HPSI mass addition, charging pump mass addition, and the equivalent mass addition that results from energy addition. The magnitude of the pressurization is determined by superposition of the mass addition curve onto the relief valve discharge curve, both of which describe mass flow rate as a function of pressurizer pressure. The equilibrium pressure is taken as the intersection of the two pressure curves, assuming liquid input and discharge, at which the mass addition rate matches the relief valve discharge flow rate.

The HPSI mass addition is obtained from the maximum volumetric delivery curves developed for emergency core cooling system calculations using the 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 greatest rate of mass addition.

A rapid RCS pressure rise occurs upon initiation of a design basis mass addition transient at PVNGS. The transient is quickly mitigated, with a peak RCS pressure at the pressurizer of less than 499 psia, due to the large capacity discharge through the SCS suction line relief valves.

3.2.1.3 Energy Addition Overpressure Event

The limiting energy addition LTOP transient at PVNGS is modeled by simulating the pressure increase in a water-solid, idle RCS due to reverse heat transfer from a hot steam generator when a single reactor coolant pump is started. This model, described in Reference 24 and applied to LTOP analyses for PVNGS, simulates the discharge from a relief valve and determines the RCS pressure during the relieving action.

The following conservative assumptions are included in the analysis of the design basis energy addition transient:

- The steam generator secondary temperature is assumed to be 100°F hotter than the primary coolant temperature throughout the transient,
- 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 RCS boundary is assumed rigid, i.e., there is no increase in RCS volume with an increase in RCS pressure or temperature. Also, no RCP seal leakage is assumed and the letdown flow paths are isolated, with mass release only through relief valves,
- The RCS pressure at the initiation of the transient is selected to be 435 psia, and
- The method for the determining the decay heat contribution is consistent with that described in CE NPSD-683-A.

With the relatively large capacity SCS relief valves installed at PVNGS, the energy addition pressure transient is quickly mitigated upon the valve opening at 3% accumulation. Analysis confirms that the peak pressure at the pressurizer during the energy addition transient remains below 499 psia and complies with NRC requirements in a conservative manner.

3.2.2 Applicable Pressure-Temperature Limits

Applicable P-T heatup and cooldown limits used to support LTOP controls are established based on the methodology described in CE NPSD-683-A. Using this methodology, the limiting rates are selected based on a comparative evaluation of the family of heatup and cooldown rate-based pressure-temperature limits (described in Section 5.0) to the peak transient pressures described in Section 3.2.1.

The applicable P-T heatup and cooldown rate limits, developed in Sections 5.6 and 5.12 respectively, are controlling over the range of RCS temperatures from the minimum bolt up temperature (80°F) to the LTOP enable temperature (221°F). RCS heatup and cooldown rates less than or equal to those shown in Table 3-1 apply to Units 1, 2 and 3 through 32 EFPY.

Table 3-1
RCS Heatup and Cooldown Rate Limits through 32 EFPY

Indicated RCS Cold Leg Temperature (°F)	Heatup Rate (°F/hr)	Cooldown Rate (°F/hr)
80° to ≤ 92°	≤ 75	≤ 30
> 92° to ≤ 100°	≤ 75	≤ 50
> 100° to ≤221°	≤ 75	≤ 100
> 221°	≤ 75	≤ 100

4.0 ADJUSTED REFERENCE TEMPERATURE

Adjusted reference temperatures (ART) for the reactor vessel beltline region are determined using NRC-accepted methodologies, as described below. Limiting ART values for the PVNGS Units 1, 2, and 3 reactor vessel beltline regions corresponding to 32 effective full power years (EFPY) for the 1/4T and 3/4T locations are:

Limiting Material	Location	ART
Unit 1 - Intermediate Shell Plate M-6701-2	1/4T	116°F
Unit 1 - Intermediate Shell Plate M-6701-2	3/4T	103°F

RT_{PTS} values for the PVNGS Units 1, 2, and 3 reactor vessels are calculated in accordance with 10 CFR Part 50.61 using the neutron fluence at the clad-to-base metal interface through 32 EFPY. The highest predicted RT_{PTS} value is 123°F, and corresponds to the intermediate shell plate M-6701-2 of Unit 1. Section 4.4 describes the determination of RT_{PTS}.

4.1 BACKGROUND

Determination of the ART values for the PVNGS Units 1, 2, and 3 reactor vessel beltline materials through 32 EFPY is discussed in this Section. ART values are conservative with respect to the measurements obtained from the irradiated surveillance capsules, as described in Section 7.

4.2 RESULTS

The predicted adjusted reference temperature and RT_{PTS} results through 32 EFPY for the limiting material in each PVNGS unit are summarized in Table 4-1. All ART values are conservatively predicted relative to the measured properties, i.e., to the Charpy transition temperature shifts measured as part of the surveillance capsule evaluations as discussed in Section 7.

4.2.1 Fluence Calculation

The peak design fast fluence at the vessel clad-to-base metal interface through 32 EFPY is 3.29E+19 n/cm^2. For conservatism, it was assumed that this peak fluence is applied to each of the PVNGS reactor vessel beltline plates and welds; i.e., no reduction factor is applied to account for axial or azimuthal variations from the peak value. From Table 1-1, it should be noted that the peak calculated neutron fluence through 32 EFPY based on evaluation of irradiated surveillance capsules is substantially less than the design value of 3.29E+19 n/cm^2 and reflects plant-specific fuel management.

The following equation from Regulatory Guide 1.99 is used to determine the attenuation of neutron fluence with distance into the plate:

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

In Equation 4-1:

f = neutron fluence at the desired location,
f_{surf} = neutron fluence at the inside vessel surface,
x = distance from the inside vessel surface, inches.

For PVNGS, the given neutron fluence is assumed to apply at the clad-to-base metal interface. The distance 'x' is measured from the clad-to-base metal surface into the plate.

4.2.2 Chemistry Factor Calculation

The chemistry factor (CF) is determined using Tables 1 and 2 of Reference 8 for the welds and base metal (plates), respectively. Chemistry factor values determined for each of these PVNGS materials and the respective initial RT_{NDT} values are reported in Tables 4-2 through 4-4.

Surveillance data, when available, can be used to determine a chemistry factor according to the following equations from Reference 8:

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

where:

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

and:

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

One test of the validity of the estimated chemistry factor consists of calculating ΔRT_{NDT} for a given fluence and comparing it with the measured ΔRT_{NDT} for that fluence. The measured ΔRT_{NDT} must fall within plus or minus one sigma (σ_Δ) of the calculated ΔRT_{NDT}, where σ_Δ = 17°F for base metal and σ_Δ = 28°F for welds (Reference 8). That assessment is described in Section 7.

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

Adjusted reference temperatures are calculated using the following equation from Reference 8:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad \text{(Eqn. 4-5)}$$

All temperatures in Equation 4-5 are in degrees Fahrenheit.

4.3 ANALYSIS DETAILS

Ferritic materials in the PVNGS reactor vessels that may have accumulated neutron fluence in excess of 1.0E+17 n/cm^2 are assessed according to CE NPSD-683-A, Reference 1. The materials considered comprise the lower, intermediate and upper shell course plates, bottom head plates and included welds.

4.3.1 Selection of Representative and Limiting Cases

For each of the PVNGS units, the ferritic plates and weld materials located within the region immediately surrounding the active core are evaluated to identify the limiting material at the 1/4T and 3/4T locations. That includes materials from the intermediate shell course for which only a small section is within the effective height of the core but receives sufficient neutron irradiation to be considered in the identification of the most limiting beltline material. The adjusted RT_{NDT} values for the limiting beltline materials are used to establish the heat-up and cool-down limits.

The neutron fluence may exceed 1.0E+17 n/cm^2 for the ferritic plate and weld materials located above and below the region immediately surrounding the active core. However, the fluence in that surrounding region is too low to make it more limiting relative to the section within the effective height of the core. The initial RT_{NDT} values of those ferritic materials are comparable to the initial RT_{NDT} values for the materials from the lower and intermediate shell courses; therefore, the adjustment to RT_{NDT} (i.e., the predicted shift) will be smaller than for

the materials surrounding the effective height of the core. Hence, the ferritic plate and weld materials located above and below the region immediately surrounding the effective height of the core in the PVNGS are not limiting with respect to establishing the heat-up and cool-down limits. These additional materials comprise the upper shell course and bottom head plates and welds that are located immediately above the intermediate shell course and below the lower shell course.

Tables 4-5 through 4-7 present the projected ART values at 1/4T and 3/4T locations in the reactor vessel wall for each of the PVNGS units. ART values are determined using the full value of margin, i.e., no credit is taken for the case where the predicted shift was less than two standard deviations ($2\sigma_\Delta$; see Section 4.4). As shown in Table 4-1, the material with the highest predicted ART at 32 EFPY for PVNGS is the Unit 1 intermediate shell plate M-6701-2. Note in Table 4-5 that comparable results are also obtained for plate M-6701-3.

4.3.2 Calculation of Fluence at 1/4T and 3/4T

As stated in Section 4.2.1, the design value of peak fast fluence through 32 EFPY for PVNGS Units 1, 2, and 3 is 3.29E+19 n/cm^2 (E > 1.0 MeV) and applies to the vessel clad-to-base metal interface. Table 4-8 provides the values of fluence and fluence factors through 32 EFPY determined at the 1/4T and 3/4T locations in the vessel wall. These fluence values are calculated using Equation 4-1, and are used in the calculation of ART in Tables 4-5 through 4-7. Fluence factors listed in Table 4-8 are calculated using Equation 4-3. The following values of depth (x) are used for the intermediate and lower shells:

Location	at 1/4T	at 3/4T
Intermediate shell	x = 2.80 in.	x = 8.39 in.
Lower Shell	x = 2.27 in.	x = 6.80 in.

4.4 LIMITING ADJUSTED REFERENCE TEMPERATURES

The adjusted reference temperatures for PVNGS at 32 EFPY are calculated using Equation 4-5. Chemistry factor and initial RT_{NDT} values for each of the PVNGS beltline materials are given in Tables 4-2 through 4-4. They are calculated using the values of fluence (f) and fluence factor (ff) given in Table 4-8. Margin values are derived using Equation 4-6 (from Reference 8) as:

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

The standard deviation for the initial RT_{NDT}, σ_i, is taken as zero, given the use of measured values for each of the PVNGS plates and welds. The standard deviation for the predicted shift, σ_Δ, is 28°F for welds and 17°F for plates. Therefore, with σ_i being zero, the margin becomes twice the value of σ_Δ. Per Reference 8, σ_Δ need not exceed one-half of the predicted shift. The projected ART values for each of these materials given in Tables 4-5 through 4-7 are determined using the full value of margin, i.e., no credit is taken in this case when one standard deviation for shift exceeded one-half the predicted shift.

The highest calculated values of ART for each PVNGS Unit appear in Table 4-1. The highest adjusted reference temperature is used to define the P-T limits for PVNGS Units 1, 2 and 3 through 32 EFPY as described in Section 6.

Similar calculations are performed for the limiting plate and weld at 32 EFPY following the fracture toughness requirements of 10 CFR Part 50.61. RT_{PTS} is calculated for each of the beltline plates and welds using the values of chemistry factor and initial RT_{NDT} shown in Tables 4-2 through 4-4, and margin as

described above. Calculations are based on a peak fluence at the vessel clad-to-base metal interface through 32 EFPY of 3.29E+19 n/cm^2 in both the lower shell and in the intermediate shell. The highest predicted RT_{PTS} is 123°F and corresponds to intermediate shell plate M-6701-2 in Unit 1. Projected RT_{PTS} values through 32 EFPY for the limiting material at each PVNGS Unit are shown in Table 4-1.

Table 4-1
Summary of Limiting ART and RT_{PTS} Values through 32 EFPY

Unit	Location	Material	1/4T ART (°F)	3/4T ART (°F)	RT_{PTS} (°F)
1	Inter. Shell	Plate M-6701-2	116	103	123
2	Inter. Shell	Plate F-765-6	74	64	78
3	Lower Shell	Plate F-6411-2	65	57	68

Table 4-2
ART Input Values for Unit 1 Beltline Materials

Beltline Material	Plate or Weld Number	Chemistry Factor (°F)	Initial RT_{NDT} (°F)
Lower Shell Plate	M-4311-1	26	– 10
Lower Shell Plate	M-4311-2	20	– 40
Lower Shell Plate	M-4311-3	20	– 20
Intermediate Shell Plate	M-6701-1	44	+ 30
Intermediate Shell Plate	M-6701-2	37	+ 40
Intermediate Shell Plate	M-6701-3	37	+ 40
Inter. Shell Axial Weld	101-124	35.45	– 50
Low. Shell Axial Weld	101-142	27.8	– 80
Inter/Low. Girth Weld	101-171	34.05	– 70

Table 4-3
ART Input Values for Unit 2 Beltline Materials

Beltline Material	Plate or Weld Number	Chemistry Factor (°F)	Initial RT_{NDT} (°F)
Lower Shell Plate	F-773-1	20	+ 10
Lower Shell Plate	F-773-2	26	0
Lower Shell Plate	F-773-3	31	– 60
Intermediate Shell Plate	F-765-4	20	– 20
Intermediate Shell Plate	F-765-5	20	+ 10
Intermediate Shell Plate	F-765-6	26	+ 10
Inter. Shell Axial Weld	101-124	33.6	– 60
Low. Shell Axial Weld	101-142	44.2	– 80
Inter/Low. Girth Weld	101-171	26.55	– 30

Table 4-4
ART Input Values for Unit 3 Beltline Materials

Beltline Material	Plate or Weld Number	Chemistry Factor (°F)	Initial RT_{NDT} (°F)
Lower Shell Plate	F-6411-1	26	– 40
Lower Shell Plate	F-6411-2	26	0
Lower Shell Plate	F-6411-3	26	– 60
Intermediate Shell Plate	F-6407-4	26	– 30
Intermediate Shell Plate	F-6407-5	31	– 20
Intermediate Shell Plate	F-6407-6	26	– 20
Inter. Shell Axial Weld	101-124	25.9	– 50
Low. Shell Axial Weld	101-142	30.65	– 50
Inter/Low. Girth Weld	101-171	34.05	– 70

Table 4-5
Predicted ART Values for Unit 1 Beltline Materials through 32 EFPY

Beltline Material	Plate or Weld Number	Predicted 1/4T Shift (°F)	1/4T ART (°F)	Predicted 3/4T Shift (°F)	3/4T ART (°F)
Lower Shell Plate	M-4311-1	31	55	23	47
Lower Shell Plate	M-4311-2	24	18	18	12
Lower Shell Plate	M-4311-3	24	38	18	32
Intermediate Shell Plate	M-6701-1	50	114	34	98
Intermediate Shell Plate	M-6701-2	42	116	29	103
Intermediate Shell Plate	M-6701-3	42	116	29	103
Inter. Shell Axial Weld	101-124	41	47	27	33
Lower Shell Axial Weld	101-142	33	9	24	0
Inter./Low. Girth Weld	101-171	40	26	30	16

Table 4-6
Predicted ART Values for Unit 2 Beltline Materials through 32 EFPY

Beltline Material	Plate or Weld Number	Predicted 1/4T Shift (°F)	1/4T ART (°F)	Predicted 3/4T Shift (°F)	3/4T ART (°F)
Lower Shell Plate	F-773-1	24	68	18	62
Lower Shell Plate	F-773-2	31	65	23	57
Lower Shell Plate	F-773-3	36	10	27	1
Intermediate Shell Plate	F-765-4	23	37	15	29
Intermediate Shell Plate	F-765-5	23	67	15	59
Intermediate Shell Plate	F-765-6	30	74	20	64
Inter. Shell Axial Weld	101-124	38	34	26	22
Lower Shell Axial Weld	101-142	52	28	39	15
Inter./Low. Girth Weld	101-171	31	57	23	49

Table 4-7
Predicted ART Values for Unit 3 Beltline Materials through 32 EFPY

Beltline Material	Plate or Weld Number	Predicted 1/4T Shift (°F)	1/4T ART (°F)	Predicted 3/4T Shift (°F)	3/4T ART (°F)
Lower Shell Plate	F-6411-1	31	25	23	17
Lower Shell Plate	F-6411-2	31	65	23	57
Lower Shell Plate	F-6411-3	31	5	23	-3
Intermediate Shell Plate	F-6407-4	30	34	20	24
Intermediate Shell Plate	F-6407-5	35	50	24	38
Intermediate Shell Plate	F-6407-6	30	44	20	34
Inter. Shell Axial Weld	101-124	30	36	20	26
Lower Shell Axial Weld	101-142	36	42	27	33
Inter./Low. Girth Weld	101-171	40	26	30	16

Table 4-8
Fluence and Fluence Factors at 1/4T and 3/4T through 32 EFPY

Location	1/4T f (n/cm^2)(1)	1/4T ff(2)	3/4T f (n/cm^2)(1)	3/4T ff(2)
Inter. Shell	1.681E+19	1.1431	4.390E+18	0.7711
Lower Shell	1.910E+19	1.1770	6.438E+18	0.8766

(1) f = fast neutron fluence.
(2) ff = fluence factor per Equation 4-3.

5.0 RCS PRESSURE-TEMPERATURE LIMITS

RCS pressure-temperature limits developed for PVNGS employ the analytical methodology approved by the NRC in CE NPSD-683-A, Reference 1, and use conservative reactor vessel beltline residual fracture toughness values based on those obtained from post-irradiation examination of surveillance specimens.

The pressure-temperature limits for the beltline region are combined with non-beltline regions, as appropriate, to develop the set of composite curves for the operational modes at PVNGS. The lower bound of these composite curves defines the pressure-temperature limit for PVNGS Units 1, 2 and 3 for each operational mode.

Pressure-temperature limits established for non-beltline locations do not change significantly as compared with the most limiting 1/4T and 3/4T beltline locations during the vessel lifetime due to the lower exposure to neutron flux experienced at non-beltline locations. Non-beltline locations are considered when updating RCS pressure-temperature limits throughout plant life since these locations are embodied in the plant design basis.

5.1 STRESS INTENSITY FACTOR

5.1.1 General

The analytical procedure for developing the PVNGS reactor vessel beltline region P-T limits uses the methods of linear elastic fracture mechanics and the guidance found in ASME Section XI Appendix G, Reference 25. For these analyses, the Mode I ("crack opening" mode) stress intensity factors are used for the solution basis coupled with a superposition technique and the influence coefficient methodology described in CE NPSD-683-A.

Pressure-temperature limit curves for the reactor coolant system are established based on the beltline limits for a family of heatup and cooldown rates. These P-T curves are then corrected to account for pressure elevation effects and instrument measurement uncertainty. The adjustment addresses RCS hydraulic pressure drop due to flow and corrects for the change in elevation between the vessel beltline region and the pressurizer, as well as for temperature and pressure instrument uncertainties.

The final P-T limits also identify the minimum boltup temperature, lowest service temperature and the flange limit. Both the minimum boltup temperature and the lowest service temperature are determined using the available material information. Minimum temperature requirements for P-T limits are determined using the criteria established by 10 CFR 50 Appendix G. LTOP enable temperatures are determined using the heat transfer results and applying the ASME Code criteria of Reference 25.

5.1.2 Pressure-Temperature Limits Calculation

Combustion Engineering developed a finite element methodology to calculate the allowable pressures due to membrane stress intensity factors in reactor pressure vessels. This methodology, approved by the staff in report CE NPSD-683-A, is used to calculate the PVNGS reactor vessel beltline pressure-temperature and LTOP limits during RCS heatup, cooldown, isothermal, and test conditions. The application of this methodology is briefly described in the following sections.

Heat Transfer Analysis

A one-dimensional, multi-node, finite element heat transfer model is used to compute the thermal gradients in the reactor vessel beltline region. The model is solved numerically to establish the temperature distribution through the vessel wall as a function of radius, time, and heatup or cooldown rates. A convective boundary condition is applied to the inside wall of the vessel and an insulated boundary on the outside vessel wall.

Variation in material properties are modeled using average properties for each material over the temperature range of interest assumed in the analysis.

Crack Tip Stress Intensity Factors for Membrane Tensile Pressure Stresses

Two-dimensional finite element models, one with a postulated crack at one-quarter of the vessel wall thickness and a second with a crack located at three-quarter of the vessel wall thickness are employed to analyze the crack tip stress intensity. The methodology for calculating the stress intensity factor corresponding to membrane tension resulting from pressure loading of the reactor vessel, K_{IM}, involves these finite element models.

These finite element crack models are loaded with internal pressure on the cylindrical inside surface. For the case with an inside crack, pressure was also applied to the crack face. Membrane stress intensity factor influence coefficients, K_{IM}, are then computed for a unit internal pressure of 1,000 psi. Finally, the K_{IM} coefficients are extracted from the contour J-integrals assuming a crack aspect ration of 1:6 and corrected using ASME Code ratios to account for three-dimensional versus two-dimensional flaws. Corrected values are used to calculate the stress intensity factors at any applied internal pressure. The membrane stress intensity factor coefficients, K_{IM} values, due to unit (1,000 psi) internal pressure loadings for PVNGS reactor vessel geometry are then used in place of the simplified pressure stress and the K_{IM} values available in the ASME Code.

Crack Tip Stress Intensity Factors for Thermal Stresses

A temperature profile-based superposition technique is used to establish the crack tip stress intensity factor due to thermal stress, with a third order polynomial fit used to model the temperature distributions in the wall. Unit stress intensity factors, K_I, are calculated for each term of the polynomial using a two-dimensional finite element modeling method. These unit values are then summed to determine the total K_I value for the applied loads under any general temperature profile in the wall that occurs during the thermal transient.

Temperature-based influence coefficients are used to determine the thermal stress intensity factor at any location. Using methods from CE NPSD-683-A, these coefficients are computed using a two-dimensional reactor vessel model with a crack adjusted to account for three-dimensional effects.

5.2 FRACTURE TOUGHNESS CRITERIA

The reference pressure stress intensity, K_{IR}, is obtained from a reference fracture toughness curve for reactor vessel low alloy steels and is defined in Appendices A and G of Section XI of the ASME Code. This reference pressure stress intensity is determined by two properties, K_{IA} and K_{IC} that represent critical values of the stress intensity factor. In this report, K_{IR} is defined as K_{IC}, with K_{IC} defined as the lower bound of static initiation critical K_I values measured as a function of temperature.

ASME Code Case N-640 permits application of the lower bound static crack initiation critical stress intensity factor equation (i.e., K_{IC} equation) as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest critical stress intensity factor equation (i.e., K_{IA}) which is based on conditions needed to arrest a propagating crack, and which is the method invoked by Appendix G to Section XI of the ASME Code. Use of the K_{IC} equation to determine the lower bound fracture toughness when computing P-T curves is more technically correct than the use of the K_{IA} equation since the rate of loading during a heatup or cooldown is slow, and since crack initiation, which is more representative of a static condition than a dynamic condition, is principally at issue. The K_{IC} equation appropriately implements the use of the static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel.

5.3 TRANSIENT PRESSURE-TEMPERATURE LIMITS

A thermal stress intensity factor, K_{IT}, is calculated at the 1/4T and 3/4T crack tip locations for any instant during the plant heatup or cooldown. Typically, thermal stress intensity factors are calculated using the temperature profile through the vessel wall as a function of time. These stress intensity factors are subtracted from the available K_{IC} value to find the allowable pressure stress intensity factor and, consequently, the allowable pressure. For PVNGS, the allowable pressure is defined as:

$$P_{all} = \frac{K_{IC} - K_{IT}}{2k_{IM}} \quad \text{for heatup and cooldown transients, and} \qquad \text{(Eqn. 5-1)}$$

$$P_{all} = \frac{K_{IC}}{2k_{IM}} \quad \text{for steady-state (isothermal) conditions.} \qquad \text{(Eqn. 5-2)}$$

where:

k_{IM} = membrane load stress intensity factor due to unit internal pressure, ksi√in/psia,
K_{IT} = thermal load stress intensity factor, ksi√in,
K_{IC} = fracture toughness, ksi√in, and
P_{all} = allowable RCS pressure, psia.

Isothermal and transient conditions are analyzed at the selected ART values. Cooldown transients are analyzed at rates of 10°F/hr, 20°F/hr, 30°F/hr, 40°F/hr, 50°F/hr, 75°F/hr and 100°F/hr, beginning at a bulk coolant temperature of 550°F and terminating at 70°F. Heatup transients are analyzed at rates of 10°F/hr, 20°F/hr, 30°F/hr, 40°F/hr, 50°F/hr, and 75°F/hr beginning at a bulk temperature of 70°F and terminating at 550°F. Hydrostatic limits are obtained only for isothermal conditions.

5.4 CRACK TIP PRESSURE-TEMPERATURE LIMITS

Crack tip pressure-temperature limits are developed by first conducting a heat transfer analysis of limiting heatup and cooldown transients. Results are extracted from the heat transfer analyses, including through-wall thermal gradient profiles and metal temperatures at the 1/4T and 3/4T crack tip locations. Thermal stress intensity factors, K_{IT}, for the heatup and the cooldown transients are then computed from the through-wall thermal gradients. Next, the allowable fracture toughness is computed based on the crack tip metal temperatures. Finally, the limiting allowable pressures for transient and steady-state conditions are computed using the above equations. The most limiting allowable pressure enveloped for each startup and cooldown transient is evaluated considering that during heatup, the thermal bending stress is compressive at the reactor vessel inside wall and tensile at the reactor vessel outside wall. Internal pressure creates a tensile stress at the inside wall and outside wall locations with the outside wall location having the larger total stress. However, neutron embrittlement, shift in material RT_{NDT}, and reduction in fracture toughness are greater at the inside location than the outside. Therefore, results from both the inside and outside flaw locations must be compared to ensure that the most limiting condition is recognized.

During cooldown, membrane and thermal bending stresses act together in tension at the reactor vessel inside wall. This results in the pressure stress intensity factor, K_{IM}, and the thermal stress intensity factor, K_{IT}, acting in unison to amplify the stress intensity. Tensile pressure stress and compressive thermal stress act in opposition at the reactor vessel outside wall, resulting in a lower total stress at the outside as compared to the inside wall location. Also, the shift in RT_{NDT}, the reduction in fracture toughness, and neutron embrittlement are less severe at the outside wall than at the inside wall location. Consequently, the inside flaw location is more limiting for the cooldown event.

Although allowable pressures for some transients are limited by steady-state conditions, calculating the crack tip P-T limits by the above method, i.e., enveloping the isothermal and all events at 1/4T and 3/4T locations and including the heatup or cooldown rates below the value under consideration, is more conservative than the ASME Appendix G method.

5.5 HYDROSTATIC AND LEAK TEST PRESSURE-TEMPERATURE LIMITS

The purpose of the hydrostatic test limit is to establish the minimum temperature required at the corresponding hydrostatic test pressure. The in-service hydrostatic test for CE NSSS designs is based on a test pressure corresponding to 1.1 times the operating pressure, with the reactor core not critical.

Pressure-temperature limits for the hydrostatic and leak tests are determined using K_{IM} values due to the applied pressure loading at isothermal conditions per ASME Code procedure. A safety factor of 1.5 is applied to K_{IM} along with the condition that $1.5*K_{IM}$ must be less than or equal to K_{IC} when establishing the maximum allowable P-T limits for these tests.

A gradual reactor coolant system temperature change of 10°F in any 1-hour period is assumed to induce negligible thermal stresses. Therefore, a change in RCS temperature of ±10°F in any 1-hour period is the maximum permitted during inservice hydrostatic and leak testing.

5.6 MINIMUM BOLTUP TEMPERATURE

The minimum reactor vessel flange boltup temperature is defined as the initial RT_{NDT} temperature for the limiting material in the stressed (flange) region, plus any effects of irradiation. Since there is no meaningful irradiation effect in the flange region, the minimum boltup temperature for PVNGS is set as the initial RT_{NDT} of 60°F plus the instrument uncertainty of 13.2°F, or 73.2°F. For practicality and conservatism, a value of 80°F is set as the minimum boltup temperature for PVNGS. This minimum boltup temperature is applicable for pressures less than 20% of the pre-service hydrostatic pressure.

5.7 MINIMUM PRESSURE REQUIREMENT

Appendix G of 10 CFR 50 specifies fracture toughness requirements for ferritic materials used in reactor coolant system pressure-retaining components in order to assure the pressure boundary integrity over its service lifetime. The minimum pressure as defined in 10 CFR 50 Appendix G is 20% of the pre-service hydrostatic test pressure. For PVNGS, this minimum uncorrected design pressure is 20% x (2500 psia x 1.25) or 625 psia.

When corrected for pressure elevation effects and temperature measurement uncertainty, the minimum pressure requirement at PVNGS becomes:

For $T \leq 200°F$, $P \leq 750$ psia,

$T_{RCS} = 200 + 13.2 = 213.2°F$

$20\% P_{pre\text{-}hydro} = (625 - 111) = 514$ psia

For $T > 200°F$, $P \leq 750$ psia,

$T_{RCS} = 200 + 13.2 = 213.2°F$

$20\% P_{pre\text{-}hydro} = (625 - 128) = 497$ psia

5.8 LOWEST SERVICE TEMPERATURE

The lowest service temperatures for piping, pumps, and valves with material thickness greater than 2.5 inches is specified in ASME Section III Article NB-3211 as equal to the highest RT_{NDT} plus 100°F, Reference 26. For PVNGS, the highest RT_{NDT} for piping, pumps, and valves is 40°F, Reference 27. With instrument uncertainty included, the lowest service temperature becomes (40°F + 100°F + 13.2°F =) 153.2°F.

5.9 FLANGE LIMITS

Minimum required temperatures for the allowable P-T limits depend on the highest RT_{NDT} of the closure flange region in the highly stressed regions. For PVNGS Units 1, 2 and 3, the highest flange region RT_{NDT} value is 60°F. This value is considered in generating the minimum temperature limits for the hydrostatic test, and for normal operation with and without the core critical.

5.10 MINIMUM TEMPERATURE REQUIREMENTS

Pressure-temperature limits for the reactor pressure vessel flange for in-service hydrostatic and leak tests, heatup and cooldown transients with the core not critical are directly evaluated using 10 CFR Part 50 Appendix G Criteria 1a or 1b (hydro and leak tests) and Criteria 2a or 2b (normal operation including heatup and cooldown). These criteria establish that when the RCS pressure is less than or equal to 20% of the pre-service hydrostatic test pressure, the minimum reactor vessel temperature must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload. When the RCS pressure is greater than 20% of the pre-service hydrostatic test pressure, the minimum reactor vessel temperature must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stresses by bolt preload plus 90°F for hydro or leak testing, or plus 160°F for normal operation, including heatup and cooldown.

Hydrostatic and Leak Tests (Appendix G, Criteria 1a and 1b)

Minimum temperature requirements for hydrostatic and leak tests with fuel in the vessel and the core not critical are given by:

$T_{min} \geq RT_{NDT\text{-}flange}$ ($P \leq 20\%\ P_{pre\text{-}hydro}$), and

$T_{min} \geq RT_{NDT\text{-}flange} + 90°F$ ($P > 20\%\ P_{pre\text{-}hydro}$),

where:

T_{min} is the minimum temperature limit required for the flange,
$RT_{NDT\text{-}flange}$ is the highest reference temperature of the material in the closure flange region, and
$P_{pre\text{-}hydro}$ is the pre-service hydrostatic test pressure.

Applying these minimum temperature limits to PVNGS Units 1, 2 and 3 results in:

Criteria 1a: Pressure ≤ 20% Pre-Service Hydrostatic Test Limit

$T_{min\text{-}Hydro} = RT_{NDT\text{-}flange} + \Delta T_{uncertainty}$
$= 60°F + 13.2°F = 73.2°F$ (A boltup temperature of 80°F is used.)

Criteria 1b: Pressure > 20% Pre-Service Hydrostatic Test Limit

$T_{min\text{-}Hydro} = RT_{NDT\text{-}flange} + 90°F + \Delta T_{uncertainty}$
$= 60°F + 90°F + 13.2°F = 163.2°F$

Normal Operation – Core Not Critical (Appendix G, Criteria 2a and 2b)

For normal operation, including heatup and cooldown, the minimum temperature requirements are evaluated using the criteria in 10 CFR 50 Appendix G. For the core not critical condition, the minimum temperature requirements are given by:

$T_{min} \geq RT_{NDT\text{-}flange}$ ($P \leq 20\%\ P_{pre\text{-}hydro}$), and

$T_{min} \geq RT_{NDT\text{-}flange} + 120°F$ ($P > 20\%\ P_{pre\text{-}hydro}$).

Applying these limits to PVNGS Units 1, 2 and 3 produces:

Criteria 2a: Pressure ≤ 20% Pre-Service Hydrostatic Test Limit

$T_{min\text{-}NoP}$ $= RT_{NDT\text{-}flange} + \Delta T_{uncertainty}$

$= 60°F + 13.2°F = 73.2°F$ (A boltup temperature of 80°F is used.)

Criteria 2b: Pressure > 20% Pre-Service Hydrostatic Test Limit

$T_{min\text{-}NoP}$ $= RT_{NDT\text{-}flange} + 120°F + \Delta T_{uncertainty}$

$= 60°F + 120°F + 13.2°F = 193.2°F$

These minimum temperatures are incorporated into the P-T limits for the hydrostatic test.

5.11 TEMPERATURE REQUIREMENTS FOR NORMAL OPERATION

Minimum temperature criteria for core critical operation, established by 10 CFR 50 Appendix G Criteria 2c and 2d, specify the following P-T limits. In the case when the RCS pressure is less than or equal to 20% of the pre-service hydrostatic test pressure, the minimum reactor vessel temperature must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stressed by bolt preload plus 40°F, or the minimum permissible temperature for the in-service hydrostatic pressure test, whichever is larger. When the RCS pressure is greater than 20% of the pre-service hydrostatic test pressure, the minimum reactor vessel temperature must be at least as high as the RT_{NDT} for the limiting material in the closure flange region stresses by bolt preload plus 160°F, or the minimum permissible temperature for the in-service hydrostatic pressure test, whichever is larger.

When the core is critical, minimum temperatures required for PVNGS Units 1, 2 and 3 are given by:

Criteria 2c: Pressure ≤ 20% Pre-Service Hydrostatic Test Limit

$T_{min\text{-}NoP}$ $= \text{Larger of } ((T_{Hydro}) \text{ or } (RT_{NDT\text{-}flange} + 40°F)) + \Delta T$

$= \text{Larger of } ((168.2°F) \text{ or } (60°F + 40°F)) + 13.2°F = 181.4°F$

Criteria 2d: Pressure > 20% Pre-Service Hydrostatic Test Limit

$T_{min\text{-}NoP}$ $= \text{Larger of } ((T_{Hydro}) \text{ or } (RT_{NDT\text{-}flange} + 160°F)) + \Delta T$

$= \text{Larger of } ((168.2°F) \text{ or } (60°F + 160°F)) + 13.2°F = 233.2°F$

The minimum in-service hydrostatic test temperature (T_{Hydro}, uncorrected) is 168.2°F and corresponds to a pressure of 2,475 psi (uncorrected) from Table 5-1. These minimum temperatures are incorporated into the P-T limits for the heatup transients in Figure 5-1, and for cooldown transients in Figure 5-2.

5.12 LTOP ENABLE TEMPERATURE LIMITS

LTOP enable temperatures are implemented at PVNGS to protect against brittle fracture during reactor start-up and shutdown operations due to low temperature overpressure events for Service Level A or B conditions. Computed LTOP heatup and cooldown enable temperatures are shown in Table 5-3. Note that the shutdown cooling system suction line ("LTOP") relief valves can remain in service above 221°F during heat-up or cooldown until the RCS pressure reaches the maximum SCS operating pressure (cf., TS LCO 3.4.13).

Heatup Transients

ASME Code criteria requires that LTOP systems must be effective at coolant temperatures less than 200°F, or at reactor coolant inlet temperatures corresponding to a reactor vessel metal temperature at the 1/4T crack tip location of less than (RT_{NDT}+50°F), whichever is greater. For heatup transients, the LTOP enable temperature is based on the 1/4T crack tip metal temperature limit of 166°F, (RT_{NDT} = 116°F) which corresponds to a fluid temperature, obtained from heat transfer analysis, of 207.3°F. The temperature lag at the 1/4T crack tip is a

function of the heatup rate, vessel dimensions, and the heat transfer properties used in the analysis. The resulting LTOP heatup enable temperature including uncertainty is 220.5°F (= 207.3°F + 13.2°F). Per ASME Code criteria, the LTOP enable temperature for heatup events is set at the maximum of (200°F; 220.5°F) or 220.5°F, which for operational purposes is rounded to 221°F.

Cooldown Transients

The PVNGS LTOP cooldown enable temperature is conservatively set equal to that of the isothermal case, and is expressed as the maximum of 200°F or RT_{NDT} + 50°F. The vessel metal temperature is taken at a distance one-fourth of the vessel thickness from the inside surface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature for the weld or base metal in the beltline region at 1/4T, measured from the vessel inner surface. This is determined using the procedures of Regulatory Guide 1.99, Revision 2.

Instrumentation correction needs to be considered to arrive at the final LTOP enable temperature. For an isothermal condition at PVNGS Units 1, 2 and 3, including instrument uncertainty, the LTOP enable temperature becomes RT_{NDT} + 50°F + uncertainty = (116°F + 50°F + 13.2°F) = 179.2°F. As this calculated temperature is below the ASME Code requirement of 200°F, the LTOP enable temperature for cooldown transients, including uncertainties, becomes (200°F + 13.2°F) = 213.2°F.

Combined Enable Temperature Limit

For practicality and conservatism, a single temperature value of 221°F is selected as the heatup and cooldown LTOP enable temperature for PVNGS Units 1, 2 and 3 as shown in Table 5-3.

5.13 PRESSURE AND TEMPERATURE CORRECTON FACTORS

All RCS pressure-temperature limits calculated for heatup, cooldown and test conditions are corrected for hydraulic conditions and instrument uncertainty. A constant temperature instrument uncertainty of +13.2°F is applied to all computed RCS temperatures. Different pressure corrections that adjust for differences in elevation between the pressurizer pressure instrument location and the reactor vessel, reactor coolant system flowrate, and pressurizer pressure instrument uncertainty are applied in different regimes of pressure and temperature. These pressure correction factors vary from 111 psi to 169 psi, Reference 27, and apply as shown below.

RCS Temperature	RCS Pressure	Pressure Correction	Temperature Correction
< 200°F	< 750 psia	– 111 psi	+13.2°F
	≥ 750 psia	– 153 psi	+13.2°F
≥ 200°F	< 750 psia	– 128 psi	+13.2°F
	≥ 750 psia	– 169 psi	+13.2°F

The applicable temperature range is consistent with the allowable RCP usage specified by Technical Specifications. That is, within the LTOP enable range, operation of two RCPs is permitted below 200°F and three RCPs at 200°F and above. Similarly, the applicable pressure range is limited by the range of the pressurizer narrow range instrumentation

Indicated RCS pressure-temperature limits are generated by subtracting the pressure correction from the calculated allowable pressure, then adding the temperature correction to the calculated RCS temperature. This correction process is followed to determine the allowable P-T limits for all heatup and cooldown transients, and for RCS test conditions.

5.14 SUMMARY OF RCS PRESSURE-TEMPERATURE LIMITS

Table 5-1, shown plotted in Figure 5-1, provides PVNGS heatup P-T limits corrected for instrument uncertainty through 32 EFPY. Table 5-2, shown plotted in Figure 5-2, illustrates comparable P-T cooldown limits. LTOP enable temperatures through 32 EFPY, corrected for uncertainties, are given in Table 5-3.

Figures 5-3 and 5-4 show the composite P-T limits, including core critical values, corrected for uncertainties. Tables 5-4, 5-5 and 5-6 list the core critical P-T limits, including corrections for instrument uncertainty, for hydrostatic testing and for heatup (75°F/hr) and cooldown (100°F /hr) events.

Figures 5-5 and 5-7 show the thermal gradients at 1/4T and 3/4T for heatup and cooldown events through 32 EFPY. Corresponding heatup and cooldown thermal stress intensity factors are shown in Figures 5-6 and 5-8.

Table 5-1
RCS Pressure and Temperature Heatup Limits through 32 EFPY

Temperature (°F)[1]	Pressure Isothermal (psia)	RCS Pressure (psia) @ Heatup Rate[1]						Hydrostatic Test (psia)
		@10°F/hr	@20°F/hr	@30°F/hr	@40°F/hr	@50°F/hr	@75°F/hr	
80	680.6	680.6	680.6	671.1	650.2	622.2	602.2	954.4
83.2	690.2	690.2	690.2	676.2	650.2	622.2	602.2	967.2
93.2	727.2	727.2	705.2	676.2	650.2	622.2	602.2	1016.2
103.2	772.2	772.2	710.2	676.2	650.2	622.2	602.2	1075.2
113.2	826.2	826.2	735.2	681.2	650.2	622.2	602.2	1148.2
123.2	893.2	893.2	778.2	700.2	653.2	622.2	602.2	1237.2
133.2	974.2	974.2	839.2	738.2	672.2	627.2	602.2	1346.2
143.2	1074.2	1074.2	918.2	790.2	705.2	645.2	602.2	1478.2
153.2	1195.2	1195.2	1018.2	862.2	754.2	676.2	604.2	1640.2
163.2	1344.2	1335.2	1142.2	954.2	819.2	721.2	617.2	1838.2
171.5	1494.8	1467.5	1269.5	1049.0	889.9	772.8	638.0	2039.1
172.1	1507.0	1478.3	1279.9	1057.0	896.0	777.3	598.0	2053.6
173.2	1525.2	1494.2	1295.2	1068.2	904.2	783.2	600.2	2080.2
183.2	1747.2	1689.2	1484.2	1213.2	1014.2	865.2	637.2	2375.2
186.7	1841.7	1772.5	1565.4	1275.5	1062.2	902.0	655.4	2500.0
193.2	2017.2	1927.2	1716.2	1391.2	1151.2	970.2	689.2	
203.2	2347.2	2217.2	1998.2	1610.2	1320.2	1101.2	757.2	
207.0	2500.0	2351.5	2129.3	1713.2	1399.2	1162.4	790.6	
211.2		2500.0	2274.2	1827.0	1486.6	1230.0	827.6	
213.2			2343.2	1881.2	1528.2	1262.2	845.2	
213.2			2327.2	1865.2	1512.2	1246.2	829.2	
217.3			2500.0	1998.9	1616.3	1327.8	874.7	
223.2				2191.2	1766.2	1445.2	940.2	
230.8				2500.0	2008.6	1634.4	1045.8	
233.2					2085.2	1694.2	1079.2	
243.2					2474.2	2000.2	1250.2	
243.7					2500.0	2018.8	1260.8	
253.2						2372.2	1461.2	
256.0						2500.0	1533.4	
263.2							1719.2	
273.2							2034.2	
283.2							2418.2	
284.9							2500.0	

(1) Corrected for instrument uncertainty and for RCS pressure and elevation effects.

Table 5-2
RCS Pressure and Temperature Cooldown Limits through 32 EFPY

Temperature (°F) [1]	RCS Pressure (psia) @ Cooldown Rate [1]							
	Isothermal	@10°F/hr	@20°F/hr	@30°F/hr	@40°F/hr	@50°F/hr	@75°F/hr	@100°F/hr
80	680.6	612.3	589.0	527.1	469.5	416.6	329.2	237.6
83.2	690.2	623.2	601.2	541.2	485.2	433.2	329.2	272.2
90.9	718.6	655.4	638.0	583.4	533.5	492.2	402.8	372.6
91.3	720.1	657.2	598.0	585.7	536.1	495.4	406.8	378.1
93.2	727.2	665.2	607.2	596.2	548.2	510.2	425.2	403.2
99.6	756.1	698.0	644.5	638.0	597.1	559.7	501.1	493.2
99.9	757.5	699.6	646.3	598.0	599.4	562.1	504.7	497.5
103.2	772.2	716.2	665.2	619.2	624.2	587.2	543.2	543.2
104.7	780.4	725.6	676.1	631.3	638.0	604.8	565.0	565.0
104.9	781.6	727.0	677.7	633.1	598.0	607.3	568.2	568.2
107.6	795.8	743.4	696.7	654.2	622.1	638.0	606.3	606.3
107.8	796.8	744.4	698.0	655.6	623.6	598.0	608.7	608.7
109.8	807.8	757.0	712.6	671.9	642.1	621.6	638.0	638.0
109.9	808.5	757.9	713.6	673.0	643.4	623.2	598.0	598.0
113.2	826.2	778.2	737.2	699.2	673.2	661.2	645.2	645.2
123.2	893.2	854.2	823.2	798.2	781.2	776.2	776.2	776.2
133.2	974.2	947.2	929.2	918.2	918.2	918.2	918.2	918.2
143.2	1074.2	1060.2	1057.2	1057.2	1057.2	1057.2	1057.2	1057.2
153.2	1195.2	1195.2	1195.2	1195.2	1195.2	1195.2	1195.2	1195.2
163.2	1344.2	1344.2	1344.2	1344.2	1344.2	1344.2	1344.2	1344.2
173.2	1525.2	1525.2	1525.2	1525.2	1525.2	1525.2	1525.2	1525.2
183.2	1747.2	1747.2	1747.2	1747.2	1747.2	1747.2	1747.2	1747.2
193.2	2017.2	2017.2	2017.2	2017.2	2017.2	2017.2	2017.2	2017.2
203.2	2347.2	2347.2	2347.2	2347.2	2347.2	2347.2	2347.2	2347.2
207.1	2500.0	2500.0	2500.0	2500.0	2500.0	2500.0	2500.0	2500.0

[1] Corrected for instrument uncertainty and for RCS pressure and elevation effects.

Table 5-3
LTOP Enable Temperature Limits through 32 EFPY

Case	RT_{NDT} 1/4T (°F)	Margin (°F)	1/4T Crack Tip Temperature (°F)	Uncorrected $T_{coolant}$ (°F)	Instrument Uncertainty (°F)	LTOP Enable (°F) [1]	Enable Temperature (°F) [1]
Heatup @ 75°F/hr	116	50	166	207.3	13.2	220.5	220.5
Heatup, Code Minimum				200	13.2	213.2	
Cooldown/Isothermal	116	50	166	166	13.2	179.2	213.2
Cooldown, Code Minimum				200	13.2	213.2	
Combined Heatup and Cooldown Limit, including uncertainty (rounded).							221.0

[1] Corrected for instrument uncertainty.

Table 5-4
Allowable In-service Hydrostatic Test Pressure

RCS Temperature (°F) [1]	P-Allowable (psia) [1]
80	0
80	514
163.2	514
163.2	1838.2
173.2	2080.2
181.4	2306

[1] Corrected temperatures and pressures.

Table 5-5
Core Critical Limits for Heatup at 75°F/hr

RCS Temperature (°F) [1]	P-Allowable (psia) [1]
181.4	0
181.4	514.0
233.2	514.0
233.2	689.2
242.0	748.7
245.7	779.2
252.1	835.7
266.0	979.3
278.9	1176.3
291.4	1423.7
320.3	2306.0

[1] Corrected temperatures and pressures.

Table 5-6
Core Critical Limits for Cooldown at 100°F/hr

RCS Temperature (°F) [1]	P-Allowable (psia) [1]
181.4	0
181.4	514.0
233.2	514.0
233.2	2017.2
235.0	2076.6
240.0	2241.6
242.2	2306.0

[1] Corrected temperatures and pressures.

Figure 5-1
RCS Heatup Pressure-Temperature Limits through 32 EFPY

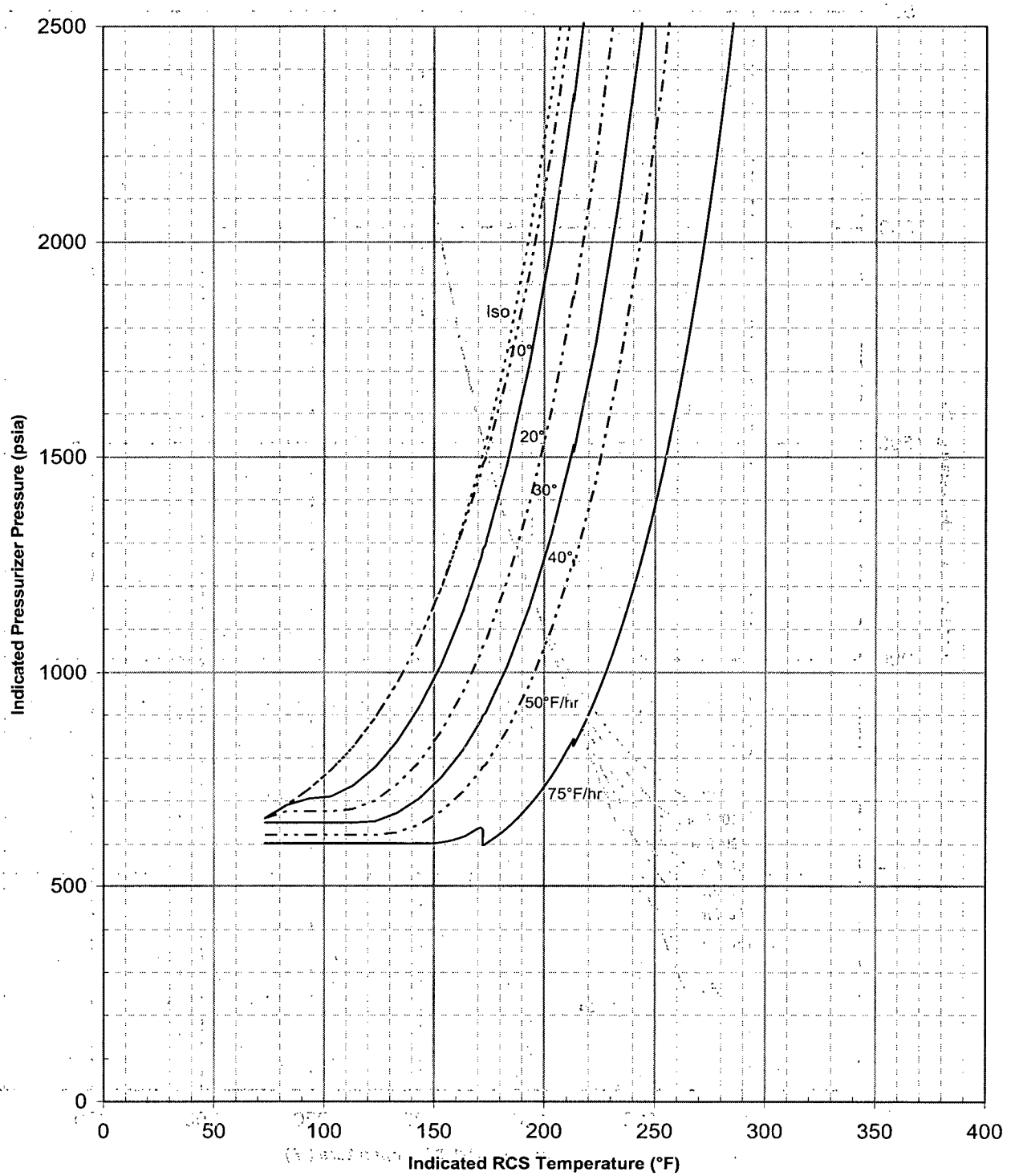


Figure 5-2
RCS Cooldown Pressure-Temperature Limits through 32 EFPY

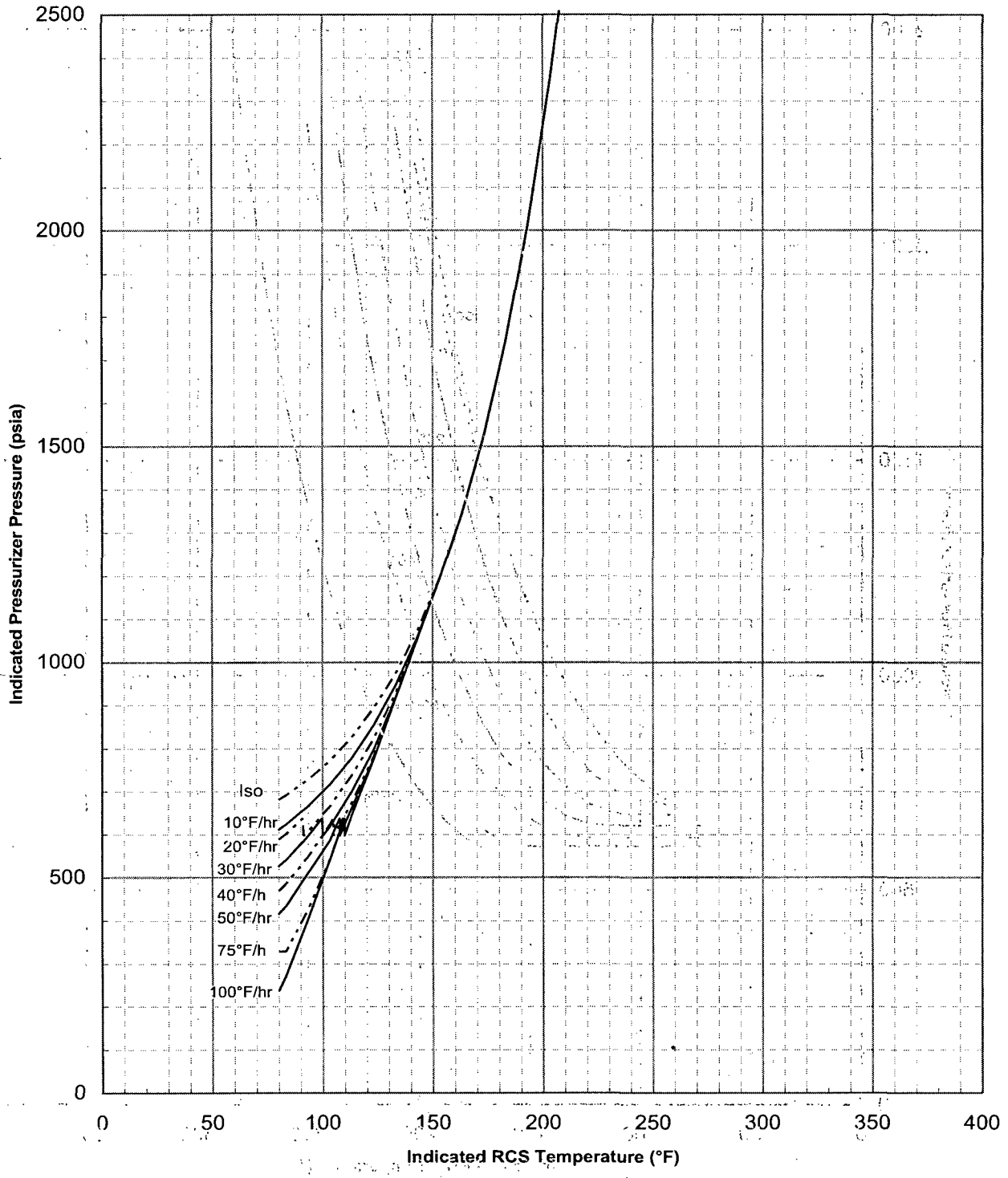


Figure 5-3
RCS Composite Pressure-Temperature Heatup Limits through 32 EFPY

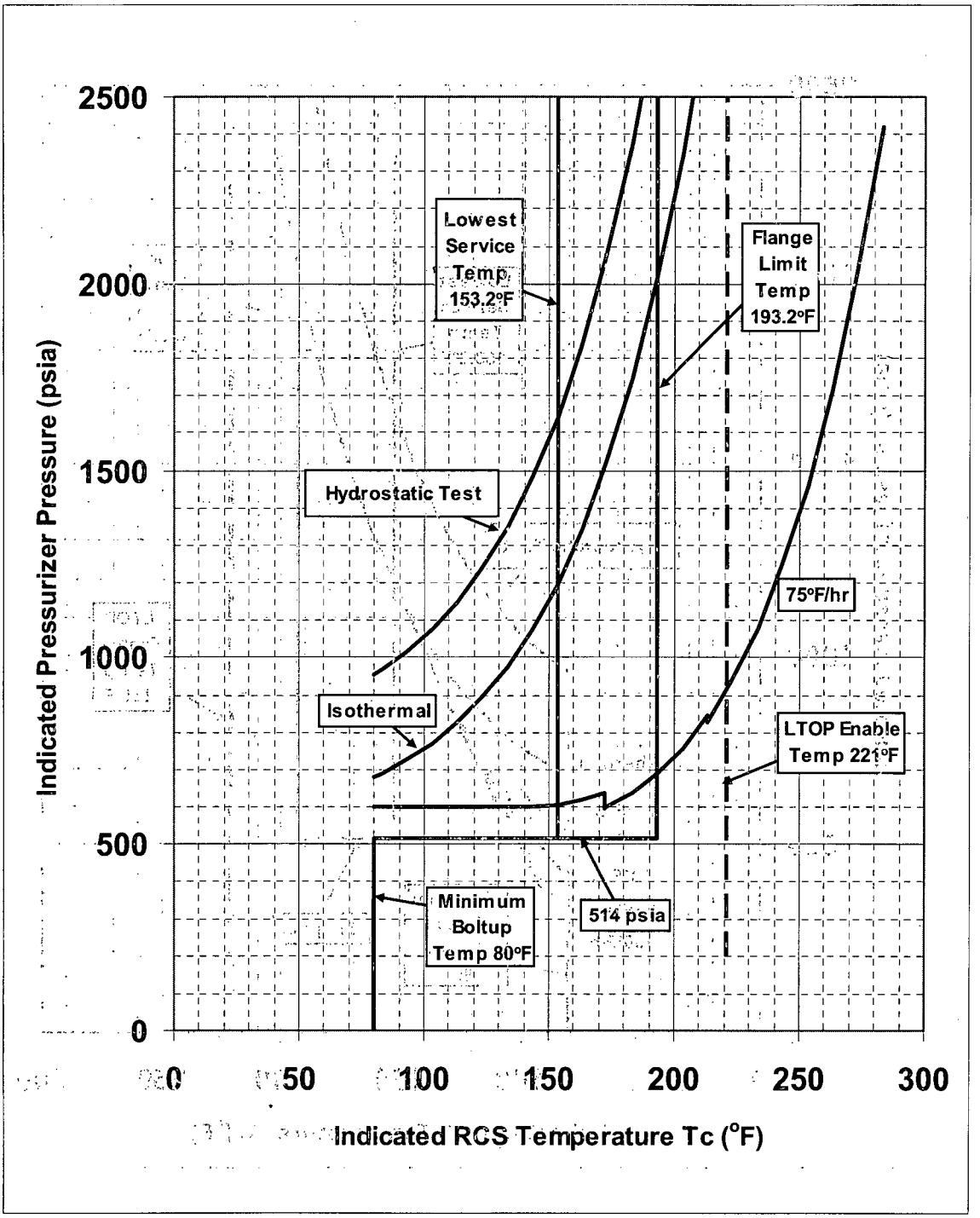


Figure 5-4
RCS Composite Pressure-Temperature Cooldown Limits through 32 EFPY

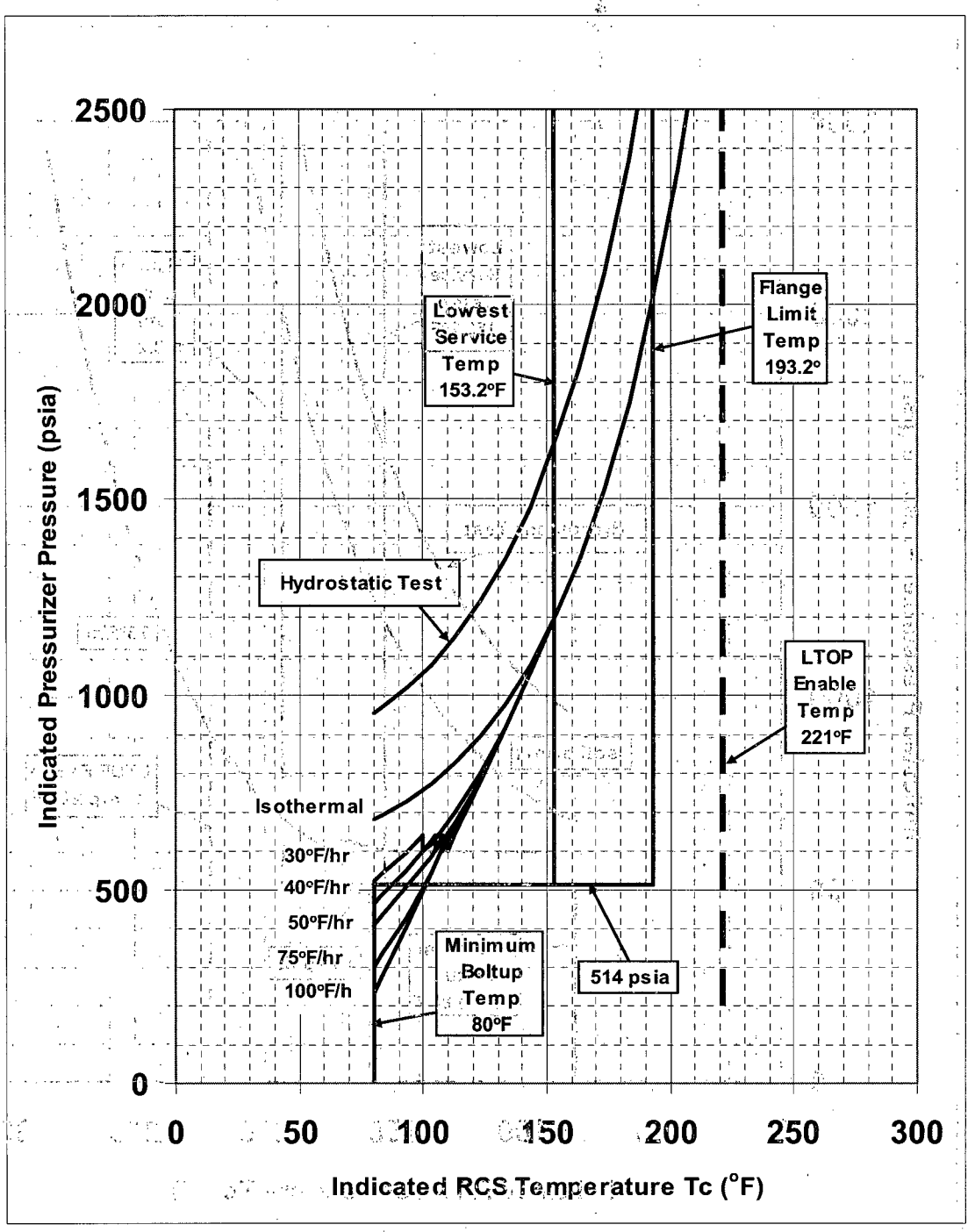


Figure 5-5
RCS Through-wall Thermal Gradients at Crack Tips
for Heatup of 75°F/hr

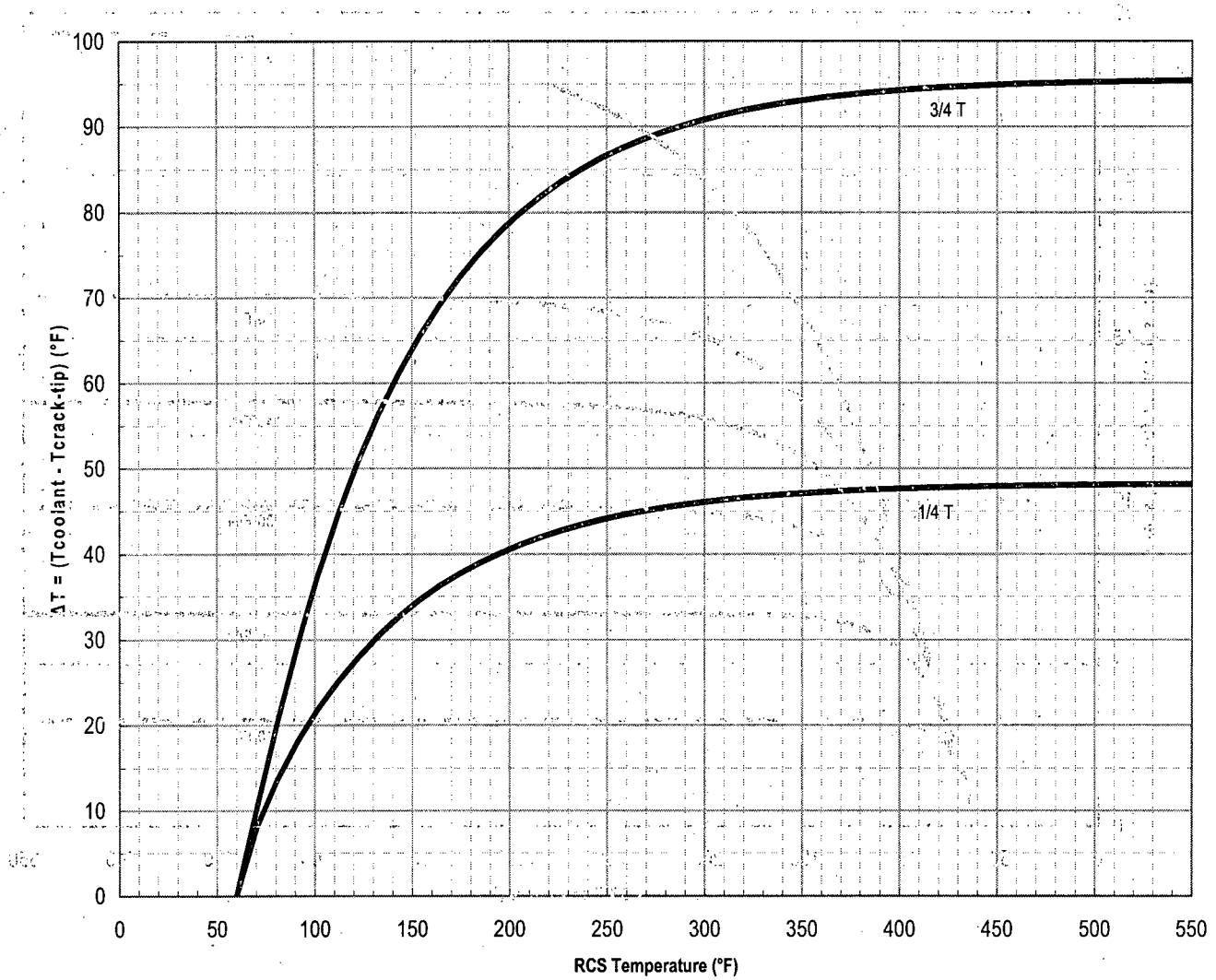


Figure 5-6
RCS Heatup Thermal Stress Intensity Factors at 3/4T Crack Tip

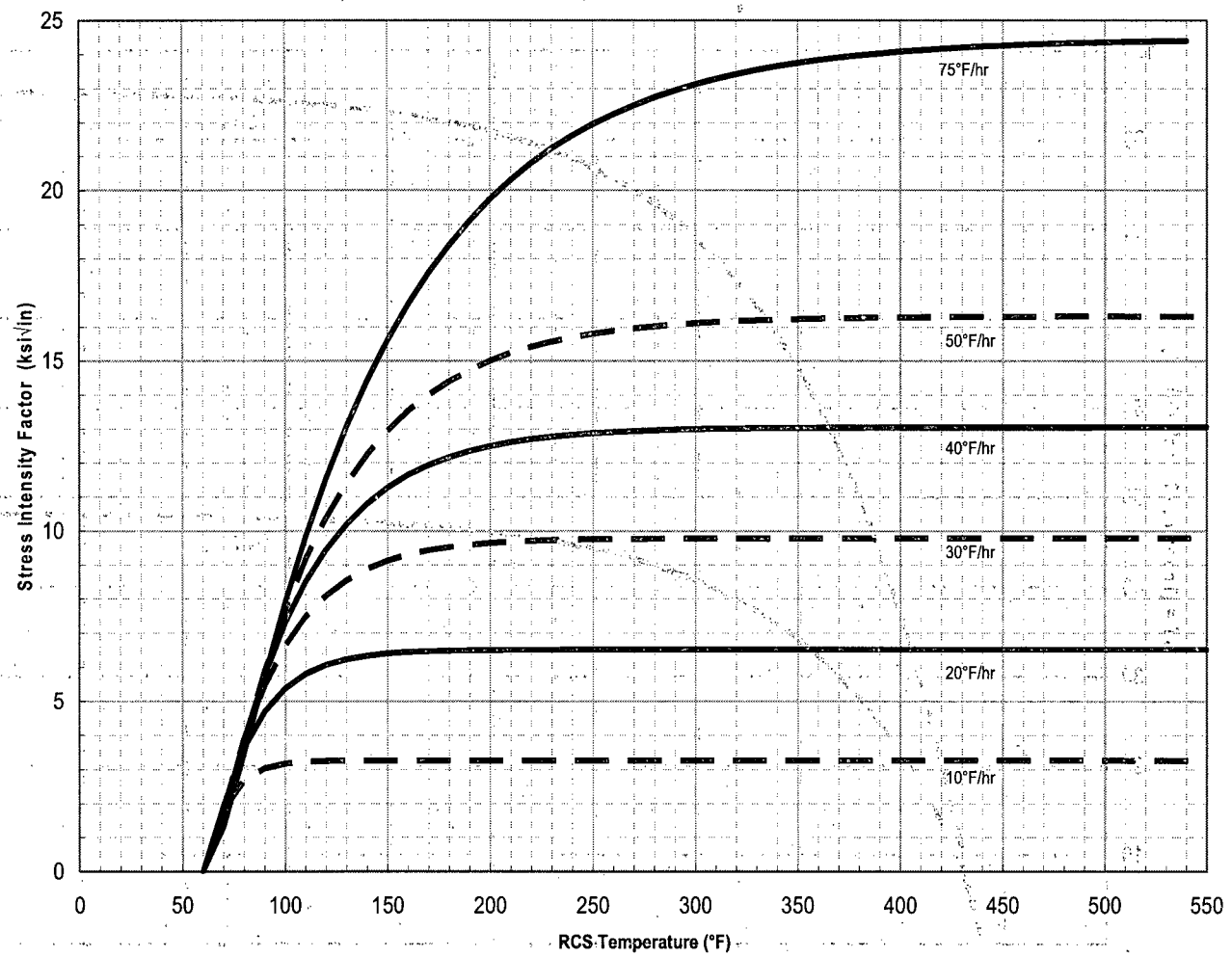


Figure 5-7
RCS Through-wall Thermal Gradients at Crack Tips
for Cooldown of 100°F/hr

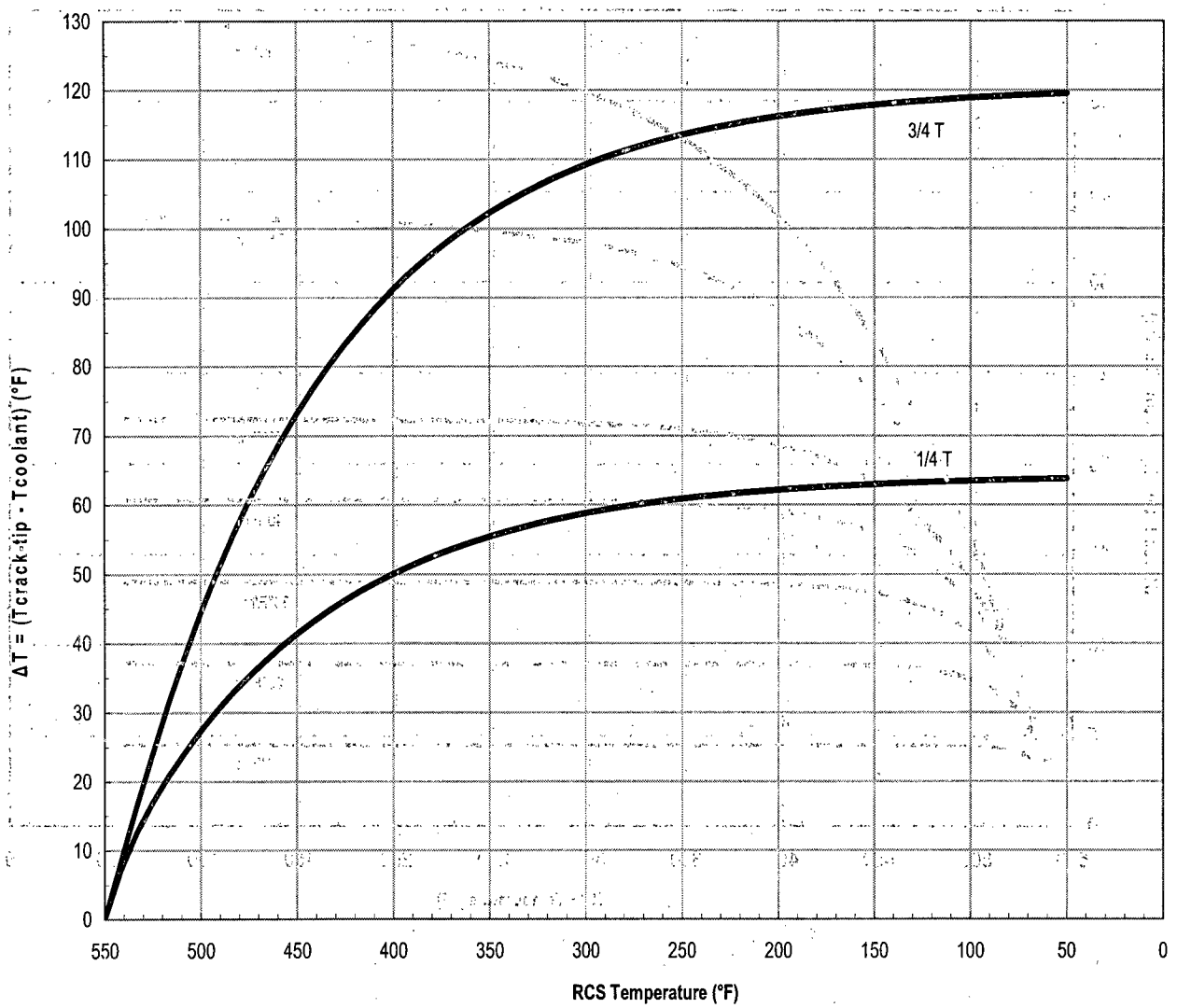
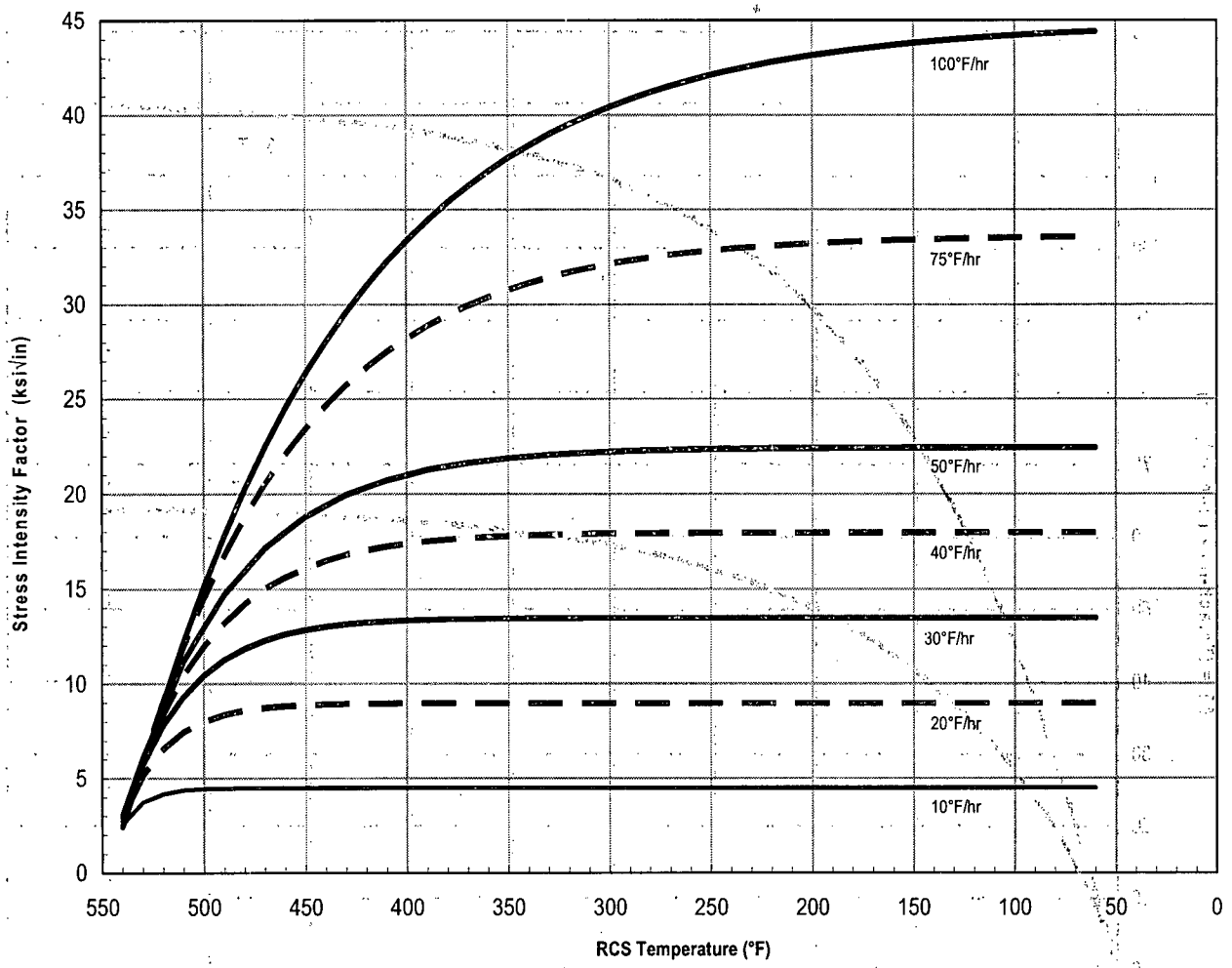


Figure 5-8
RCS Cooldown Thermal Stress Intensity Factors at 1/4T Crack Tip



6.0 MINIMUM TEMPERATURE REQUIREMENTS

The minimum temperature requirements specified in Appendix G of 10 CFR 50 are applied to the pressure-temperature curves using the NRC-approved methodologies as described in Section 6.0 of CE NPSD-683-A, Reference 1.

The lowest service temperature is established for PVNGS based on the limiting RT_{NDT} for the reactor coolant pumps. Also, pressure-temperature limits developed for PVNGS use the more conservative of either the lowest service temperature or other minimum temperature requirement for the reactor vessel when the RCS is pressurized to greater than 20% of the pre-service 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 20% of the pre-service hydrostatic test pressure. For PVNGS, the pre-service hydrostatic test pressure is defined as 1.25 times the design pressure. When developing these pressure-temperature limits, the minimum pressure establishes the point of transition between the various temperature-only based pressure-temperature limits (for example: minimum boltup and the lowest service temperature or flange limits).

For PVNGS Units 1, 2 and 3, the minimum (uncorrected) pressure of 625 psia is calculated in Section 5.0. The limiting minimum pressure for PVNGS Units 1, 2 and 3 through 32 EFPY, corrected for instrument uncertainty, elevation and flow, is shown in Table 6-1. Table 6-2 lists the minimum indicated temperature values applied to the pressure-temperature curves of PVNGS Units 1, 2 and 3 through 32 EFPY.

Table 6-1
Minimum Indicated RCS Pressure through 32 EFPY

Temperature(1)	Minimum Pressure(1)
$T_{RCS} < 200°F$	$P_{RCS} = 514$ psia
$T_{RCS} \geq 200°F$	$P_{RCS} = 497$ psia
(1) Corrected temperatures and pressures.	

Table 6-2
Minimum Indicated RCS Temperature through 32 EFPY

Requirement	Minimum Temperature(1)
Minimum Boltup Temperature	80°F
Minimum Hydrostatic Test Temperature	181.4°F
Lowest Service Temperature	153.2°F
Minimum Flange Limit (Normal Operation)	193.2°F
Minimum Flange Limit (Hydrostatic Test)	163.2°F
(1) Corrected temperatures.	

This page is intentionally blank.

7.0 APPLICATION OF SURVEILLANCE CAPSULE DATA

Post-irradiation surveillance capsule test results for PVNGS units are reported in References 5, 6 and 7. Test results were evaluated with respect to the credibility criteria of Regulatory Guide 1.99, Reference 8. Results of the credibility assessment are:

- The surveillance program plates and welds are those judged to be most likely controlling with regard to radiation-induced embrittlement,
- Charpy data scatter does not cause ambiguity in the determination of the 30 ft-lb shift,
- Measured RT_{NDT} shifts are consistent with the predicted shifts,
- Capsule irradiation temperature matches that of the vessel wall, and
- Correlation monitor data falls within the scatter band for that material and therefore meets the credibility test.

Two or more credible data sets are available for each of the PVNGS units. Those surveillance data sets are used to assess the chemistry factor and the margin term in accordance with the methodology prescribed in Position 2.1 of Regulatory Guide 1.99. Calculated chemistry factor values for the surveillance plate and weld materials are shown in Tables 7-1, 7-2 and 7-3 for PVNGS Units 1, 2 and 3, respectively. The corresponding credibility test for the surveillance capsule measurements and the measured shifts for the correlation monitor material are shown in Tables 7-4, 7-5 and 7-6.

The derived chemistry factor for the Unit 1 surveillance plate (M-6701-2) and surveillance weld (Heat 90071) are 27.5°F and 4.9°F, respectively, as shown in Table 7-1. These chemistry factor values compare conservatively with their respective values determined following Position 1.1 of Regulatory Guide 1.99, chemistry factors of 37°F and 27.8°F. Table 7-4 demonstrates that the surveillance plate and weld measurements are predictable given that the difference between the measured and predicted shift is less than one standard deviation for the predicted shift (17°F for plates and 28°F for welds). Similar conservative results for Unit 2 plate (F-773-1) and surveillance weld (Heat 3P7317) are given in Tables 7-2 and 7-5, and for Unit 3 plate (F-6411-2) and surveillance weld (Heat 4P7869) in Tables 7-3 and 7-6. Therefore, surveillance results are shown to be predictable and credible for each of the PVNGS units.

The correlation monitor materials from each of the PVNGS units meet the credibility test to be within the scatter band of the database for that material. This is demonstrated in Tables 7-4, 7-5 and 7-6 in which the shift measurements available to date are compared to predictions based on a chemistry factor determined following Position 1.1 of Regulatory Guide 1.99. The difference between the measured and predicted shift is less than 19°F for all seven measurements, and the average value of these differences is 7.7°F.

The calculation of adjusted reference temperature, ART, for use in determining pressure-temperature limits is described in Section 4. The most limiting (highest) value of ART from the three PVNGS units is applied to all three units. Even though the information presented in this section demonstrates that the post-irradiation surveillance capsule test results for PVNGS units are credible, the calculation of ART takes no credit for those credible results. This is conservative given that:

- The most limiting (highest) value of ART from the three PVNGS units is applied to all three units, and
- The derived chemistry factors from the credible surveillance data are all conservative relative to the chemistry factors determined following Position 1.1 of Regulatory Guide 1.99.

Supplemental surveillance data are not used in the calculations of the chemistry factor. Therefore, the issue of whether or not the copper and nickel content of the surveillance weld differs from that of the vessel weld, and how such differences are applied to adjusted values of ART, is not applicable.

Table 7-1
Chemistry Factors for Unit 1 Surveillance Plates and Weld Materials

Material	Capsule	Capsule f[1]	ff[2]	ΔRT_{NDT}[3]	ff*ΔRT_{NDT}	$(ff)^2$
Plate M-6701-2 Longitudinal	137	3.65E+18	0.7216	34.2°F	24.678	0.5207
	230	8.76E+18	0.9629	15.3°F	14.734	0.9271
Plate M-6701-2 Transverse	137	3.65E+18	0.7216	13°F	9.386	0.5207
	230	8.76E+18	0.9629	31.9°F	30.720	0.9271
				Sum:	79.518	2.8956
	$CF_{M-6701-2} = \sum(ff * RT_{NDT}) \div \sum(ff)^2 = (79.518 \div 2.8956) = 27.5°F$					
Weld (Heat 90071)	137	3.65E+18	0.7216	0°F[4]	0	0.5207
	38	6.28E+18	0.8697	6.7°F	5.829	0.7563
	230	8.76E+18	0.9629	5.1°F	4.913	0.9271
				Sum:	10.742	2.2041
	$CF_{Weld} = \sum(ff * RT_{NDT}) \div \sum(ff)^2 = (10.742 \div 2.2041) = 4.9°F$					

(1) f = fluence (n/cm^2, E > 1.0 MeV).
(2) ff = fluence factor = $f^{(0.28 - 0.1*\log f)}$
(3) ΔRT_{NDT} values are the measured 30 ft-lb shift values.
(4) Actual value for the weld = – 2.8°F.

Table 7-2
Chemistry Factors for Unit 2 Surveillance Plates and Weld Materials

Material	Capsule	Capsule f[1]	ff[2]	ΔRT_{NDT}[3]	ff*ΔRT_{NDT}	$(ff)^2$
Plate F-773-1 Longitudinal	137	3.87E+18	0.7372	13.3°F	9.804	0.5434
	230	9.92E+18	0.9978	17.7°F	17.660	0.9955
Plate F-773-1 Transverse	137	3.87E+18	0.7372	9.5°F	7.003	0.5434
	230	9.92E+18	0.9978	19.3°F	19.257	0.9955
				Sum:	53.724	3.0778
	$CF_{F-773-1} = \sum(ff * RT_{NDT}) \div \sum(ff)^2 = (53.724 \div 3.0778) = 17.5°F$					
Weld (Heat 3P7317)	137	3.87E+18	0.7372	0 °F	0	0.5434
	230	9.92E+18	0.9978	2.5 °F	2.494	0.9955
				Sum:	2.494	1.5389
	$CF_{Weld} = \sum(ff * RT_{NDT}) \div \sum(ff)^2 = (2.494 \div 1.5389) = 1.6°F$					

(1) f = fluence (n/cm^2, E > 1.0 MeV).
(2) ff = fluence factor = $f^{(0.28 - 0.1*\log f)}$
(3) ΔRT_{NDT} values are the measured 30 ft-lb shift values.

Table 7-3
Chemistry Factors for Unit 3 Surveillance Plates and Weld Materials

Material	Capsule	Capsule f[1]	ff[2]	ΔRT_{NDT}[3]	ff*ΔRT_{NDT}	$(ff)^2$
Plate F-6411-2 Longitudinal	230	9.07E+18	0.9726	6.3°F	6.128	0.9460
Plate F-6411-2 Transverse	142[4]	3.48E+18	0.7090	13.1°F	9.287	0.5026
	230	9.07E+18	0.9726	9.2°F	8.948	0.9460
				Sum:	24.363	2.3947
	$CF_{F-6411-2} = \Sigma(ff * RT_{NDT}) \div \Sigma(ff)^2 = (24.363 \div 2.3947) = 10.2°F$					
Weld (Heat 4P7869)	142[4]	3.48E+18	0.7090	27.5°F	19.497	0.5026
	230	9.07E+18	0.9726	24.1°F	23.440	0.9460
				Sum:	42.937	1.4487
	$CF_{Weld} = \Sigma(ff * RT_{NDT}) \div \Sigma(ff)^2 = (42.937 \div 1.4487) = 29.6°F$					

(1) f = fluence (n/cm^2; E > 1.0 MeV).
(2) ff = fluence factor = $f^{(0.28 - 0.1*\log f)}$
(3) ΔRT_{NDT} values are the measured 30 ft-lb shift values.
(4) Irradiated at 137 degree position in Unit 3 vessel.

Table 7-4
Credibility of Surveillance Measurements for Unit 1

Material	Capsule	CF	ff[2]	Measured Shift (°F)	Predicted Shift (CF*ff) (°F)	Difference (°F)
Plate M-6701-2 Longitudinal	137	27.5	0.7216	34.2	19.8	+ 14.4
	230	27.5	0.9629	15.3	26.5	− 11.2
Plate M-6701-2 Transverse	137	27.5	0.7216	13.0	19.8	− 6.8
	230	27.5	0.9629	31.9	26.5	+ 5.4
Weld (Heat 90071)	137	4.9	0.7216	− 2.8	3.5	− 6.3
	38	4.9	0.8697	6.7	4.3	+ 2.4
	230	4.9	0.9629	5.1	4.7	+ 0.4
Correlation Monitor Material	137	131.7[1]	0.7216	101.3	95.0	+ 6.3
	38	131.7[1]	0.8697	114.1	114.5	− 0.4
	230	131.7[1]	0.9629	129.2	126.8	+ 2.4

(1) Chemistry factor based on 0.174 Cu and 0.665 Ni using Table 2 of Regulatory Guide 1.99, R02.
(2) ff = fluence factor = $f^{(0.28 - 0.1*\log f)}$

Table 7-5
Credibility of Surveillance Measurements for Unit 2

Material	Capsule	CF	ff[2]	Measured Shift (°F)	Predicted Shift (CF*ff) (°F)	Difference (°F)
Plate F-773-1 Longitudinal	137	17.5	0.7372	13.3	12.9	+ 0.4
	230	17.5	0.9978	17.7	17.5	+ 0.2
Plate F-773-1 Transverse	137	17.5	0.7372	9.5	12.9	– 3.4
	230	17.5	0.9978	19.3	17.5	+ 1.8
Weld (Heat 3P7317)	137	1.6	0.7372	0	1.2	– 1.2
	230	1.6	0.9978	2.5	1.6	+ 0.9
Correlation Monitor Material	137	131.7[1]	0.7372	116.0	97.1	+ 18.9
	230	131.7[1]	0.9978	132.4	131.4	+ 1.0

(1) Chemistry Factor based on 0.174 Cu and 0.665 Ni using Table 2 of Regulatory Guide 1.99, R02.
(2) ff = fluence factor = $f^{(0.28 - 0.1*\log f)}$

Table 7-6
Credibility of Surveillance Measurements for Unit 3

Material	Capsule	CF	ff[2]	Measured Shift (°F)	Predicted Shift (CF*ff) (°F)	Difference (°F)
Plate F-5411-2 Longitudinal	230	10.2	0.9726	6.3	9.9	– 3.6
Plate F-6411-2 Transverse	142	10.2	0.7090	13.1	7.2	+ 5.9
	230	10.2	0.9726	9.2	9.9	– 0.7
Weld (Heat 4P7869)	142	29.6	0.7090	27.5	21.0	+ 6.5
	230	29.6	0.9726	24.1	28.8	– 4.7
Correlation Monitor Material	142	131.7[1]	0.7090	82.5	93.4	– 10.9
	230	131.7[1]	0.9726	141.8	128.1	+ 13.7

(1) Chemistry Factor based on 0.174 Cu and 0.665 Ni using Table 2 of Regulatory Guide 1.99, R02.
(2) ff = fluence factor = $f^{(0.28 - 0.1*\log f)}$

8.0 REFERENCES

1. CE NPSD-683-A, Rev 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.

2. NRC Generic Letter 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

3. U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

4. H.D. Lemmel, P.K. McLaughlin, V.G. Pronyaev, "ENDF/B-VI Release 7, The U.S. Evaluated Nuclear Data Library for Neutron Reaction Data by the U.S. National Nuclear Data Center – 1990 including revisions up to April 2000," International Atomic Energy Agency report IAEA-NDS-100, Rev. 10, June 2000.

5. WCAP-16374-NP, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program," February 2005.

6. WCAP-16524-NP, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 2 Reactor Vessel Radiation Surveillance Program," February 2006.

7. WCAP-16449-NP, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 3 Reactor Vessel Radiation Surveillance Program," August 2005.

8. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1998.

9. Combustion Engineering, Reactor Vessel as-Built Drawings E-78173-161-001, R03 (Unit 1), E-79173-161-001, R02 (Unit 2) and E-65173-161-001, R01 (Unit 3).

10. RSICC Computer Code Collection CCC-650, DOORS 3.2, "One, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," April 1998.

11. RSIC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.

12. A. Schmittroth, FERRET Data Analysis Code, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

13. RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994.

14. ASTM E 1018-01, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)," American Society for Testing and Materials.

15. ASTM Standard E 944-02, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)," American Society for Testing and Materials.

16. ASTM E185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials.

17. TR-V-MCM-012, "Arizona Public Service Company, Palo Verde Unit 1, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program," Combustion Engineering Report dated January 31, 1987.

18. TR-V-MCM-013, "Arizona Public Service Company, Palo Verde Unit 2, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program," Combustion Engineering Report dated November 4, 1992.

19. TR-V-MCM-014, "Arizona Public Service Company, Palo Verde Unit 3, Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program," Combustion Engineering Report dated November 6, 1992.

20. Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C.

21. TR-V-MCM-002, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Palo Verde Unit 1 Reactor Vessel Materials," Combustion Engineering Report dated July 14, 1982.

22. TR-V-MCM-004, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Palo Verde Unit 2 Reactor Vessel Materials," Combustion Engineering Report dated June 30, 1983.

23. TR-V-MCM-010, "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Palo Verde Unit 3 Reactor Vessel Materials," Combustion Engineering Report dated November 5, 1992.

24. WCAP-15688, Rev 00, "CE-NSSS LTOP Energy Addition Transient Analysis Methodology," May 2001.

25. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection against Failure," 2001 Edition with 2003 Addenda.

26. ASME Boiler and Pressure Vessel Code Section III, Article NB-3211 and Article NB-2332, 2001 Edition with 2003 Addenda.

27. Letter, A. Meeden (APS) to J. Olszewski (Westinghouse), "APS Palo Verde Units 1, 2 and 3 PTLR Plant Data Request," Letter No. 448-00708 dated October 17, 2007.

WCAP-16835-NP, Revision 0

Westinghouse Non-Proprietary Class 3

Westinghouse

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

ENCLOSURE 1, ATTACHMENT 6

APS Responses to the NRC Request for Additional Information Related to the San Onofre Nuclear Generating Station (SONGS) PTLR Amendment Request

The precedent cited in Section 4.2 of the Evaluation of the Proposed TS Change was the PTLR amendment request for the San Onofre Nuclear Generating Station. (SONGS). Southern California Edison (SCE) submitted a license amendment request to the NRC by letter dated January 28, 2005 (ADAMS Accession No. ML050320286), and supplemented this request by letter dated January 12, 2006 (ADAMS Accession No. ML060190101), for SONGS Units 2 and 3 Operating License amendments to relocate the RCS P/T limits and LTOP limits from the TSs to a licensee-controlled PTLR. The NRC approved the SONGS Operating License amendments in a letter dated July 13, 2006 (ADAMS Accession No. ML062170006).

In their January 12, 2006, submittal, SCE identified, and provided responses to, nine NRC requests for additional information (RAIs) related to the PTLR amendment request. Provided below are APS responses to the nine RAIs.

NRC RAI No. 1 for SONGS

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

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

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

APS Response No. 1

Responses to the 26 action items of the staff's safety evaluation (SE) on topical report CE-NPSD-683, Revision 6, dated March 16, 2001, are provided in Section 3.3 of the Evaluation of the Proposed TS Change.

NRC RAI No. 2 for SONGS

*The ASME Code, Section XI, Appendix G provides a methodology for calculating stress intensity factors corresponding to membrane tension (K_{IM}) and thermal stress (K_{IT}) for the postulated axial defect. Calculations of K_{IT} are based on stress influence coefficients from finite element modeling (FEM) analyses for inside (1/4T) and outside (3/4T) surface flaws. Calculations of the maximum allowable K_{IM} are based on a closed-form solution to an equation such as $2K_{IM} + K_{IT} < K_{IC}$, where K_{IT} has been determined from solutions based on stress influence coefficients, and K_{IC} was determined using the equation representing the analytical approximation to the lower bound fracture toughness curve, K_{IC} (in ksi*sqrt(in.)) = 33.2 + 20.734exp[0.02(T -RT_{NDT})], where RT_{NDT} is the material nil-ductility transition reference temperature and T is the actual temperature of the material.*

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

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

a. *Discuss the methodology for calculating the thermal stress intensity factor, K_{IT}.*

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

Per your response to action item 21 in RAI 1, if your methodology applies the CE NSSS method for calculating K_{IM} stress intensity values, then your

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

APS Response No. 2

Westinghouse methodology for calculating the stress intensity factors corresponding to membrane tension resulting from pressure loading of a CE NSSS reactor vessel, $K_{IM,}$ is based on a two-dimensional finite element model with a unit internal pressure loading. This model considers a 1/4-thickness crack originating from the inside surface as well as a second model with a 1/4-thickness crack originating from the outside vessel surface. These models were loaded with internal pressure on the cylindrical inside surface. For the case with an inside crack, pressure was also applied to the crack face. Membrane stress intensity factor influence coefficients (K_{IM}) were then computed for unit internal pressure of 1000 psi. These K_{IM} coefficients are then corrected to represent three-dimensional surface cracks with an aspect ratio of 1-to-6 using the ASME Code ratios for 3D versus 2D flaws. Finally, these corrected values are used to calculate the actual stress intensity factors at any applied internal pressure.

In order to comply with the conditions listed in the safety evaluation for CE NSPD-683-A, a request for exemption from the requirements of 10CFR Part 50, Appendix G is provided as Enclosure 2 of this submittal.

NRC RAI No. 3 for SONGS

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

APS Response No. 3

PVNGS Units 1, 2, and 3 reactor vessel stress intensity factors (K_{IT}) and through-wall thermal gradients as a function of heatup or cooldown rate are discussed in Section 5.1 and presented in Figures 5-5 through 5-8 of WCAP-16835 (Attachment 5).

The membrane stress intensity factor coefficient (K_{IM}) due to unit (1000 psi) internal pressure loading for the PVNGS reactor vessel geometry with a base metal thickness of 11.2 inches is 25.0 (ksi√in) for a 1/4T crack depth location and 23.3 (ksi√in) at the 3/4T crack depth location.

NRC RAI No. 4 for SONGS

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

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

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

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

APS Response No. 4

The PTLR methodology of CE NSPD-683-A, Revision 06, first conducts a transient heat transfer analysis for all heatup and cooldown transients. Results from the heat transfer analyses, through-wall thermal gradient profiles, as well as the metal temperatures at the inside (1/4T) and outside (3/4T) crack tip locations are extracted. From the through-wall thermal gradients, the thermal stress intensity factors, K_{IT}, for all heatup and cooldown transients are then computed for all transients. Allowable fracture toughness is then computed using the crack tip metal temperatures. The membrane stress intensity factor, K_{IM} is then computed for a unit pressure loading case. Finally, limiting

allowable pressures are computed for all transients as well as steady-state conditions.

This approach is conservative when compared to the ASME Appendix G method as it includes the steady-state condition for all heatup and cooldown transients. Allowable pressures for some of these transients are limited by the steady-state condition at high pressures.

NRC RAI No. 5 for SONGS

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

a. *Those for pressure test conditions with the Reactor Coolant System (RCS) pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure.*

b. *Those for pressure test conditions with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure.*

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

d. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure at times the reactor is not in the critical operating mode.*

e. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure less than or equal to 20% of the reactor's preservice hydrostatic test pressure at times the reactor is in the critical operating mode.*

f. *Those for normal operating conditions (including heatups and cooldowns of the reactor and transient operating conditions) with the RCS pressure greater than 20% of the reactor's preservice hydrostatic test pressure at times the reactor is in the critical operating mode.*

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

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

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

APS Response No. 5

The pressure-temperature curves for pressure testing conditions and normal operations with the core critical and core not critical are in compliance with the appropriate minimum temperature requirements as given in Table 1 of Appendix G to 10 CFR Part 50. This is illustrated below along with each of the minimum temperature requirements of (a) through (f).

PVNGS Units 1, 2, and 3

Design pressure	= 2500 psia
Normal operating pressure	= 2250 psia
Preservice hydrostatic pressure	= 3125 psia
Minimum bolt-up temperature	= 80°F
Flange region RT_{NDT}	= 60°F
Initial piping, pumps and valves RT_{NDT}	= 40°F
Adjusted RT_{NDT} at 1/4T after 32 EFPY	= 116°F
Adjusted RT_{NDT} at 3/4T after 32 EFPY	= 103°F
20% Preservice hydrostatic pressure	= 625 psia

Preservice hydrostatic pressure with correction for instrument uncertainty
= 20% x preservice hydro pressure + RCS instrument uncertainty
= 625 psia – 111 psi = 514 psia for T_{RCS} < 200°F

= 625 psia – 128 psi = 497 psia for $T_{RCS} \geq 200°F$

Inservice hydrostatic pressure
= 1.1 x (Operating Pressure) + (pressurizer instrument uncertainty)
= 1.1 x (2250 psia) + 70 psi = 2545 psia

Minimum Temperature requirements for Inservice Hydrostatic/Leak Tests

a) Minimum temperature for pressures at or below 625 psia per 10 CFR 50 App G, Table 1, item 1.a
= highest flange RT_{NDT} + instrument uncertainty
= 60 + 13.2 = 73.2°F (80°F used)

b) Minimum temperature for pressures above 625 psia, per 10 CFR 50 App G, Table 1, item 1.b
= highest flange RT_{NDT} + 90°F + instrument uncertainty
= 60 + 90 + 13.2 = 163.2°F

Minimum Temperature requirements for Normal Operation (core not critical)

c) Minimum temperature for pressure at or below 625 psia, per 10 CFR 50 App G, Table 1, item 2.a
= highest flange RT_{NDT} + instrument uncertainty
= 60 + 13.2 = 73.2°F (80°F used)

d) Minimum temperature for pressure above 625 psia, per 10 CFR 50 App G, Table 1, item 2.b
= highest flange RT_{NDT} + 120°F + instrument uncertainty
= 60 + 120 + 13.2 = 193.2°F

Minimum Temperature requirements for Normal Operation (core critical)

e) Minimum temperature for pressure at or below 625 psia, per 10 CFR 50 App G, Table 1, item 2.c.
Larger of [(T_{Hydro}) or ($RT_{NDT\text{-}flange}$ + 40°F)] + instrument uncertainty
= [(168.2) or (60 + 40)] + 13.2 = 181.4°F

The minimum inservice hydrostatic test temperature ($T_{Hydro,\ uncorrected}$) is 168.2°F and corresponds to an uncorrected pressure of 2475 psi. When corrected for instrument uncertainty, these values correspond to a temperature of 181.4°F and a pressure of 2322.1 psia per Table 5-1 of WCAP-16835 (Attachment 5).

Lowest service temperature per ASME Section III, Division 1, Article NB-2332
= Initial piping, pumps and valves RT_{NDT} + 100°F + instrument uncertainty
= 40 + 100 + 13.2 = 153.2°F

Minimum temperature required for normal operation for pressures below 20% preservice hydro
= 181.4°F as given in Section 5.11 of WCAP-16835.

f) Minimum temperature for pressure above 625 psia, per 10 CFR 50 App G, Table 1, item 2.d:
= Larger of [(T_{Hydro}) or ($RT_{NDT\text{-}flange}$ + 160°F)] + instrument uncertainty
= [(168.2) or (60 + 160)] + 13.2 = 233.2°F

Minimum temperature required normal operation for pressures at or above 20% preservice hydro
= 233.2°F as given in Section 5.11 of WCAP-16835.

These minimum pressure and temperature requirements for hydrostatic test, and heatup and cooldown transients for PVNGS Units 1, 2, and 3 with the core not critical or critical, corrected for instrument uncertainty and RCS pressure and elevation effects are tabulated in Tables 5-1 and 5-2 of WCAP-16835 (Attachment 5).

NRC RAI No. 6 for SONGS

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

APS Response No. 6

The calculated reactor vessel pressure and temperature limit values shown in Tables 5-1 through 5-6 of WCAP-16835 (Attachment 5) are adjusted for instrument uncertainty and for RCS pressure and elevation effects. These instrumentation corrections ensure that the calculated beltline pressure-temperature limits are conservatively interpreted by pressurizer pressure and RCS temperature instrumentation. Section 5.13 of WCAP-16835 describes the application of instrumentation corrections at PVNGS.

The pressure correction factors applied to Tables 5-1 through 5-6 of WCAP-16835 of (Attachment 5) consist of three components:

1. A pressure differential corresponding to the static water head between the pressurizer water level and the reference point in the reactor vessel (ΔP_{ELEV});

2. The flow-induced pressure drop between the reactor vessel downcomer and the surge nozzle in the hot leg (ΔP_{FLOW}; a value that depends on the number of operating RCPs); and

3. The pressurizer pressure instrumentation loop uncertainty (ΔP_{INSTR}).

These components are individually established using conservative assumptions, then summed into the pressure correction factor. Pressure correction factors are subtracted from the analytical values to conservatively reduce the allowable pressure limit. The explicit pressure correction factor values applied depend on the number of operating RCPs and the pressure instrument in service. For Palo Verde, the pressure correction, including instrument uncertainty ranges from -111 psid to -169 psid as shown in Section 5.13 of WCAP-16835 (Attachment 5).

The heatup and cooldown data in Tables 5-1 and 5-2 of WCAP-16835 (Attachment 5) are also adjusted for temperature instrumentation uncertainty. For PVNGS, an uncertainty of +13.2 °F is added to all computed temperatures as shown in Section 5.13 of WCAP-16835.

NRC RAI No. 7 for SONGS

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

APS Response No. 7

The proposed pressure-temperature limit curves for PVNGS are confirmed to be effective through 32 EFPY. These curves are based on the RT_{NDT} shifts for the projected beltline fluence through 32 EFPY for the most restrictive of Units 1, 2, or 3.

NRC RAI No. 8 for SONGS

Criterion 7 of the Table in Attachment I to GL 96-03 specifies that an analysis of the credibility of the surveillance data must be provided in the PTLR. Regulatory Position 2.1 of Regulatory Guide (RG) 1.99, Revision 2 specifies that when two or more credible surveillance data sets become available from the reactor in question, they may be used to determine the Adjusted Reference Temperature (ART) values. If the procedure of Regulatory Position 2.1 for determining the ART values based on the surveillance data

results in a higher value for the ART than that given by using the procedures of Regulatory Position 1.1 of the RG, RG 1.99, Revision 2 specifies that the surveillance data should be used for the ART and chemistry factor determination. If the procedure of Regulatory Position 2.1 results in a lower value for the ART, either may be used.

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

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

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

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

a. *Table 7-1 of the SONGS 2 PTLR provides chemistry factors for the two surveillance materials plate C-6404-2 and weld 9-203. Please indicate how these chemistry factors were derived.*

b. *There is no explicit calculation in the SONGS 2 PTLR demonstrating that chemistry factor values for the limiting materials derived from the tables in RG 1.99, Revision 2 would result in limiting ART values that are more conservative than those determined using chemistry factors derived from surveillance data. Per your response to action item 24 in RAI 1 please supplement Section 7.0 of the PTLR for SONGS 2 with detailed calculations of the chemistry factors for each of the surveillance materials based on the calculation methods specified in Regulatory Position 2.1 of RG 1.99, Revision 2.*

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

APS Response No. 8

An analysis performed in accordance with Position 2.1 of RG 1.99 found that credibility criteria are met for surveillance plate and weld data from capsules withdrawn from

PVNGS. Tables 7-4 through 7-6 in WCAP-16835 (Attachment 5) demonstrate the credibility of surveillance measurements for PVNGS Units 1, 2, and 3, respectively. It is also confirmed that ART values for PVNGS reactor vessel beltline materials were calculated using Regulatory Position 1.1 of RG 1.99, Revision 2. This is conservative because analysis based on Regulatory Position 2.1 produce credible results that would yield lower values for predicted adjusted reference temperatures.

Chemistry factors for the limiting PVNGS surveillance plate and weld materials are shown in Tables 4-2, 4-3 and 4-4 of WCAP-16835 and were derived using Tables 1 and 2 of RG 1.99, Revision 2. These chemistry factors are compared to those derived using Regulatory Position 2.1 of RG 1.99, Revision 2 in the following table:

PVNGS	Material ID	Position 1.1 Chemistry Factor	Position 2.1 Chemistry Factor
Unit 1 Plate	M-6701-2	37°F	27.5°F
Unit 1 Weld	Heat 90071	34.1°F	4.9°F
Unit 2 Plate	F-773-1	20°F	17.5°F
Unit 2 Weld	Heat 3P7317	26.6°F	1.6°F
Unit 3 Plate	F-6411-2	26°F	10.2°F
Unit 3 Weld	Heat 4P7869	34.1°F	29.6°F

The preceding demonstrates that the chemistry factors derived based on the PVNGS surveillance plate and weld data are less than those derived using regulatory position 1.1 of RG 1.99, Revision 2. Therefore, the calculation of adjusted reference temperature using chemistry factors based on Regulatory Position 1.1 will result in more conservative values than would be obtained using plant-specific chemistry factors based on Regulatory Position 2.1. Predicted ART values at 1/4T and 3/4T for PVNGS are given in tables 4-5 through 4-7 of WCAP-16835 (Attachment 5).

<u>NRC RAI No. 9 for SONGS</u>

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

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

APS Response No. 9

Chemistry and fluence factors for surveillance weld materials and the measured ΔRT_{NDT} values, obtained per Position 2.1, for the PVNGS surveillance capsule weld materials are shown in Tables 7-1 through 7-3 of WCAP-16835 (Attachment 5). Adjusted values of ΔRT_{NDT} are not used in the determination of the chemistry factor for the surveillance weld because the copper and nickel content of the surveillance welds do not differ from that of the vessel welds. The bases for the chemical content of the vessel and surveillance plates and welds are provided in the APS response to Generic Letter 92-01.

ENCLOSURE 2

Application for Exemption from Certain 10 CFR Part 50, Appendix G Requirements when computing Pressure-Temperature Limits for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3

1.0 Introduction

2.0 Exemption Request

3.0 Discussion

- 3.1 Exemption is Authorized by Law
- 3.2 Granting this Exemption Will Not Present an Undue Risk to the Public Health and Safety
- 3.3 Granting this Exemption is Consistent with the Common Defense and Security
- 3.4 Special Circumstances Support the Issuance of an Exemption

4.0 Precedent

5.0 Conclusion

6.0 References

1.0 Introduction

Appendix G to 10 CFR 50 establishes fracture toughness requirements to be applied to ferritic reactor coolant pressure boundary materials of light water nuclear power reactors. The purpose of such requirements is to ensure adequate margins of safety exist during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code forms the basis for the requirements promulgated in Appendix G to 10 CFR Part 50. The rules of ASME Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components" are used when developing pressure and temperature limits for the beltline region of the PVNGS reactor vessels. The sections, editions and addenda of the ASME Boiler and Pressure Vessel Code, and any limitations and modifications thereof, which are approved by the staff for use in developing pressure and temperature limits, are specified in 10 CFR 50.55a.

The methodology developed by Combustion Engineering to calculate RCS pressure-temperature curves, heatup and cooldown limits and LTOP requirements is documented in topical report CE NPSD-683-A (Ref. 1). The staff noted in its March 16, 2001 safety evaluation for this report that "[t]he CE NSSS [nuclear steam supply system] methodology does not invoke the methods in the 1995 edition of Appendix G to the Code for calculating K_{IM} factors, and instead applies FEM [finite element modeling] methods for estimating the K_{IM} factors for the RPV shell ... the staff has determined that the K_{IM} calculation methods apply FEM modeling that is similar to that used for the determination of the K_{IT} factors [as codified in the ASME Code, Section XI, Appendix G]. The staff has also determined that there is only a slight non-conservative difference between the P/T limits generated from the 1989 edition of Appendix G to the Code and those generated from CE NSSS methodology as documented in Evaluation No. 063-PENG-ER-096, Revision 00. The staff considers that this difference is reasonable and that it will be consistent with the expected improvements in P/T generation methods that have been incorporated into the 1995 edition of Appendix G to the Code."

The staff has advised licensees to specify whether membrane stress intensity factors due to pressure loading, K_{IM}, are determined by obtaining a closed-form solution (per the ASME Code, Section XI, Appendix G) or determined by applying finite element modeling methods (per CE NPSD-683-A, Revision 6). Stress intensity values, K_{IM}, for PVNGS are computed using the CE NSSS finite element modeling methods, therefore, APS requests an exemption from the requirements of Appendix G to 10 CFR Part 50 to apply this model when calculating applicable Units 1, 2, and 3 pressure-temperature curves and LTOP limits.

2.0 Exemption Request

Reactor coolant system pressure-temperature curves and LTOP limits for PVNGS Units 1, 2, and 3 are based on the specific methodology developed by Combustion Engineering and approved by the NRC in CE NPSD-683-A. This methodology employs a finite element analysis model and a crack stress intensity factor, K_{IC}, of ASME Code Case N-640 (Ref. 2). Results produced by this method are slightly less conservative than the use of K_{IA} stress intensity factors and the linear elastic fracture mechanics methodology promulgated in Appendix G to 10 CFR Part 50. Specifically, Section IV.A.2 of Appendix G to 10 CFR Part 50 establishes the following criteria for generating plant-specific pressure-temperature limits:

- The pressure-temperature limits for an operating plant must be at least as conservative as those that would be generated if the methods of analysis and the margins of safety from Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code were applied.

Pursuant to 10 CFR 50.12, APS hereby applies for an exemption from the requirements of the above 10 CFR 50, Appendix G, criterion. This exemption is requested since the specific RCS pressure-temperature limits developed for PVNGS employ a finite element modeling methodology developed by Combustion Engineering and applied to CE NSSS plants for calculating K_{IM} stress intensity values.

APS addresses and satisfies the criteria of 10 CFR 50.12 in this exemption request. As required by 10 CFR 50.12(a)(1), and as more fully discussed below, this exemption is authorized by law, does not present an undue risk to the public health and safety, and is consistent with the common defense and security. Further, in accordance with 10 CFR 50.12(a)(2), the request demonstrates that special circumstances support issuance of the exemption.

3.0 Discussion

The reference pressure stress intensity, K_{IR}, used for calculation of RCS pressure and temperature limits at PVNGS is obtained from a reference fracture toughness curve for reactor vessel low alloy ferritic steels and is defined in Appendices A and G of Section XI of the ASME Code. This reference pressure stress intensity is determined by two properties, K_{IA} and K_{IC} that represent critical values of the stress intensity factor. For PVNGS, K_{IR} is defined as K_{IC}, with K_{IC} defined as the lower bound of static initiation critical K_I values measured as a function of temperature.

Title 10 CFR 50 Appendix G criteria require that pressure-temperature limit curves be generated from the most conservative combinations of the limiting P/T data points and the minimum temperature requirements listed in Appendix G to 10 CFR Part 50. The NRC Staff endorsed Appendix G to the ASME Code through the 1995 Edition at the time that the PTLR analysis methodology described in CE NSPD-683-A was approved.

ASME Code Case N-640 permits application of the lower bound static crack initiation critical stress intensity factor equation (i.e., K_{IC} equation) as the basis for establishing the P/T curves in lieu of using the lower bound crack arrest critical stress intensity factor equation (i.e., K_{IA}) which is based on conditions needed to arrest a propagating crack, and which is the method invoked by Appendix G to Section XI of the ASME Code. Use of the K_{IC} equation to determine the lower bound fracture toughness when computing P/T curves is more technically correct than the use of the K_{IA} equation since the rate of loading during a heatup or cooldown is slow, and since crack initiation, which is more representative of a static condition than a dynamic condition, is principally at issue. The K_{IC} equation appropriately implements the use of the static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel.

Appendix G to 10 CFR 50 required the use of the conservative K_{IA} equation since 1974, when the equation was codified. Use of the conservative K_{IA} equation was considered necessary due to a limited knowledge of reactor pressure vessel material properties at the time. A significant amount of additional materials property data have been collected about RPV fabrication materials since 1974 and have provided the staff with a better understanding of how the RPV materials behave in service. For this reason, the staff has concluded that this additional information is sufficient to permit a lower bound static crack initiation critical stress intensity factor (K_{IC} equation) coupled with a finite element analysis methodology to be used when calculating P/T limits, as described in Section 1.4.2 of the NRC Safety Evaluation related to Topical Report CE NPSD-683-A, Revision 6 (Ref. 1). In addition, P/T curves based on the K_{IC} equation will enhance overall plant safety by opening the P/T operating window with the greatest safety benefit in the region of low temperature operations. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

Exemptions from the requirements of 10 CFR 50 Appendix G may be granted by the Commission in accordance with 10 CFR 50.12. For the reasons discussed below, the exemption criteria in Section 50.12 are satisfied by this application.

3.1 Exemption is Authorized by Law

Title 10 CFR 50.12(a)(1) requires a demonstration that an exemption from NRC regulations is authorized by law. This demonstration is found in 10 CFR 50.60 which defines acceptance criteria for fracture prevention measures for normal operation of light water nuclear power reactors.

Paragraph (a) of 10 CFR 50.60 requires that PVNGS Units 1, 2, and 3 meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in 10 CFR 50 Appendices G and H. Paragraph (b) of 10 CFR 50.60 advises that proposed alternatives to the described requirements in 10 CFR 50 Appendices G and H may be used when an exemption is granted by the

Commission pursuant to 10 CFR 50.12. Accordingly, this exemption request is authorized by law, as required by Section 50.12(a)(1).

3.2 Granting this Exemption Will Not Present an Undue Risk to the Public Health and Safety

Title 10 CFR 50.12(a)(1) requires a demonstration that the granting of an exemption from the requirement in question will not present an undue risk to the public health and safety. As demonstrated below, this exemption request fully satisfies this criterion.

Requirements to monitor and control the pressure and temperature imposed on the PVNGS Units 1, 2, and 3 reactor coolant system pressure boundaries during heatup, cooldown, testing and normal operation remain unchanged as a result of this exemption request. Further, any conceivable risk would be equivalent to that inherent in any other license application where an exemption request to apply the K_{IC} crack stress intensity factors was permitted by the staff.

Any risk to the public health and safety created by granting this exemption request will be mitigated by a number of factors. First, existing licensee programs and activities which serve to ensure safe plant operation (e.g., operational, maintenance, engineering, and corrective action programs and processes) will remain in effect during operation of PVNGS Units 1, 2, and 3. Second, the full array of NRC inspection and oversight activities will remain in effect, including the agency's authority to shut down any or all units at PVNGS. Third, these inspection and oversight activities will be further and fully informed by NRC Staff review of this PVNGS license amendment request, which will have been completed by the time the exemption request is granted. Fourth, no changes are made to the methods used to develop the PTLR nor to the application of such P/T results to operation of the Palo Verde Nuclear Generating Station, therefore granting the requested exemption will not present an undue risk to the health and safety of the public.

3.3 Granting this Exemption is Consistent with the Common Defense and Security

NRC requirements relating to maintaining the integrity of the reactor coolant system pressure boundary are fully met by this exemption request. The exemption requested in no way affects the security or safeguards features or programs at PVNGS. Such features and programs will remain in full effect during the term of each unit's operating license. Accordingly, granting the requested exemption is consistent with the common defense and security.

3.4 Special Circumstances Support the Issuance of an Exemption

Title 10 CFR 50.12(a)(2) requires a showing of at least one of six "special circumstances" to support issuance of the requested exemption. One of the special circumstances identified in Section 50.12(a)(2) applies to this request, that is, the application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of the regulations in 10 CFR Part 50, Appendix G, is to provide an acceptable margin of safety against brittle failure of the RCS during any condition of normal operation to which the pressure boundary may be subjected over its service lifetime. Special circumstances, pursuant to 10 CFR 50.12(a)(2)(ii), apply to this exemption request in that continued operation of PVNGS Units 1, 2, and 3 with P/T limit curves developed in accordance with the ASME Code, Section XI, Appendix G, without the authorization to utilize the alternative K_{IM} calculational methodology of CE NPSD-683-A, Revision 6, is not necessary to achieve the underlying purpose of 10 CFR Part 50, Appendix G. Application of the calculational methodology documented in CE NPSD-683-A, Revision 6, in lieu of the calculational methodology specified in the ASME Code, Section XI, Appendix G, provides an acceptable alternative evaluation procedure that will continue to meet the underlying purpose of 10 CFR Part 50, Appendix G.

Therefore, APS requests an exemption based on the special circumstances of 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."

4.0 Precedent

The analysis methodology of CE NPSD-683-A employs an alternate finite element analysis method for calculating stress intensity factors for the reactor pressure vessel shell. Upon review of an application by Southern California Edison (SCE) for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 (Ref. 3), the NRC determined that sufficient information was presented to assess the method for calculating K_{IM} factors. Except for loading inputs, the staff found that the K_{IM} calculation methods, as applied to the P/T limits developed for SONGS-2 and -3, utilize finite element analysis modeling in a manner that is similar to that endorsed by Appendix G of the ASME Code. The staff also determined that only a slight non-conservative difference existed between the P/T limits generated from the 1989 edition of Appendix G to the ASME Code as compared to those generated using the methodology of CE NPSD-683-A. The staff found this difference to be reasonable and consistent with the expected improvements in P/T generation methods that have been incorporated into later editions of Appendix G to the ASME Code. Therefore, the staff concluded that the methodology of CE NPSD-683-A for generating P/T limits is equivalent to the current methodology in the 1995 edition of Appendix G to the Code and is acceptable for P/T limit applications.

Consistent with the conditions imposed by the staff's safety evaluation concerning topical report CE NPSD-683-A, Revision 06, SCE's January 12, 2006, submittal (Ref. 4) included a request for an exemption from certain requirements of 10 CFR Part 50, Appendix G when calculating P/T limits. The specific exemption requested involved the application of CE Nuclear Steam Supply System finite element analysis methodology for calculating K_{IM} stress intensity values due to internal pressure loading rather than the linear elastic fracture mechanics methods promulgated in Appendix G to 10 CFR 50 and described in Appendix G to the ASME Code Section XI. The NRC staff authorized the Southern California Edison exemption request in a letter dated June 5, 2006 (Ref. 5).

5.0 Conclusion

Title 10 CFR 50.60(b) permits licensees to use alternatives to the requirements of Appendix G to Part 50 if an exemption is granted by the Commission pursuant to the provisions and exemption acceptance criteria of 10 CFR 50.12. The staff has previously granted permission to Southern California Edison through the exemption request process to apply CE finite element methods and ASME Code Case N-640 to the calculation of plant-specific P/T limits (Refs. 5 and 6).

Analytical procedures employed by Westinghouse to develop the PVNGS reactor vessel P/T limits use the finite element analysis methods developed by Combustion Engineering and the guidance found in Appendix G of ASME Section XI. Use of K_{IC} crack stress criteria to calculate the allowable fracture toughness when establishing P/T limits for PVNGS is consistent with Section XI of the 2001 Edition of the ASME Code. The justification presented in this application provides sufficient grounds for issuance of the requested exemption to APS.

As required by Section 50.12 of the NRC regulations, the exemption sought is authorized by law, presents no undue risk to public health and safety, is consistent with the common defense and security, and is supported by special circumstances. A thorough safety and technical review of the PVNGS PTLR and proposed changes to the Technical Specifications by the staff are sufficient to provide reasonable assurance of continued safe operation of PVNGS Units 1, 2, and 3 without increased risk to the health and safety of the public and without any potential environmental impact. Accordingly, APS respectfully requests that the NRC grant the exemption from the requirements of Section IV.A.2 of Appendix G to 10 CFR Part 50 as applied to the development of RCS pressure-temperature limits for PVNGS Units 1, 2, and 3.

6.0 References

1. Combustion Engineering Owners Group Topical Report CE NPSD-683-A, Revision 6, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001 (ADAMS Accession No. ML011350387).

2. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1."

3. Letter from Southern California Edison Company (SCE) to the NRC, "San Onofre Nuclear Generating Station Units 2 and 3, Docket Nos. 50-361 and 50-362, Proposed Change Number NPF-10/15-551, License Amendment Request, 'Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)'," January 28, 2005 (ADAMS Accession No. ML050320286).

4. Letter from Southern California Edison Company (SCE) to the NRC, "San Onofre Nuclear Generating Station, Units 2 and 3, Docket Nos. 50-361 and 50-362, Proposed Change Number NPF-10/15-551, License Amendment Request, 'Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)'," January 12, 2006 (ADAMS Accession No. ML060190101).

5. Letter, N. Kalyanam (NRC) to R. M. Rosenblum (SCE), "San Onofre Nuclear Generating Station, Units 2 and 3 - Exemption from the Requirements of Appendix G to 10 CFR Part 50, (TAC NOS. MC5773 AND MC5774)" dated June 5, 2006 (ADAMS Accession No. ML061730433).

6. Letter from the NRC to Southern California Edison Company (SCE), "San Onofre Nuclear Generating Station, Units 2 And 3 - Issuance of Amendments Re: Reactor Coolant System (RCS) Pressure And Temperature Limits Report (PTLR) (TAC NOS. MC5773 AND MC5774)," July 13, 2006 (ADAMS Accession No. ML062170006).