Enclosure 6

Vogtle Electric Generating Plant Units 1 and 2
Non-proprietary version of WCAP-16142-NP, Rev. 1

Westinghouse Non-Proprietary Class 3

WCAP-16142-NP
Revision 1

February 2004

Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

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Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2

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February 2004

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1 INTRODUCTION

10 CFR Part 50, Appendix G contains requirements for pressure-temperature limits for the primary system, and requirements for the metal temperature of the closure head flange and vessel flange regions. The pressure-temperature limits are to be determined using the methodology of ASME Section XI, Appendix G [1], but the flange temperature requirements are specified in 10CFR50 Appendix G. This rule states that the metal temperature at the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure, which is 621 psig for a typical PWR, and 300 psig for a typical BWR.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, outside surface stresses in this region typically reach over 70 percent of the steady state stress, without being at steady state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{Ia} fracture toughness, in the mid 1970s, to ensure that appropriate margins would be maintained.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1." ASME Code Case N-640 was approved for use without conditions by the NRC in Regulatory Guide 1.147 [16].

Figure 1-1 illustrates the problem created by the flange requirements for a typical PWR heatup curve. It is easy to see that the heatup curve using K_{Ic} provides for a much higher allowable pressure through the entire range of temperatures. For this plant, however, the benefit is negated at temperatures below RT_{NDT} +120°F because of the flange requirement of 10 CFR Part 50, Appendix G. The flange requirement of 10 CFR 50 was originally developed using the K_{Ia} fracture toughness, and this report will show that use of the newly accepted K_{Ic} fracture toughness for flange considerations leads to the conclusion that the flange requirement can be eliminated for Vogtle Units 1 and 2.

Revision 1. Created to correct errata in the report. No technical changes were made.

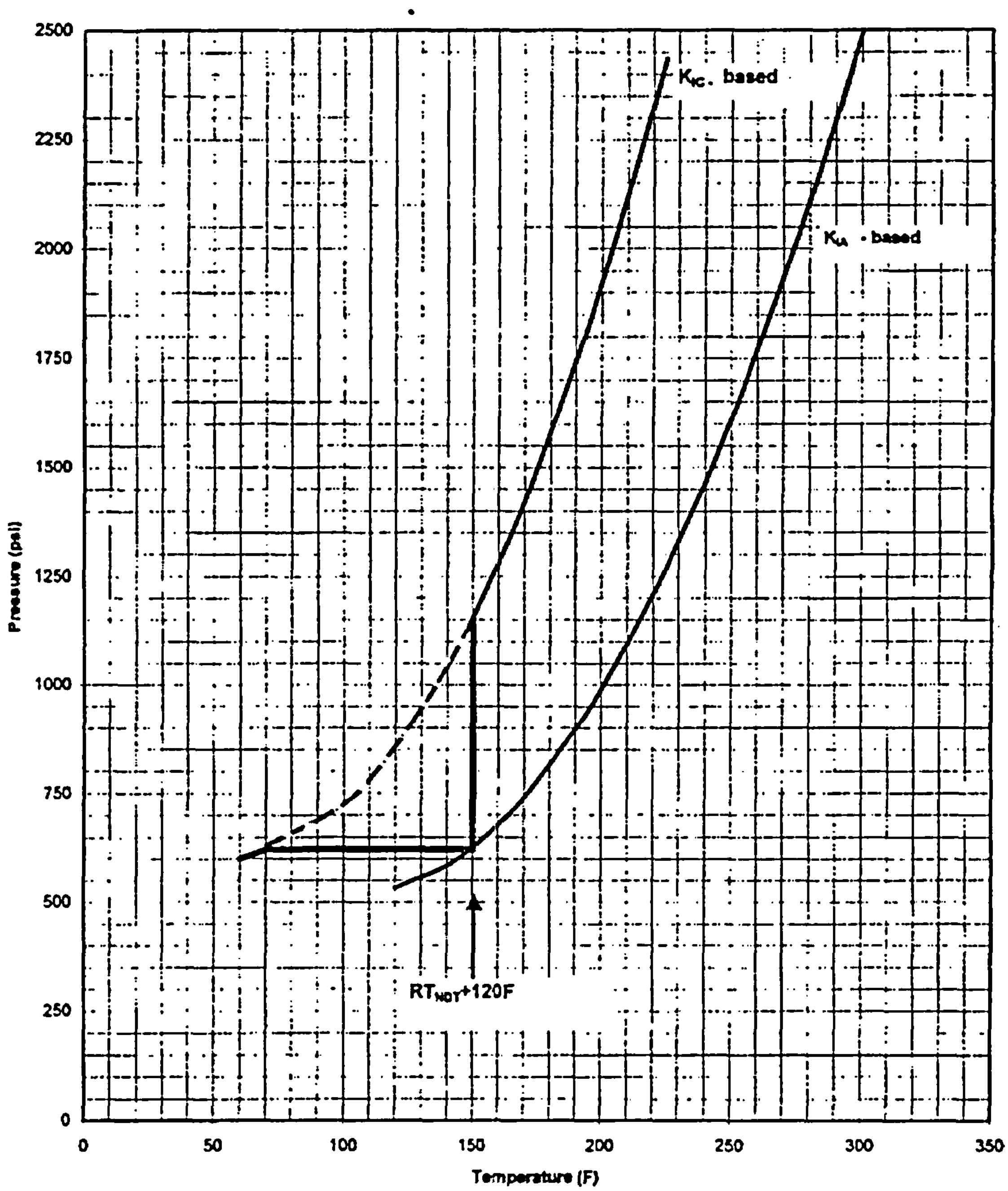


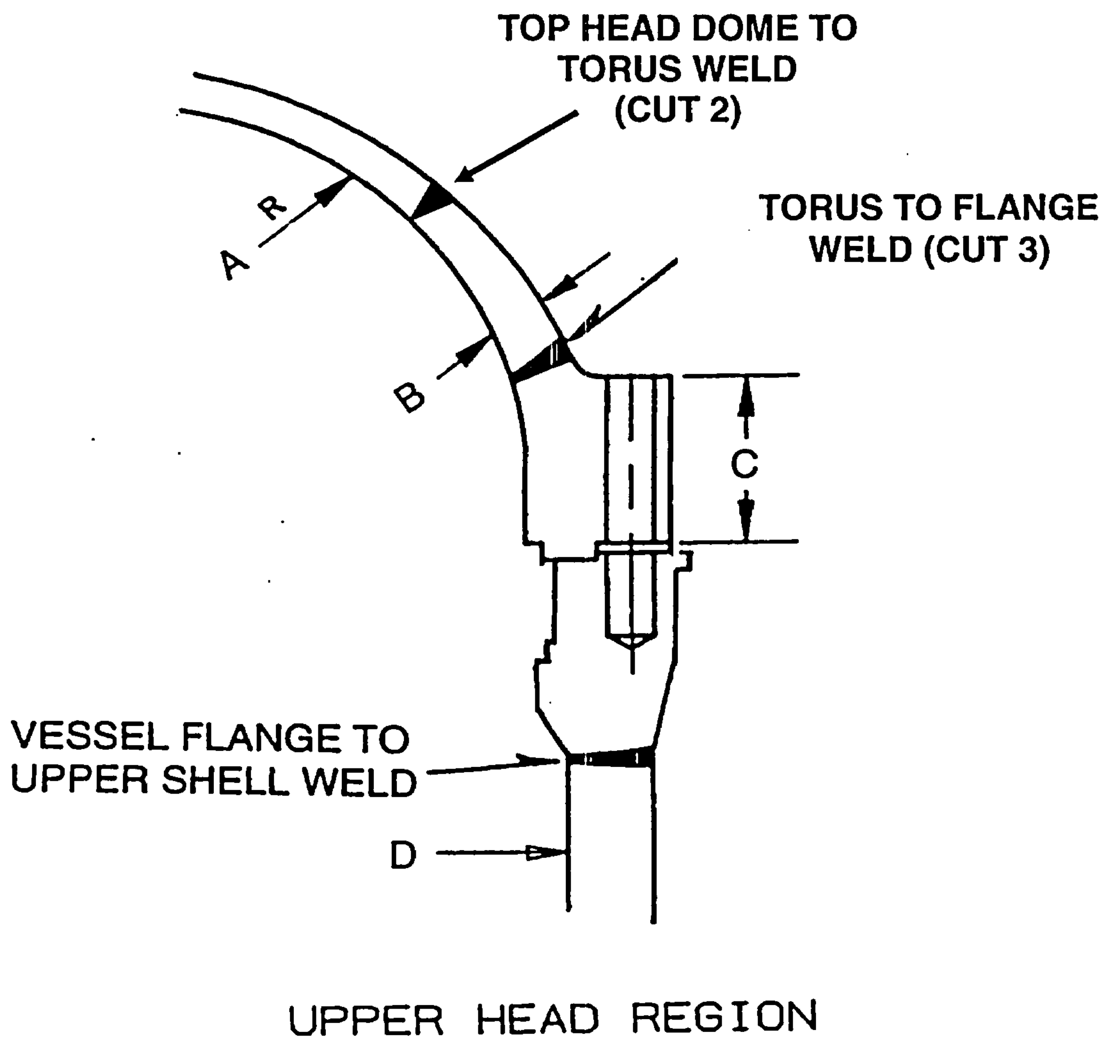
Figure 1-1 Illustration of the Impact of the Flange Requirement for a Typical PWR Plant

2 TECHNICAL APPROACH

The evaluation presented here is intended to cover the Vogtle Units 1 and 2 reactor vessels. Fracture evaluations have been performed on the closure head geometry specific to these units, and results will be tabulated and discussed. The geometry of the closure head region for Vogtle Units 1 and 2 is shown in Figure 2-1.

Stress analyses have been performed, and these stress results were used to perform fracture mechanics evaluations. Details of the finite element stress analysis results are provided in Appendix C. The highest stress location in the closure head and vessel flange region is in the head, just above the bolting flange. This corresponds with the location of two welds as shown in Figure 2-1. The highest stressed location is near the outside surface of the head in that region, and so the fracture evaluations have assumed a flaw at the outside surface.

The goal of the evaluation is to compare the structural integrity of the closure head during the boltup, plant heatup and plant cooldown processes, to the structural integrity during steady state operation. The question to be addressed is: With the higher K_{Ic} fracture toughness now known to be applicable, is there still a concern about the structural integrity of the closure head during boltup?



	Vogtle Units 1 and 2
A	86.0
B	7.00
C	27.25
D	170.88

NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 2-1 Geometry of the Upper Head/Flange Region of the Vogtle Units 1 and 2 Reactor Vessels

3 FRACTURE ANALYSIS METHODS AND MATERIAL PROPERTIES

The fracture evaluation was carried out using the approach suggested by Section XI Appendix G [1]. A semi-elliptic surface flaw was postulated to exist in the highest stressed region, which is at the outside surface of the closure flange. The flaw depth was assumed to encompass a range of depths into the wall thickness, and the shape was set at a length six times the depth.

3.1 STRESS INTENSITY FACTOR CALCULATIONS

One of the key elements of a fracture evaluation is the determination of the driving force or stress intensity factor (K_I). In most cases, the stress intensity factor for the structural integrity calculations utilized a representation of the actual stress profile rather than a linearization. The stress profile was represented by a cubic polynomial:

$$\sigma(x) = A_0 + A_1 x + A_2 x^2 + A_3 x^3 \qquad (3\text{-}1)$$

where:

x	=	the coordinate distance into the wall, in.
σ	=	stress perpendicular to the plane of the crack, ksi
A_i	=	coefficients of the cubic fit

For the surface flaw with length six times its depth, the stress intensity factor expression of Raju and Newman (Ref. 2) was used. The stress intensity factor K_I can be calculated anywhere along the crack front. The point of maximum crack depth is represented by $\phi = 0$, and this location was found to also be the point of maximum K_I for the cases considered here. The following expression is used for calculating K_I as a function of the angular location around the crack front (ϕ). The units of K_I are $\text{ksi}\sqrt{\text{in}}$.

$$K_I = \left[\frac{\pi a}{Q}\right]^{0.5} \sum_{j=0}^{3} G_j\left(a/c,\ a/t,\ t/R,\ \phi\right) A_j\, a^j \qquad (3\text{-}2)$$

The boundary correction factors G_0, G_1, G_2, and G_3 are obtained by the procedure outlined in reference (2). The dimension "a" is the crack depth, "c" is the crack half length, "t" is the wall thickness, "R" is the inside radius, and "Q" is the flaw shape factor, which can be approximated by $Q = 1 + 1.464\,(a/c)^{1.65}$.

3.2 FRACTURE TOUGHNESS

Another key element in a fracture evaluation is the fracture toughness of the material. The fracture toughness has been taken directly from the reference curves of Appendix A, Section XI [1]. In the transition temperature region, these curves can be represented by the following equations:

$$K_{Ic} = 33.2 + 20.734 \exp[0.02\,(T - RT_{NDT})] \qquad (3\text{-}3)$$

$$K_{Ia} = 26.8 + 12.445 \exp[0.0145\,(T - RT_{NDT})] \qquad (3\text{-}4)$$

where K_{Ic} and K_{Ia} are in $\text{ksi}\sqrt{\text{in}}$.

The upper shelf temperature regime requires utilization of a shelf toughness which is not specified in the ASME Code. A value of 200 $ksi\sqrt{in}$ has been used here. This value is consistent with general practice in such evaluations, as shown for example in reference 3, which provided the background and technical basis for the development of Appendix A of Section XI.

The final key element in the determination of the fracture toughness is the value of RT_{NDT}, which is a material parameter determined from Charpy V-notch and drop-weight tests.

The value of RT_{NDT} for the closure flange region of the Vogtle units was obtained from the reactor vessel record reports [13, 14] and the certified material test reports or determined from Charpy tests and drop-weight tests [15]. The results are shown in Table 3-1. The highest value was 20°F and so this value was used for the illustrations to be discussed in Sections 4 and 5.

3.3 IRRADIATION EFFECTS

Neutron irradiation has been shown to produce embrittlement which reduces the toughness properties of reactor vessel steels. The decrease in the toughness properties can be assessed by determining the shift to higher temperatures of the reference nil-ductility transition temperature, RT_{NDT}.

The location of the closure flange region is such that the irradiation levels are very low and therefore the fracture toughness is not measurably affected.

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4 FLANGE INTEGRITY

The first step in evaluation of the closure head/flange region is to examine the stresses. The stresses which are affected by the boltup event are the axial, or meridional stresses, which are perpendicular to the nominal plane of the closure head to flange weld. The stresses in this region during the entire heatup and cooldown process are summarized in Appendix C.

The boltup is the key condition to review here, in comparison with the heatup and cooldown operation, since the flange requirement applies to boltup conditions. No other transients result in stresses in this region at low temperatures. One might suggest that the cooldown might be of similar concern, but the boltup is governing for a number of reasons:

1. The heatup and cooldown transient is structured to ensure generous margins are maintained (SF = 2) for a large flaw in the irradiated beltline region, not for the unirradiated flange region.

2. The cooldown transient has much higher temperatures in the head region than the boltup, and

3. The thermal stresses caused by the cooldown transient tend to counteract the boltup stresses; cooldown thermal stresses are tensile on the inside surface and compressive on the outside surface.

Table 4-1 provides a comparison of the stresses at boltup with those at the governing time step of heatup and cooldown which is end of heatup. It is easy to see that the stresses at boltup are mostly bending, with a very small membrane stress. As the vessel is pressurized, the membrane stresses increase. These results were taken from a finite element analysis of the heatup/cooldown process, and the boltup stress alone was compared with the most limiting time step of the entire heatup/cooldown transient, which includes pressure, thermal, and boltup stresses.

The relative impact of these stresses can best be addressed through a fracture evaluation. A semi-elliptic surface flaw was postulated at the outer surface of the closure head flange, and the stress intensity factor, K_I, (or crack driving force) was calculated. The results are shown for cut 3 weld region in Figure 4-1, and for the cut 2 weld region in Figure 4-2. For a semi-elliptic surface flaw with depth equal to 10 percent of the wall thickness postulated in the highest stress region of the head, the following values were determined for the stress intensity factor.

Boltup:	$K_I = 24.84\ ksi\sqrt{in}$ (for $a/t = 0.1$)
End of Heatup:	$K_I = 49.21\ ksi\sqrt{in}$ (for $a/t = 0.1$)

It will be useful to highlight the difference in the integrity for the head region using the two values of fracture toughness. The boltup temperature for a typical PWR is 60°F, so if RT_{NDT} = 20°F the ASME reference toughness values are $K_{Ia} = 49.0\ ksi\sqrt{in}$ and $K_{Ic} = 79.3\ ksi\sqrt{in}$. Using the K_{Ia} toughness (which was the basis for the original flange requirements) it can be seen that the toughness exceeds the applied stress intensity factor for boltup for flaws of any depth in the head thickness. From Figure 4-1, the smallest margin = $1 - K_I/K_{Ia} = 0.39$, when $a/t = 0.36$. For the heatup and cooldown transient, the coolant temperature at the governing time steps, near the end of heatup, is 557°F. The fracture toughness is therefore 200 $ksi\sqrt{in}$, so again the margin is very large.

Using the K_{Ic} toughness, which has now been adopted by Section XI [1] for P-T Curves, it can be seen that there is also a significant margin between the fracture toughness and the applied stress intensity factor, for both the boltup and the heatup cooldown transient. Another objective of the requirements in Appendix G is to assure that fracture margins are maintained to protect against service induced cracking due to environmental effects. Since the governing flaw is on the outside surface (the inside is in compression) where there are no environmental effects, there is even greater assurance of fracture margin. Therefore, it may be concluded that the integrity of the closure head/flange region is not a concern for the Vogtle units using the K_{Ic} toughness. There are two possible mechanisms of degradation for this region, thermal aging and fatigue.

Effect of Fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region.

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Table 4-1 Stress Distributions at the Closure Flange Region – Vogtle Units 1 and 2		
Distance (x/t)	**Boltup Stress at Cut 3 (ksi)**	**Heatup* (344 minutes) at Cut 2 (at p=2317 psig, t=557°F)**
0 (ID)	-14.38	-15.32
0.1	-10.77	
0.2	-7.83	-3.42
0.3	-5.14	
0.4	-2.66	4.55
0.5	-0.26	
0.6	2.16	12.15
0.7	4.72	
0.8	7.54	21.76
0.9	11.24	
1.0 (OD)	19.70	38.77

* With boltup stress superimposed.

Notes: Cut 3 has the highest boltup stress.
Cut 2 has the highest transient stress.

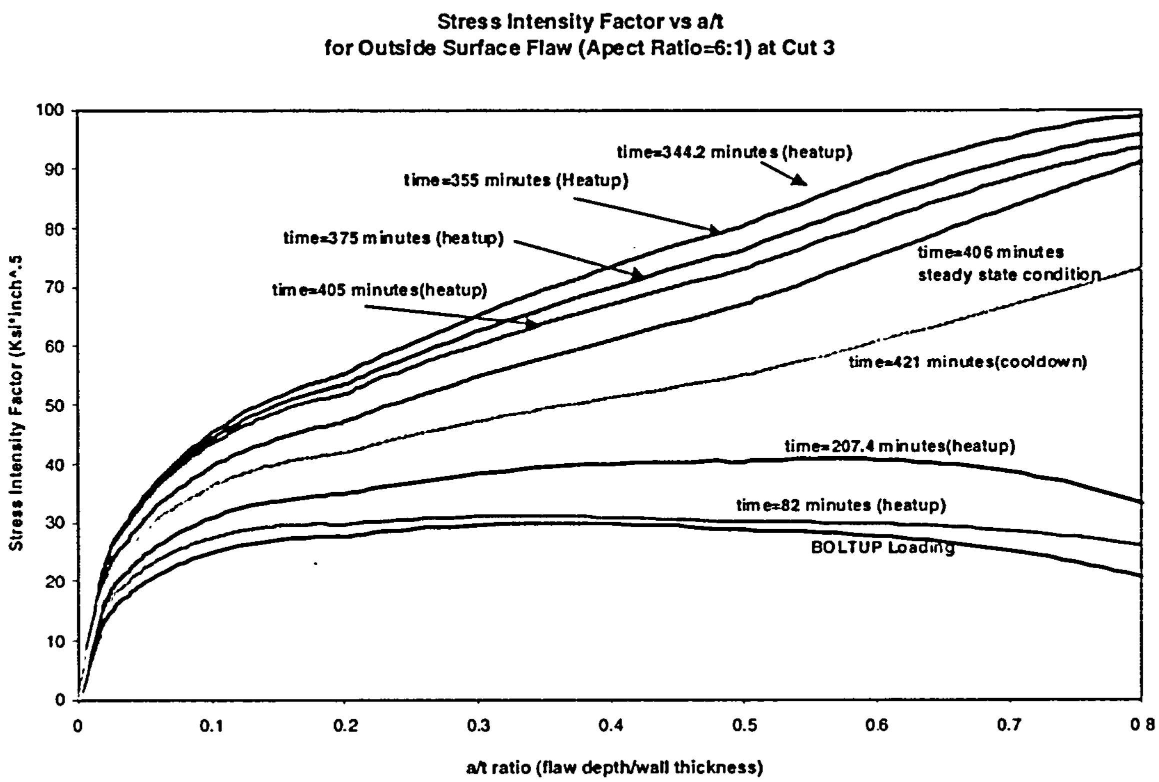

Figure 4-1 **Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Torus to Flange Region Weld for Vogtle Units 1 and 2 (stress intensity factor units are $ksi\sqrt{in}$)**

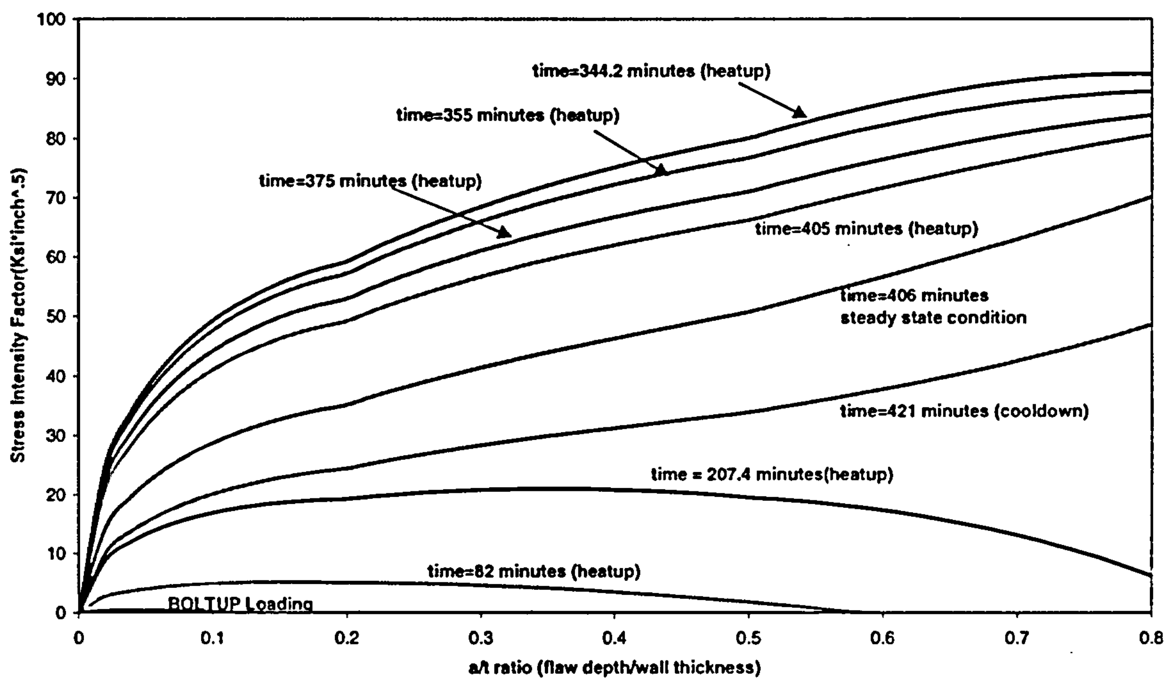

Figure 4-2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Dome to Torus Region Weld for Vogtle Units 1 and 2 (stress intensity factor units are $ksi\sqrt{in}$)

5 ARE FLANGE REQUIREMENTS NECESSARY?

Using the K_{IC} curve can support the elimination of the flange temperature requirement. This can be illustrated by examining the stress intensity factor change for a postulated flaw as the vessel is heated and pressurized after boltup, progressing up to steady state operation.

The stresses at the region of interest are shown in Table 4-1, for the end of heatup, as well as boltup. Included here are the stress distributions through the wall, showing that the highest stress location for this region is the outer surface.

As the vessel is pressurized, the stresses in the closure flange region gradually change from mostly bending stresses to a combination of bending and membrane stresses. The stress intensity factor, or driving force, increases for a postulated flaw at the outside surface, as the vessel is pressurized.

A direct comparison between the original basis for the boltup requirement and the new K_{IC} approach is provided in Table 5-1. This table provides calculated boltup requirements for all the designs, using a safety factor of 2, and a reference flaw depth of $a/t = 0.10$, which was used by Randall as the basis for the original requirement [11]. Before discussing the table, it will be helpful to discuss the basis for the reference flaw, in light of current technology, and using the results of the Performance Demonstration Initiative.

Basis for the Reference Flaw Size. Regulatory Guide 1.150 stimulated improvement in examinations of the clad to base-metal interface. The same techniques have been used for more than 10 years for reactor vessel head examinations performed from the outside surface. Capability demonstrations for the clad to base-metal interface have been conducted at the EPRI NDE Center since 1983. These demonstrations were performed initially for the belt-line region. However, similar techniques are used for both the vessel belt-line and the reactor vessel head, although the head exams are done manually.

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The flange temperature requirement. Finally, a simple illustration can be used to demonstrate clearly that no flange requirement is needed at the Vogtle units. The lower bound K_{Ic} fracture toughness from the ASME Code is 33.2 ksi-sq-rt-in., which means that the toughness cannot be lower than that value. Study of Figures 4-1 and 4-2, for a postulated flaw of ten percent of the thickness, shows that the stress intensity factor does not exceed this value until a time step of about 300 minutes, or 5 hours into the heatup. By this time, the flange temperature is greater than 200°F, so there can be no possibility of fracture. All other locations within the flange have lower stresses, so this statement applies to the entire flange. This clearly shows that the flange requirement can be eliminated for Vogtle Units 1 and 2.

Table 5-1 Comparison of Various Plant Designs Boltup Requirements

Plant	K_I (ksi$\sqrt{in}$) (a/t = .1)	K_I (ksi$\sqrt{in}$) (with a/t = 0.1, SF = 2)	$T - RT_{NDT}$ (°F) using K_{Ic} (a/t = .10)	$T - RT_{NDT}$ (°F) using K_{Ia} (a/t = .10)
CE	30.0	60.0	13	68
B&W	39.4	79.8	41	100
Vogtle Units	24.9	49.8	0*	43
W 3 Loop	28.7	57.5	8	63
GE (CBI 251")	38.7	77.4	38	97
GE (B&W 251")	48.0	96.0	56	118
GE (CE 218")	25.1	50.2	0*	43

* The calculated value of $T\text{-}RT_{NDT}$ is negative, so zero is used for conservatism.

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Figure 5-1 **Probability of Correct Rejection/Reporting (PCR) Considering Passed plus Failed Candidates, Appendix VIII Supplement 4, Detection from the Outside Surface. Reporting Criterion A′ = 0.15 inch, TWE Represents Flaw Depth.**

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Figure 5-2 Probability of Correct Rejection/Reporting (PCR) Considering Only Passed Candidates, Appendix VIII Supplement 4, Detection from the Outside Surface. Reporting Criterion A' = 0.15 inch, TWE Represents Flaw Depth.

6 SAFETY IMPLICATIONS OF THE FLANGE REQUIREMENT

There are important safety implications which are associated with the flange requirement, as illustrated by Figure 6-1. The safety concern is the narrow operating window at low temperatures forced by the flange requirement. The flange requirement sets a pressure limit of 621 psi for a PWR (20 percent of hydrotest pressure). Thus, no matter how good the toughness of the vessel, the P-T limit curve may be superceded by the flange requirement for temperatures below RT_{NDT} + 120°F. This requirement was originally imposed to ensure the integrity of the flange region during boltup, but Section 4 has shown that this is no longer a concern.

The flange requirement can cause severe operational limitations when instrument uncertainties are added to the lower temperature range limit (621 psi), for the Low Temperature Overpressure Protection system of PWRs. The minimum pressure required to cool the seals of the main coolant pumps is 325 psi, so the operating window sometimes becomes very small, as shown schematically in Figure 6-1. If the operator allows the pressure to drop below the pump seal limit, the seals could fail, causing the equivalent of a small break LOCA, a significant safety problem. Elimination of the flange requirement will significantly widen the operating window for most PWRs.

An example will be provided to illustrate this situation for an operating PWR plant, Byron Unit 1. This is a forging-limited vessel at 12 EFPY, with a low leakage core, and low copper weld material in the core region. The vessel has excellent fracture toughness, which means that the flange notch is very prominent, as shown in the vessel heatup curve of Figure 6-2. As illustrated before in Figure 6-1, Byron has the LTOP setpoints significantly below the flange requirement of 621 psi, because of a relatively large instrument uncertainty. The setpoints of the two power operated relief valves are staggered by about 16 psi to prevent a simultaneous activation. The two PORVs have different instrument uncertainties, and for conservatism the higher uncertainty is used. A similar situation exists for cooldown, as shown in Figure 6-3.

Elimination of the flange requirement for the case of Byron Unit 1 would mean that the PORV curve could become level at 604/587 psig, which are the leading/trailing setpoints to protect the PORV downstream piping, through the temperature range of the 350°F down to boltup at 60°F. The operating window between the leading PORV and the pump seal limit rises from 121 psig (446-325) to 262 psig (587-325). This change will make a significant improvement in plant safety by reducing the probability of a small LOCA, and easing the burden on the operators.

This is only one example of the impact of the flange requirement. Every operating PWR plant will have a different situation, but the operational safety level will certainly be generally improved by the elimination of this unnecessary requirement. The flange impact for Vogtle Unit 1, for example, is shown in Figures 6-4 and 6-5 [12].

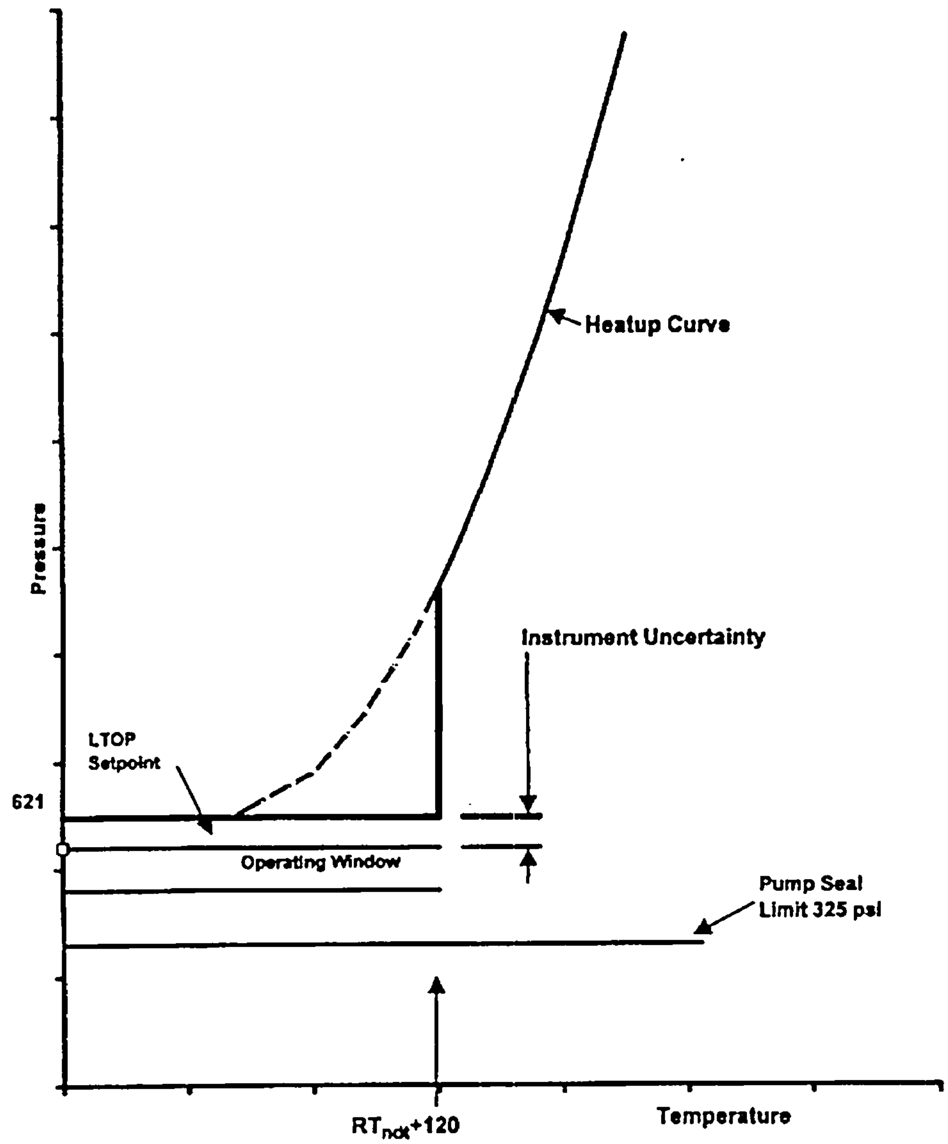

Figure 6-1 Illustration of the Flange Requirement and its Effect on the Operating Window for a Typical Heatup Curve

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 5P-5933 (using surv. capsule data)
LIMITING ART VALUES AT 12 EFPY: 1/4T, 70°F
3/4T, 60°F

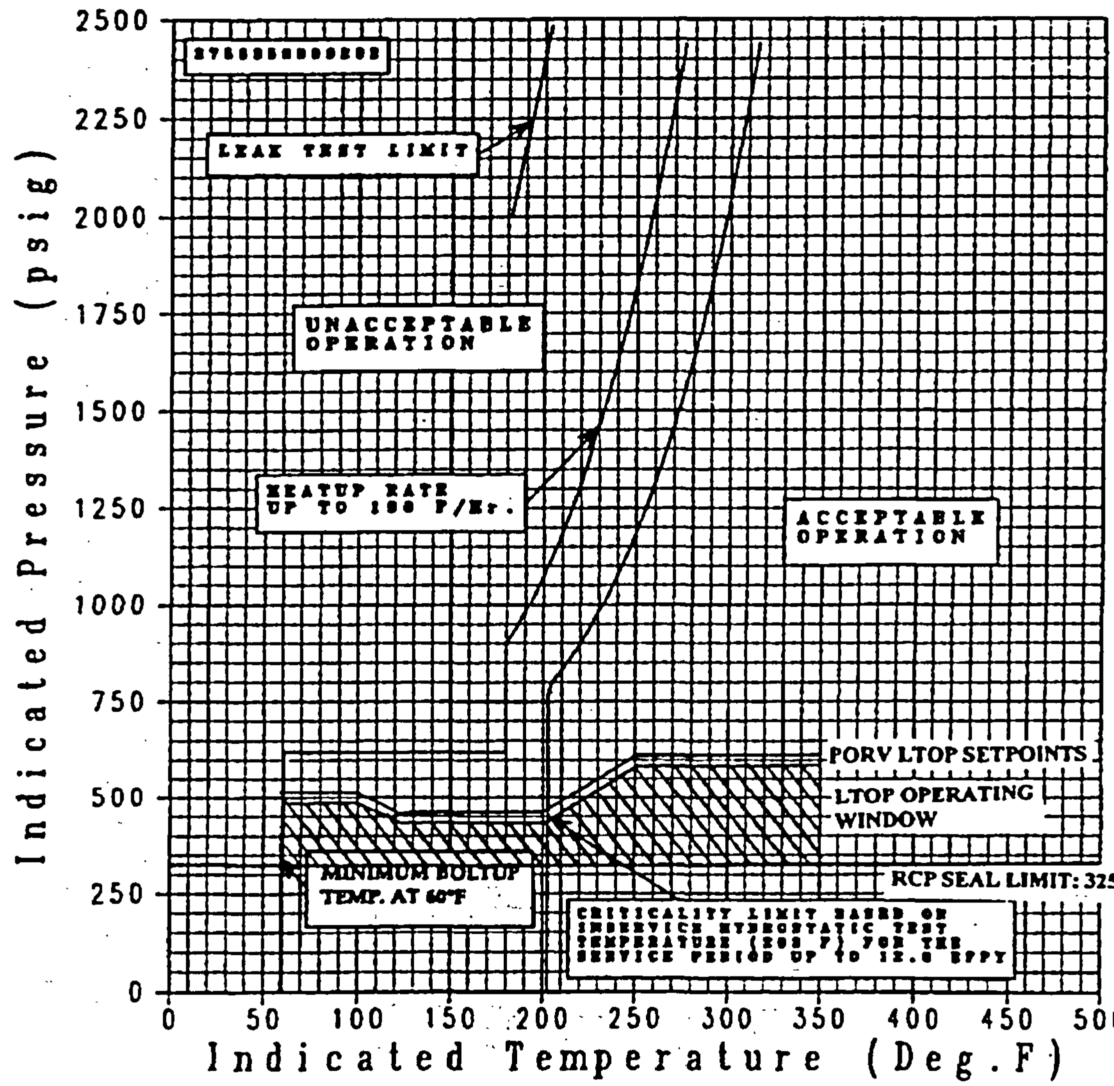

Figure 6-2 Illustration of the Actual Operating Window for Heatup of Byron Unit 1, a Low Copper Plant at 12 EFPY

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 5P-5933 (using surv. capsule data)
LIMITING ART VALUES AT 12 EFPY: 1/4T, 70°F
3/4T, 60°F

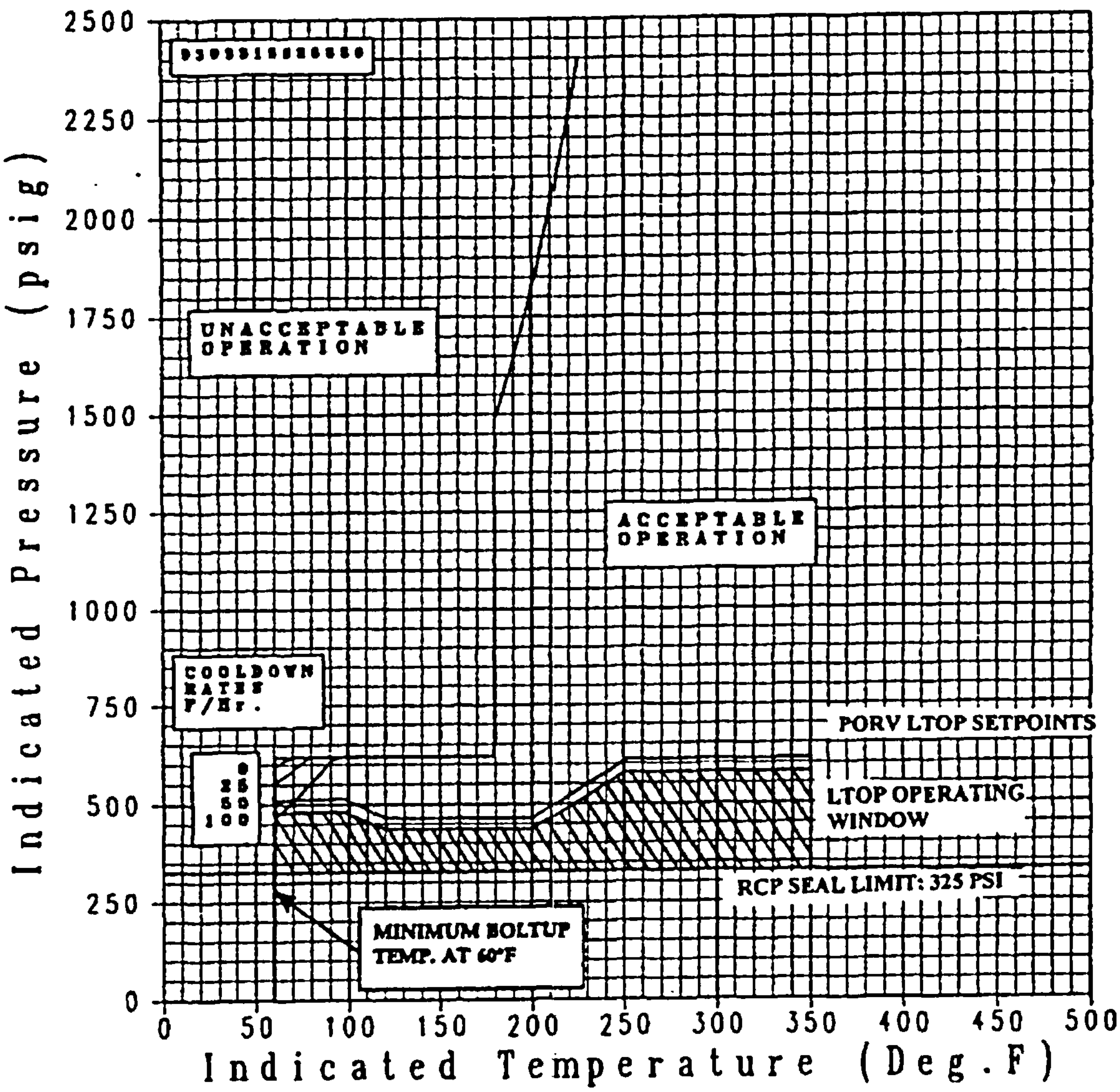

Figure 6-3 Illustration of the Actual Operating Window for Cooldown of Byron Unit 1, a Low Copper Plant at 12 EFPY

VOGTLE ELECTRIC GENERATING PLANT (VEGP) - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

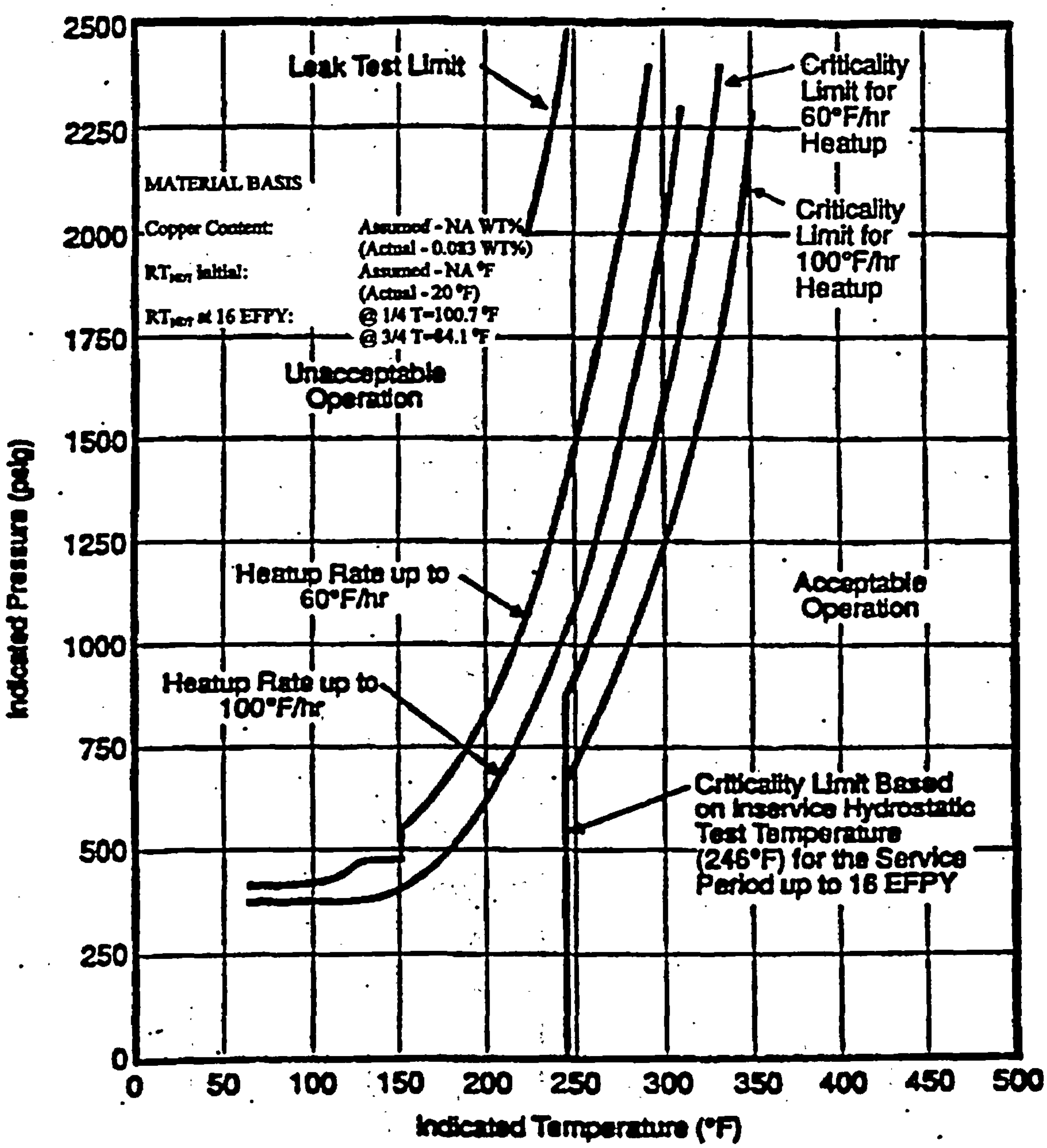

Figure 6-4 Illustration of the Flange Notch for Vogtle Unit 1, Heatup Curve

VOGTLE ELECTRIC GENERATING PLANT (VEGP) - UNIT 1
PRESSURE AND TEMPERATURE LIMITS REPORT

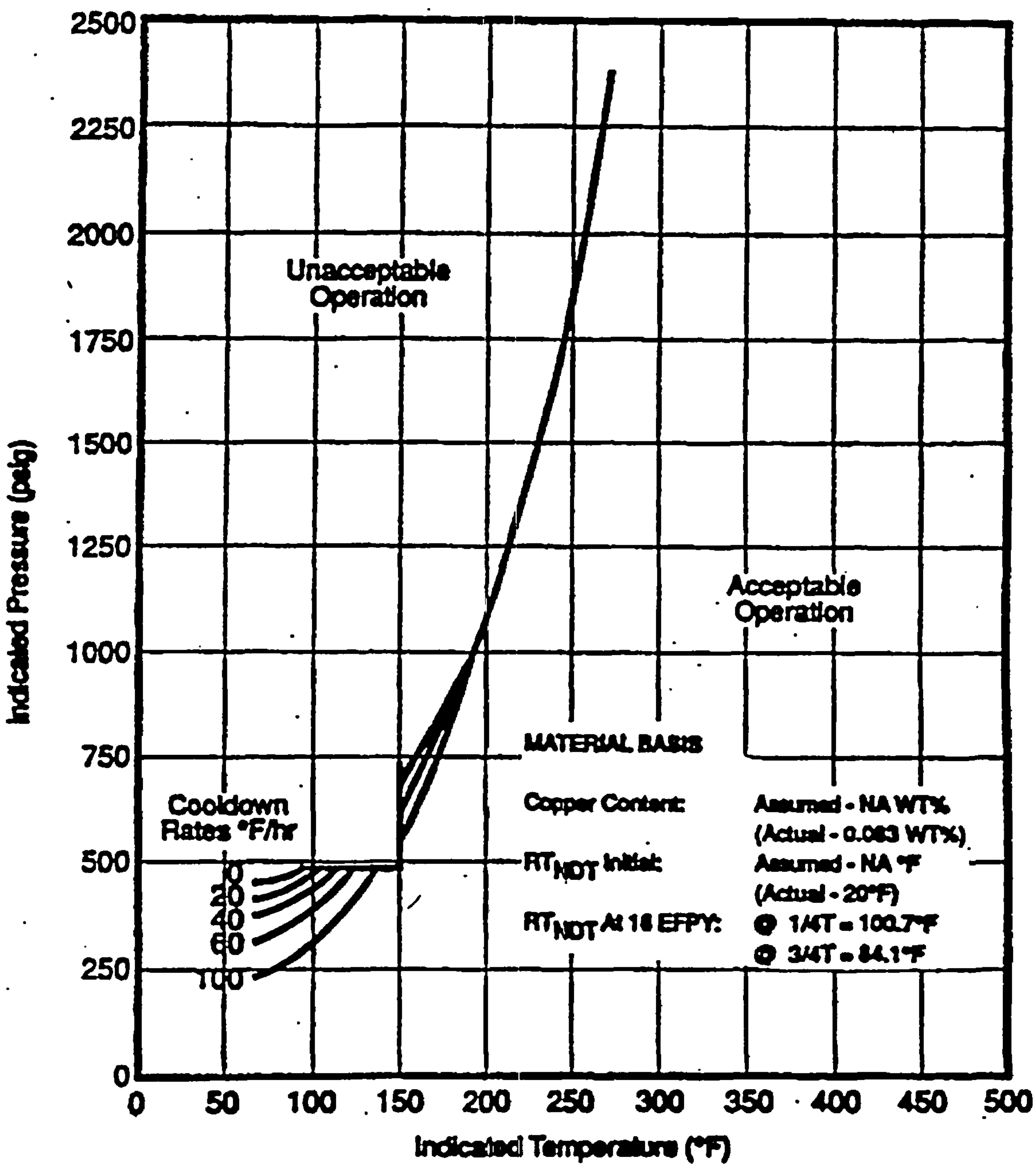

Figure 6-5 Illustration of the Flange Notch for Vogtle Unit 1, Cooldown Curves

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APPENDIX A
REACTOR PRESSURE VESSEL INSPECTION RELIABILITY*

F. L. Becker

EPRI

Charlotte NC

1 ABSTRACT

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*Presented at the Joint EC-IAEA Technical Meeting on Improvements in Inservice Inspection Effectiveness, Pettan, The Netherlands, November 2002, to be published.

3 DETECTION

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3.1 OUTSIDE SURFACE DEMONSTRATION

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Figure 1 Probability of Detection Performance for Passed and Passed Plus Failed Candidates for Appendix VIII Supplement 4, from the Outside Surface as a function of the flaw through wall extent (TWE). Both automated and manual techniques are included.

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Figure 2 POD for Inside Surface Examinations, Pass and Pass + Failed Candidates, Passed and Pass Plus Failed Candidates are included.

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3.2 COMBINED ID AND OD DETECTION

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Figure 3 Probability of Detection for Automated RPV Examinations Considering Both Inside and Outside Access. Passed and Passed Plus Failed Candidates are shown.

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Figure 4 POD for Pass and Failed Candidates, Considering ID and OD Automated Demonstrations and Manual OD Demonstrations.

4 SIZING

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Figure 5 Histogram of Depth Successful Sizing Candidate Test Scores, Appendix VIII, Supplement 4. Examinations Were Performed Both From the Inside and Outside Surfaces.

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Figure 6 Sizing Error Surface Model

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Figure 7 Plan View of Sizing Error Surface Model

5 ACCEPTABILITY EVALUATION

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Figure 8 Probability of Correct Sizing for Passed Candidates, Appendix VIII Supplement 4. Reporting Threshold A′ = 0.15 inch.

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Figure 9 Probability of Correct Rejection/Reporting (PCR) for automated techniques, Considering Passed and Passed plus Failed Candidates, includes both inside and outside surface information. Reporting Criterion A' = 0.15 inch.

6 SUMMARY

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7 REFERENCES

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APPENDIX B
THERMAL AGING OF FERRITIC RPV STEELS AT REACTOR OPERATING TEMPERATURES

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APPENDIX C
STRESS DISTRIBUTIONS IN THE CLOSURE HEAD REGION

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APPENDIX D
FLANGE INSPECTION RESULTS:
VOGTLE UNITS

D-1 Introduction

The information below provides information about the Vogtle RPV head flange region inservice inspections (when the inspections were conducted, the extent of coverage achieved, inspection methods used, etc.). This information supports the discussion in the body of this report regarding the quality of inspections cited to support the assumed reference flaw size (page 5-1). More specifically, this evaluation demonstrates how the inspections conducted at Vogtle, Units 1 and 2, support the assumption of a 0.1T flaw size in the flange evaluation.

D-2 Inspection History

The Vogtle Unit 1 and 2 Closure Head to Flange welds (nominal 8.25-inch thick) were examined during the second ten-year interval to the 1989 Edition of ASME Section XI. The vessel flange examination history is shown in Table D-1. Examinations have also been performed during PSI and the first ten-year interval to earlier editions of ASME Section XI. Magnetic particle and ultrasonic examination of the examination volume (Figure IWB-2500-5) was performed as follows:

Magnetic Particle and Ultrasonic Examinations

Unit 1:

- 100% of the weld length was examined during the 1R7 outage (Fall 1997).
- This outage was in the first ISI period of the second ten-year ISI interval.
- Both the magnetic particle and the ultrasonic examinations had no recordable indications.

Unit 2:

- 100% of the weld length was examined during the 2R7 outage (Fall 1999).
- This outage was in the first ISI period of the second ten-year ISI interval.
- Both the magnetic particle and the ultrasonic examinations had no recordable indications.

SNC submitted a relief request (RR-4) to the NRC for limited ultrasonic examination coverage for both Vogtle 1 and 2. This relief request was submitted as part of the second ten-year ISI Program. This relief request was approved by a safety evaluation report (SER) to SNC in an NRC letter dated December 31, 1998.

D-3 Magnetic Particle Examination Techniques

In order to detect flaws open to the outer diameter (OD) surface, a magnetic particle examination of the weld and adjacent areas as required in Figure IWB-2500-5 was performed prior to each ultrasonic examination using SNC procedure MT-V-505. This examination procedure was written in accordance with Article 7 of the ASME Boiler and Pressure Vessel Code Section V. The acceptance standards and extent of coverage was in accordance with Section XI of the ASME Boiler and Vessel Code examination category B-A. One hundred percent of the required examination surface per figure IWB-2500-5 was achieved and no indications recorded.

D-4 Ultrasonic Examination Techniques

In addition to the magnetic particle examination, an ultrasonic examination was performed in accordance with SNC procedure UT-V-411. This procedure was written to comply with ASME Section XI, Appendix I, Article 4 of ASME Section V, 1989 Edition and NRC Regulatory Guide 1.150. The ultrasonic acceptance standards were in accordance with Section XI requirements.

The ultrasonic examination system was calibrated on basic calibration block 402A. In order to record and evaluate flaws throughout the volume of the weld, a distance amplitude correction curve (DAC) was generated from side drilled holes and a 2% ID notch (approx. 0.105-inch deep, excluding clad) into the base material, which is used for far surface resolution.

Scanning of the component is performed at a minimum of 6 dB over the reference gain at which the DAC is established. The increase in scanning sensitivity further increases the probability of detection. The 1989 Edition of ASME Section XI, Appendix I requires that reflectors that produce a response greater than 20% DAC be investigated. In addition, the examiner is required to determine whether the indication originates from a flaw or is a geometric indication.

During these examinations at both Vogtle units, the ultrasonic examination had limitations due to the flange configuration and lifting lug obstructions. The combined coverage was calculated to be approximately 68%.

D-5 Summary

The magnetic particle surface examination will detect flaws open to the surface.

The ultrasonic examinations will detect flaws throughout the volume. The ultrasonic examinations are conducted with scanning sensitivities (at least 2X or +6 dB) over calibration.

The recording requirements for ultrasonic examination are extremely low (20% DAC).

The ultrasonic examinations were calibrated on a 2% notch from calibration block 402A into the ferritic base material. It is expected that an ultrasonic response from the 2% notch would be extremely sensitive when compared to a 0.1T (10%), flaw, which is of concern.

A high percentage of coverage (approximately 90%) from the head side was obtained with the ultrasonic examination.

A high percentage of coverage (100%) was obtained with the magnetic particle examination.

There were no indications recorded with either the magnetic or ultrasonic examinations for either the Unit 1 or Unit 2 head to flange weld.

D-6 Conclusion

The probability of detection for flaws on the high stress region of the outer surface of the closure head is very high due to the magnetic particle examination being performed with no limitations.

Based on the ability of the ultrasonic system to detect ID surface reflectors (2% ID notch), there is very high probability that flaws 0.1 T (10%) will be detected for the accessible volumes. Although the ID surface was not fully examined in those areas where the RPV lifting lugs are located, a significant amount of the outer 25% of the examination area was scanned in all four directions by ultrasonic examination and therefore, significant through-wall indications would have been recorded.

Table D-1 Reactor Vessel Flange Examination History

Component	Description	Examination	Sensitivity	Coverage	Results	Comments/Schedule
Unit 1 Reactor Vessel Head	Flange to head weld (11201-V6-001-W02)	Surface examination (magnetic particle) and Ultrasonic exams using 0, 45, 60 and 70 degree scans. Examinations have been performed to NRC RG 1.150.	The surface technique is capable of detecting indications with a major dimension of 1/16th of an inch. The sensitivity of the ultrasonic exams is based on signal responses from calibration block side-drilled holes and ID notch.	Achieved 100% coverage for the surface examination. The volumetric examination was limited to approximately 65% due to flange configuration and 3 integrally mounted lifting lugs, (see NRC approved Relief Request RR-4).	No recordable indications for either the surface or ultrasonic examinations.	This examination was performed twice during the Unit's commercial operation, last performed in October 1997 (1R7). The next examination will be performed in the third interval (2008).
	Flange	Visual examination	Technique requires lighting and distance sufficient to detect scratches or pits 0.001 to 0.003 of an inch.	Entire flange area on head and vessel is examined, however, the defect size criteria are specifically applied to the o-ring seating surface areas.	There have been no recordable indications that have required repair.	This exam is performed each refueling outage.
Unit 1 Reactor Vessel	Flange ligament (11201-V6-001-L01 through L54)	Ultrasonic examination of the threads in the flange and a 1-inch annular volume around the flange stud holes.	The sensitivity of the ultrasonic exam is based on signal responses from calibration block side-drilled holes.	Achieved 100% coverage for the Volumetric examination.	No recordable indications.	Ligament exams divided into stud regions (54) and spread across the ISI periods. The last exams performed March 2002 (1R10). The remaining third scheduled for March 2005 (1R12).
	Flange to shell weld (11201-V6-001-W03)	Mechanized Ultrasonic ID exams (tool) using 0, 45, 60 and 70 degree scans and manual exams from the flange face with a "sled" apparatus. Examinations have been performed to NRC RG 1.150.	The sensitivity of the ultrasonic exam is based on signal responses from calibration block side-drilled holes and ID notch.	Achieved 100% coverage for the Volumetric examination using combination of Shell side and flange face examinations. Expect to achieve essentially 100% with Appendix VIII techniques.	Code acceptable indications recorded.	This examination was performed once during the Unit's commercial operation in March 1996 (1R6). The next examination is scheduled for September 2006 (1R13). VEGP plans to submit relief to use Appendix VIII qualified techniques in lieu of Section V and RG 1.150.

Table D-1 (cont.) Reactor Vessel Flange Examination History						
Component	**Description**	**Examination**	**Sensitivity**	**Coverage**	**Results**	**Comments/Schedule**
Unit 2 Reactor Vessel Head	Flange to head weld (21201-V6-001-W02)	Surface examination (magnetic particle) and Ultrasonic exams using 0, 45, 60 and 70 degree scans. Examinations have been performed to NRC RG 1.150.	The surface technique is capable of detecting indications with a major dimension of 1/16th of an inch. The sensitivity of the ultrasonic exams is based on signal responses from calibration block side-drilled holes and ID notch.	Achieved 100% coverage for the surface examination. The volumetric examination was limited to approximately 65% due to flange configuration and 3 integrally mounted lifting lugs, (see NRC approved Relief Request RR-4).	No recordable indications for either the surface or ultrasonic examinations.	This examination was performed twice during the Unit's commercial operation, last performed in September 1999 (2R7). The next examination will be performed in the third interval (2008).
	Flange	Visual examination	Technique requires lighting and distance sufficient to detect scratches or pits 0.001 to 0.003 of an inch.	Entire flange area on head and vessel is examined, however, the defect size criteria are specifically applied to the o-ring seating surface areas.	There have been no recordable indications that have required repair.	This exam is performed each refueling outage.
Unit 2 Reactor Vessel	Flange ligament (21201-V6-001-L01 through L54)	Ultrasonic examination of the threads in the flange and a 1-inch annular volume around the flange stud holes.	The sensitivity of the ultrasonic exam is based on signal responses from calibration block side-drilled holes.	Achieved 100% coverage for the Volumetric examination.	No recordable indications.	Ligament exams divided into stud regions (54) and spread across the ISI periods. The last exams performed October 2002 (2R9). The remaining third scheduled for September 2005 (2R11).

Table D-1 Reactor Vessel Flange Examination History (cont.)

Component	Description	Examination	Sensitivity	Coverage	Results	Comments/Schedule
	Flange to shell weld (21201-V6-001-W03)	Mechanized Ultrasonic ID exams (tool) using 0, 45, 60 and 70 degree scans and manual exams from the flange face with a "sled" apparatus. Examinations have been performed to NRC RG 1.150.	The sensitivity of the ultrasonic exam is based on signal responses from calibration block side-drilled holes and ID notch.	Achieved 100% coverage for the Volumetric examination using combination of Shell side and flange face examinations. Expect to achieve essentially 100% with Appendix VIII techniques.	Code acceptable indications recorded.	This examination was performed once during the Unit's commercial operation in March 1998 (2R6). The next examination is scheduled for March 2007 (2R12). VEGP plans to submit relief to use Appendix VIII qualified techniques in lieu of Section V and RG 1.150.

Enclosure 7

Vogtle Electric Generating Plant Units 1 and 2
TS, Bases, and PTLR Amendment

Enclosure 7

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and
Pressure and Temperature Limits Report

TS, Bases, and PTLR Amendment

Proposed Changes

The proposed changes to the Technical Specifications (TS) are as follows:

- Section 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), will be revised to reference to the NRC-approved methodology for developing P/T limits and COPS setpoints (WCAP-14040-A, Rev. 4) and the methodology used to justify eliminating the reactor vessel closure head/vessel flange requirements (WCAP-16142-P, Revision 1).

- Section 3.4.12, Cold Overpressure Protection Systems (COPS), will be revised to change the RCS vent size in LCO 3.4.12b from 2.14 square inches to 1.5 square inches.

The proposed changes to the VEGP Unit 1 and Unit 2 PTLRs are consistent with the revised P/T limits and COPS setpoints, and the information used to develop the revised P/T limits and COPS setpoints. Tables 1 and 2 below provide a cross-reference of the current PTLR Figure and Table number to the revised PTLR Figure and Table number, where applicable.

Table 1- Changes to VEGP Unit 1 PTLR

Current Vogtle Unit 1 PTLR Figure Number	Revised Vogtle Unit 1 PTLR Figure Number
2.1-1	2-1
2.1-2	2-2
2.2-1	3-1
Current Vogtle Unit 1 PTLR Table Number	**Revised Vogtle Unit 1 PTLR Table Number**
2.1-1	2-1
2.1-2	2-2
2.2-1	3-1
3.0-1	5-1
3.0-2	5-2
3.0-3	5-3
3.0-4	5-4
3.0-5	5-7
3.0-6	5-5 and 5-6
3.0-7	5-4
3.0-8	Deleted
3.0-9	5-8
3.0-10	5-9
5.0-1	Deleted

Enclosure 7

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and
Pressure and Temperature Limits Report

TS, Bases, and PTLR Amendment

Table 2- Changes to VEGP Unit 2 PTLR

Current Vogtle Unit 2 PTLR Figure Number	Revised Vogtle Unit 2 PTLR Figure Number
2.1-1	2-1
2.1-2	2-2
2.2-1	3-1
Current Vogtle Unit 2 PTLR Table Number	**Revised Vogtle Unit 2 PTLR Table Number**
2.1-1	2-1
2.1-2	2-2
2.2-1	3-1
3.0-1	5-1
3.0-2	5-2
3.0-3	5-3
3.0-4	5-4
3.0-5	5-7
3.0-6	5-5 and 5-6
3.0-7	5-4
3.0-8	5-8
3.0-9	5-9

Basis for Proposed Changes

One of the proposed changes to the TS involves revising Section 3.4.12, Cold Overpressure Protection Systems (COPS), to change the RCS vent size in LCO 3.4.12 b. The RCS vent size was calculated based on the revised Appendix G limits developed utilizing ASME Code Case N-640 and without the reactor vessel flange temperature requirement. The revised RCS vent size is capable of mitigating the limiting cold overpressure transient. The other proposed changes to the TS involve revising the references in Section 5.6.6 to reflect the NRC-approved methodology for developing P/T limits and COPS setpoints (WCAP-14040-A, Rev. 4) and the methodology used to justify eliminating the reactor vessel closure head/vessel flange requirements (WCAP-16142-P, Revision 1). The PTLR currently contains the pressure and temperature (P/T) limits, including heatup and cooldown rates, and the nominal power operated relief valve (PORV) setpoints for cold overpressure protection. The concept of the PTLR is that the limits contained in it, which are also plant-specific and vary with vessel fluence, can be revised without prior NRC approval provided that they are calculated using an NRC-approved methodology. The revised VEGP Unit 1 and 2 PTLRs are consistent with NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The surveillance capsule withdrawal schedule is contained in UFSAR Table 5.3.1-8 for Unit 1 and UFSAR Table 5.3.1-9 for Unit 2. The methodology for determining the limits specified in the PTLR is discussed in Enclosure 1.

Enclosure 8

Vogtle Electric Generating Plant Units 1 and 2
Significant Hazards Consideration Evaluation

Enclosure 8

Vogtle Electric Generating Plant
Request to Revise Technical Specifications and Pressure and Temperature Limits Report

Significant Hazard Consideration Evaluation

Proposed Changes

Southern Nuclear Operating Company (SNC) proposes to revise the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications (TS). The proposed changes would revise Section 3.4.12 to change the RCS vent size in LCO 3.4.12 b and would revise Section 5.6.6 to incorporate references to WCAP-14040-A, Rev. 4 and WCAP-16142-P, Revision 1. The revised VEGP Unit 1 and 2 PTLRs are consistent with NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." The following is a detailed description of the proposed changes.

Section 3.4.12, Cold Overpressure Protection Systems (COPS), will be revised to change the RCS vent size in LCO 3.4.12 b from 2.14 square inches to 1.5 square inches.

Section 5.6.6, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), will be revised to reference to the NRC-approved methodology for developing P/T limits and COPS setpoints (WCAP-14040-A, Rev. 4) and the methodology used to justify eliminating the reactor vessel closure head/vessel flange requirements (WCAP-16142-P, Revision 1).

Evaluation

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The proposed changes to the Technical Specifications and PTLRs do not affect any plant equipment, test methods, or plant operation, and are not initiators of any analyzed accident sequence. Operation in accordance with the proposed TS will ensure that all analyzed accidents will continue to be mitigated by the SSCs as previously analyzed. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?

   No. The proposed changes do not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. The changes to the P-T limits and COPS setpoints will ensure that appropriate fracture toughness margins are maintained to protect against reactor vessel failure during both normal and low temperature operation. The changes to the P-T limits and COPS setpoints are consistent with the methodology approved by the NRC in WCAP-14040, Rev. 4. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   No. The proposed changes will not adversely affect the operation of plant equipment or the function of any equipment assumed in the accident analysis. The utilization of ASME Code Case N-640 maintains the relative margin of safety commensurate with that which existed at the time that ASME B&PV Code, Section XI, Appendix G was approved in 1974 and will ensure an acceptable margin of safety. Therefore, the proposed changes do no involve a significant reduction in any margin of safety.

Enclosure 9

Vogtle Electric Generating Plant Units 1 and 2
Pen and Ink Changes

Technical Specification Changes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Cold Overpressure Protection Systems (COPS)

LCO 3.4.12 A COPS shall be OPERABLE with all safety injection pumps incapable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two RCS relief valves, as follows:

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 440 psig and ≤ 460 psig, or

3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint within specified limits.

b. The RCS depressurized and an RCS vent of ≥ 2.14 square inches (based on an equivalent length of 10 feet of pipe).

1.5

APPLICABILITY: MODE 4,
MODE 5,
MODE 6 when the reactor vessel head is on.

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

1. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

2. The safety injection pumps are not required to be incapable of injecting into the RCS until 4 hours after entering MODE 4 from MODE 3 provided the temperature of one or more RCS cold legs has not decreased below 325°F.

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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

LCO 3.4.3 "RCS Pressure and Temperature (P/T) Limits"

b. The power operated relief valve lift settings required to support the Cold Overpressure Protection Systems (COPS) shall be established and documented in the PTLR for the following:

LCO 3.4.12 "Cold Overpressure Protection Systems"

Insert

c. The RCS pressure and temperature limits for Unit 1 shall be those previously reviewed and approved by the NRC in Amendment No. 87 to Facility Operating License NPF-68. The RCS pressure and temperature limits for Unit 2 shall be those previously reviewed and approved by the NRC in Amendment No. 65 to Facility Operating License NPF-81. The acceptability of the P/T and COPS limits are documented in NRC letter "Vogtle Electric Generating Plant, Units 1 and 2 - Acceptance for Referencing of Pressure Temperature Limits Report," February 12, 1996. Specifically, the limits and methodology are described in the following documents:

1. Amendment No. 87 to Facility Operating License No. NPF-68, Vogtle Electric Generating Plant, Unit 1, June 8, 1995.

2. Amendment No. 65 to Facility Operating License No. NPF-81, Vogtle Electric Generating Plant, Unit 2, June 8, 1995.

(continued)

Insert

The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. WCAP-16142-P, Rev. 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2."

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

~~3. Letter from C. I. Grimes, NRC, to R. A. Newton, Westinghouse Electric Corporation, "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'" October 16, 1995.~~

~~4. Letter from C. K. McCoy, Georgia Power Company, to U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, "Vogtle Electric Generating Plant, Pressure and Temperature Limits Report," Enclosures 1 and 2, January 26, 1996.~~

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

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.6.8 PAM Report

When a Report is required by Condition G or K of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

Bases Changes

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section ~~III~~, XI Appendix G (Ref. 2).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section ~~III~~, XI Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.

(continued)

BASES

REFERENCES (continued)

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. WCAP-14040-A, Revision ~~2~~ 4, ~~January 1996.~~

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. In MODE 4 and below, overpressure prevention falls to two OPERABLE RCS relief valves or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the COPS must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the COPS requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the COPS acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, ~~examples of which follow:~~ as discussed below.

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

~~a. Inadvertent actuation of pressurizer heaters;~~

~~b. Loss of RHR cooling; or~~

a ~~c.~~ Reactor coolant pump (RCP) startup with temperature asymmetry ~~within the RCS or~~ between the RCS and steam generators.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

~~Heat Input Type Transients~~ (continued)

The following are required during the COPS MODES to ensure that mass and heat input transients do not occur, which either of the COPS overpressure protection means cannot handle:

a. Rendering both safety injection pumps incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. With no reactor coolant pump running, this value is reduced to 25°F at an RCS temperature of 350°F and varies linearly to 50°F at an RCS temperature of 200°F. LCO 3.4.6, "RCS Loops—MODE 4," and LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when both centrifugal charging pumps are actuated. Thus, the LCO requires both safety injection pumps to be incapable of injecting into the RCS during the COPS MODES.

Since neither one RCS relief valve nor the RCS vent can handle the pressure transient caused by accumulator injection when RCS temperature is low, the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits shown in the PTLR. The setpoints are derived by analyses that model the performance of the COPS, assuming

(continued)

BASES

APPLICABLE SAFETY ANALYSES

PORV Performance (continued)

the mass injection transient of two centrifugal charging pumps and the positive displacement pump injecting into the RCS, and the heat injection transient of starting an RCP with the RCS 50°F colder than the secondary coolant. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

-----NOTE-----
Although the positive displacement pump (PDP) was replaced with the normal charging pump (NCP), the current mass injection transient analysis assumes two centrifugal charging pumps and the positive displacement pump. Westinghouse performed an evaluation of the effect of replacing the PDP with the NCP and obtained acceptable results without reanalysis of the mass injection transient. Reference Westinghouse letter, GP-16838 from J. L. Tain to J. B. Beasley, Jr., dated August 13, 1998, COPS PORV Setpoint for New Charging Pump.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the COPS analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 440 psig and 460 psig (Ref. 9) will pass flow greater than that required for the limiting COPS transient while maintaining RCS pressure less than the P/T limit curve.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

RHR Suction Relief Valve Performance (continued)

As the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for COPS.

The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of ~~2.14~~ 1.57 square inches (based on an equivalent length of 10 feet of pipe; i.e., a vent capable of relieving ~~670~~ 685 gpm waterflow at ~~470~~ 722 psig) is capable of mitigating the allowed COPS overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the COPS configuration, with both safety injection pumps incapable of injecting into the RCS, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The COPS satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires that the COPS is OPERABLE. The COPS is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires both safety injection pumps to be incapable of injecting into the RCS and all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

(continued)

BASES

LCO (continued)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two RCS relief valves, as follows:

1. Two OPERABLE PORVs; or

A PORV is OPERABLE for the COPS when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits. The PORVs (PV-455A and PV-456A) are powered from 125 V MCCs 1/2AD1M and 1/2BD1M, respectively. The PORVs are to be considered OPERABLE whenever these MCCs are available to supply power.

2. Two OPERABLE RHR suction relief valves; or

An RHR suction relief valve is OPERABLE for the COPS when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between 440 psig and 460 psig, and testing has proven its ability to open at this setpoint.

3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of 1.5 ≥ ~~2.14~~ square inches (based on an equivalent length of 10 feet of pipe, i.e., capable of relieving ~~670~~ 685 gpm at ~~470~~ 722 psig).

Each of these methods of overpressure prevention is capable of mitigating the limiting COPS transient.

APPLICABILITY

This LCO is applicable in MODE 4, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES in MODES 1, 2, and 3. LCO 3.4.10, "Pressurizer Safety Valves," requires the

(continued)

BASES

ACTIONS (continued)

E.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

F.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, C, D, or E is not met; or

c. The COPS is inoperable for any reason other than Condition A, B, C, D, or E.

The vent must be sized ≥ ~~2.14~~ 1.5 square inches (based on an equivalent length of 10 feet of pipe) to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the ~~plant~~ unit in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, both safety injection pumps are verified incapable of injecting into the RCS, and the accumulator discharge isolation valves are verified closed and locked out.

The safety injection pumps are rendered incapable of injecting into the RCS through at least two independent means such that a single failure or single action will not result in an injection into the RCS.

The Frequency of within 4 hours after initial entry into MODE 4 from MODE 3 and prior to RCS cold leg temperature decreasing below 325°F (for the safety injection pumps) and 12 hours thereafter (for the safety injection pumps and accumulators) is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.3

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO. For Train A, the RHR suction relief valve is PSV-8708A and the suction isolation valves are HV-8701A and B. For Train B, the RHR suction relief valve is PSV-8708B and the suction isolation valves are HV-8702A and B.

The RHR suction valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The RCS vent of ≥ ~~2.14~~ 1.5 square inches (based on an equivalent length of 10 feet of pipe) is proven OPERABLE by verifying its open condition either:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

space

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

to be

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ 350°F and every 31 days on each required PORV to verify and, as necessary, adjust ~~its~~ lift setpoint. The COT will verify the setpoint is within the ~~PTLR~~ allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ 350°F. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time.

(continued)

Replace entire PTLR
with Revision 2

~~VOGTLE ELECTRIC GENERATING PLANT - UNIT 1~~

~~PRESSURE AND TEMPERATURE LIMITS REPORT~~

~~REVISION 1~~

~~Prepared by: Jimmy Paul Cash 3-19-01~~

~~Reviewed by: J. G. Bailey 03-20-01~~

Replace entire PTLR
with Revision 2

~~VOGTLE ELECTRIC GENERATING PLANT - UNIT 2~~

~~PRESSURE AND TEMPERATURE LIMITS REPORT~~

~~REVISION 1~~

Prepared by: Jimmy Paul Cash 3-19-01

Reviewed by: J. L. Bailey 03-20-01

Enclosure 10

Vogtle Electric Generating Plant Units 1 and 2
Final TS, Bases, and PTLR Changes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Cold Overpressure Protection Systems (COPS)

LCO 3.4.12 A COPS shall be OPERABLE with all safety injection pumps incapable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two RCS relief valves, as follows:

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 440 psig and ≤ 460 psig, or

3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint within specified limits.

b. The RCS depressurized and an RCS vent of ≥ 1.5 square inches (based on an equivalent length of 10 feet of pipe).

APPLICABILITY: MODE 4,
MODE 5,
MODE 6 when the reactor vessel head is on.

---NOTE---

1. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

2. The safety injection pumps are not required to be incapable of injecting into the RCS until 4 hours after entering MODE 4 from MODE 3 provided the temperature of one or more RCS cold legs has not decreased below 325°F.

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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

LCO 3.4.3 "RCS Pressure and Temperature (P/T) Limits"

b. The power operated relief valve lift settings required to support the Cold Overpressure Protection Systems (COPS) shall be established and documented in the PTLR for the following:

LCO 3.4.12 "Cold Overpressure Protection Systems"

c. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. WCAP-16142-P, Rev. 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2."

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

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

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.6.8 PAM Report

When a Report is required by Condition G or K of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G (Ref. 2).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.

(continued)

BASES

REFERENCES (continued)

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. WCAP-14040-A, Revision 4.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. In MODE 4 and below, overpressure prevention falls to two OPERABLE RCS relief valves or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the COPS must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the COPS requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the COPS acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients as discussed below.

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

a. Reactor coolant pump (RCP) startup with temperature asymmetry between the RCS and steam generators.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The following are required during the COPS MODES to ensure that mass and heat input transients do not occur, which either of the COPS overpressure protection means cannot handle:

a. Rendering both safety injection pumps incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. With no reactor coolant pump running, this value is reduced to 25°F at an RCS temperature of 350°F and varies linearly to 50°F at an RCS temperature of 200°F. LCO 3.4.6, "RCS Loops — MODE 4," and LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when both centrifugal charging pumps are actuated. Thus, the LCO requires both safety injection pumps to be incapable of injecting into the RCS during the COPS MODES.

Since neither one RCS relief valve nor the RCS vent can handle the pressure transient caused by accumulator injection when RCS temperature is low, the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limits shown in the PTLR. The setpoints are derived by analyses that model the performance of the COPS, assuming

(continued)

BASES

APPLICABLE SAFETY ANALYSES

PORV Performance (continued)

the mass injection transient of two centrifugal charging pumps and the positive displacement pump injecting into the RCS, and the heat injection transient of starting an RCP with the RCS 50°F colder than the secondary coolant. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

--NOTE--
Although the positive displacement pump (PDP) was replaced with the normal charging pump (NCP), the current mass injection transient analysis assumes two centrifugal charging pumps and the positive displacement pump. Westinghouse performed an evaluation of the effect of replacing the PDP with the NCP and obtained acceptable results without reanalysis of the mass injection transient. Reference Westinghouse letter, GP-16838 from J. L. Tain to J. B. Beasley, Jr., dated August 13, 1998, COPS PORV Setpoint for New Charging Pump.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the COPS analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 440 psig and 460 psig (Ref. 9) will pass flow greater than that required for the limiting COPS transient while maintaining RCS pressure less than the P/T limit curve.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

RHR Suction Relief Valve Performance (continued)

As the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for COPS.

The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 1.5 square inches (based on an equivalent length of 10 feet of pipe, i.e., a vent capable of relieving 685 gpm waterflow at 722 psig) is capable of mitigating the allowed COPS overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the COPS configuration, with both safety injection pumps incapable of injecting into the RCS, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The COPS satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires that the COPS is OPERABLE. The COPS is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires both safety injection pumps to be incapable of injecting into the RCS and all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

(continued)

BASES

LCO (continued)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two RCS relief valves, as follows:

   1. Two OPERABLE PORVs; or

      A PORV is OPERABLE for the COPS when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits. The PORVs (PV-455A and PV-456A) are powered from 125 V MCCs 1/2AD1M and 1/2BD1M, respectively. The PORVs are to be considered OPERABLE whenever these MCCs are available to supply power.

   2. Two OPERABLE RHR suction relief valves; or

      An RHR suction relief valve is OPERABLE for the COPS when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between 440 psig and 460 psig, and testing has proven its ability to open at this setpoint.

   3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

b. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ 1.5 square inches (based on an equivalent length of 10 feet of pipe, i.e., capable of relieving 685 gpm at 722 psig).

   Each of these methods of overpressure prevention is capable of mitigating the limiting COPS transient.

APPLICABILITY

This LCO is applicable in MODE 4, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits in MODES 1, 2, and 3. When the reactor vessel head is off, overpressurization cannot occur.

(continued)

BASES

APPLICABILITY (continued)

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

Two Notes modify the ACTIONS table. Note 1 prohibits entry into MODE 6 with the vessel head on from MODE 6 and MODE 5 from MODE 6 with the vessel head on. Entry into MODE 4 from MODE 5 is already prohibited by LCO 3.0.4. Note 2 permits entry into MODE 4 from MODE 3 with a PORV that is inoperable for the purpose of cold overpressure protection provided that RCS temperature is maintained above 275°F, and, within 36 hours, either: the PORV is restored to OPERABLE status; or, an RHR suction relief valve is placed in service so that the requirements of LCO 3.4.12 are met. Otherwise, the reactor vessel must be depressurized and vented in accordance with Required Action F.1. With only one PORV OPERABLE, the COPS remains capable of mitigating a design basis cold overpressurization event. However, the system cannot withstand a single failure of the remaining PORV. The current COPS enable temperature is established very conservatively at 350°F. However, the application of ASME Code Case N-514 would allow the enable temperature to be lowered to less than 275°F. Therefore, when entering this LCO from MODE 3 with one required PORV inoperable, maintaining RCS temperature above 275°F minimizes actual exposure to a cold overpressure event. Furthermore, requiring action within 36 hours minimizes the exposure to a single failure while allowing sufficient time to either restore the inoperable PORV or to place RHR in service. Note 2 is only applicable to the condition of entering MODE 4 from MODE 3 with one required PORV inoperable for the purpose of cold overpressure protection. If operating in MODE 4 and a failure of a required RCS relief valve occurs, Condition D applies.

(continued)

BASES

ACTIONS (continued)

A.1

With one or more safety injection pumps capable of injecting into the RCS, RCS overpressurization is possible.

Rendering the safety injection pumps incapable of injecting into the RCS within 4 hours to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1, C.1, and C.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 350°F, an accumulator pressure of 678 psig cannot exceed the COPS limits if the accumulators are fully injected. Depressurizing the accumulators below the COPS limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and that the likelihood that an event requiring COPS during this time is small.

D.1

In MODE 4, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves in any combination of the PORVS and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

(continued)

BASES

ACTIONS (continued)

E.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

F.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, C, D, or E is not met; or

c. The COPS is inoperable for any reason other than Condition A, B, C, D, or E.

The vent must be sized ≥ 1.5 square inches (based on an equivalent length of 10 feet of pipe) to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the unit in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, both safety injection pumps are verified incapable of injecting into the RCS, and the accumulator discharge isolation valves are verified closed and locked out.

The safety injection pumps are rendered incapable of injecting into the RCS through at least two independent means such that a single failure or single action will not result in an injection into the RCS.

The Frequency of within 4 hours after initial entry into MODE 4 from MODE 3 and prior to RCS cold leg temperature decreasing below 325°F (for the safety injection pumps) and 12 hours thereafter (for the safety injection pumps and accumulators) is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.3

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO. For Train A, the RHR suction relief valve is PSV-8708A and the suction isolation valves are HV-8701A and B. For Train B, the RHR suction relief valve is PSV-8708B and the suction isolation valves are HV-8702A and B.

The RHR suction valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The RCS vent of ≥ 1.5 square inches (based on an equivalent length of 10 feet of pipe) is proven OPERABLE by verifying its open condition either:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12 b.

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ 350°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ 350°F. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time.

(continued)

Southern Nuclear Company

Vogtle Unit 1

Pressure Temperature Limits Report

Revision 2, February 2004

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List Of Tables

List Of Figures

1.0 RCS Pressure Temperature Limits Report (PTLR)

This PTLR for Vogtle Unit 1 has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6. The TS addressed in this report are listed below:

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

LCO 3.4.12 Cold Overpressure Protection Systems (COPS)

Revisions to the PTLR shall be provided to the NRC after issuance.

2.0 RCS Pressure and Temperature (P/T) Limits

The limits for TS 3.4.3 are presented in the subsections which follow and were developed using the NRC approved methodology in WCAP-14040, Revision 4[1] with exception of WCAP-16142-P, Revision 1[2] (Elimination of the Flange Requirement). The operability requirements associated with COPS are specified in LCO 3.4.12 and were determined to adequately protect the RCS against brittle fracture in the event of a cold overpressure transient in accordance with the methodology specified in TS 5.6.6.

2.1 RCS P/T Limits (LCO 3.4.3)

2.1.1 The minimum boltup temperature is 60°F

2.1.2 The RCS temperature rate-of-change limits are:

a. A maximum heatup rate of 100°F in any one hour period.
b. A maximum cooldown rate of 100°F in any one hour period.
c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.3 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2-1 and 2-2.

3.0 Cold Overpressure Protection Systems (LCO 3.4.12)

The setpoints for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsections which follow. These setpoints have been developed using the NRC-approved methodology specified in TS 5.6.6.

3.1 Pressurizer PORV Setpoints

The pressurizer PORV setpoints are specified in Figure 3-1 and Table 3-1. The limits for the COPS setpoints are contained in the 36 EFPY steady-state curves (Table 2-2), which are beltline conditions and are not compensated for pressure differences between the pressurizer transmitter and the reactor midplane/beltline or for instrument inaccuracies. The pressure difference between the pressurizer transmitter and the reactor vessel midplane/beltline with four reactor coolant pumps in operation is 74 psi.

Note: These setpoints include an allowance for the 50°F thermal transport effect for heat injection transients. A calculation has been performed to confirm that the setpoints will maintain the system pressure within the established limits when the pressure difference between the pressure transmitter and reactor midplane and maximum temperature/pressure instrument uncertainties are applied to the setpoints.

4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in UFSAR Table 5.3.1-8. The results of these examinations shall be used to update Figures 2-1, 2-2, and 3-1.

The pressure vessel steel surveillance program (WCAP-11011[4]) is in compliance with Appendix H[3] to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature RT_{NDT}, which is determined in accordance with ASTM E23 [5]. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Code Case N-640[6] of Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"[7]. The surveillance capsule removal schedule meets the requirements of ASTM E185-82[8]. The removal schedule is provided in UFSAR Table 5.3.1-8.

5.0 Supplemental Data Tables

Table 5-1 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2[9], predictions.

Table 5-2 shows calculations of the surveillance material chemistry factors using surveillance capsule data. Note that in the calculation of the surveillance weld chemistry factor, the ratio procedure from Regulatory Guide 1.99, Revision 2 was followed. The ratio in question is equal to 1.02.

Table 5-3 provides the required Vogtle Unit 1 reactor vessel toughness data.

Table 5-4 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves and the PTS evaluation.

Table 5-5 and 5-6 show the calculation of the 1/4T and 3/4T adjusted reference temperature at 36 EFPY for each beltline material in the Vogtle Unit 1 reactor vessel. The limiting beltline material was the intermediate shell plate B8805-2.

Table 5-7 provides a summary of the adjusted reference temperature (ART) values of the Vogtle Unit 1 reactor vessel beltline materials at the 1/4T and 3/4T locations for 36 EFPY.

Table 5-8 provides RT_{PTS} values for Vogtle Unit 1 at 36 EFPY.

Table 5-9 provides RT_{PTS} values for Vogtle Unit 1 at 54 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B8805-2
LIMITING ART VALUES AT 36 EFPY: 1/4T, 110°F
3/4T, 95°F

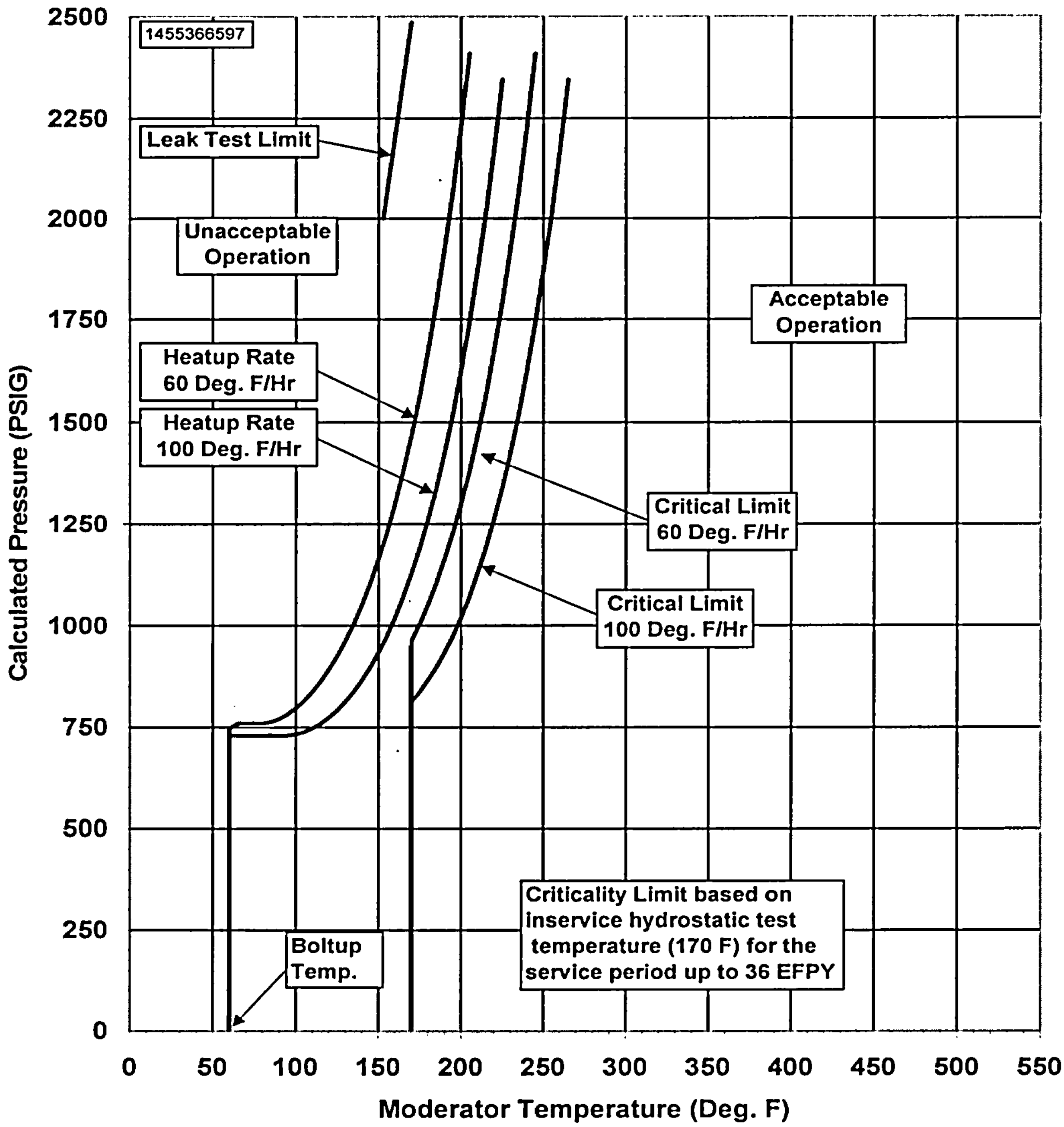

Figure 2-1 Vogtle Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 36 EFPY (Without Margins for Instrumentation Errors) *(Plotted Data provided on Table 2-1)*

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B8805-2
LIMITING ART VALUES AT 36 EFPY: 1/4T, 110°F
3/4T, 95°F

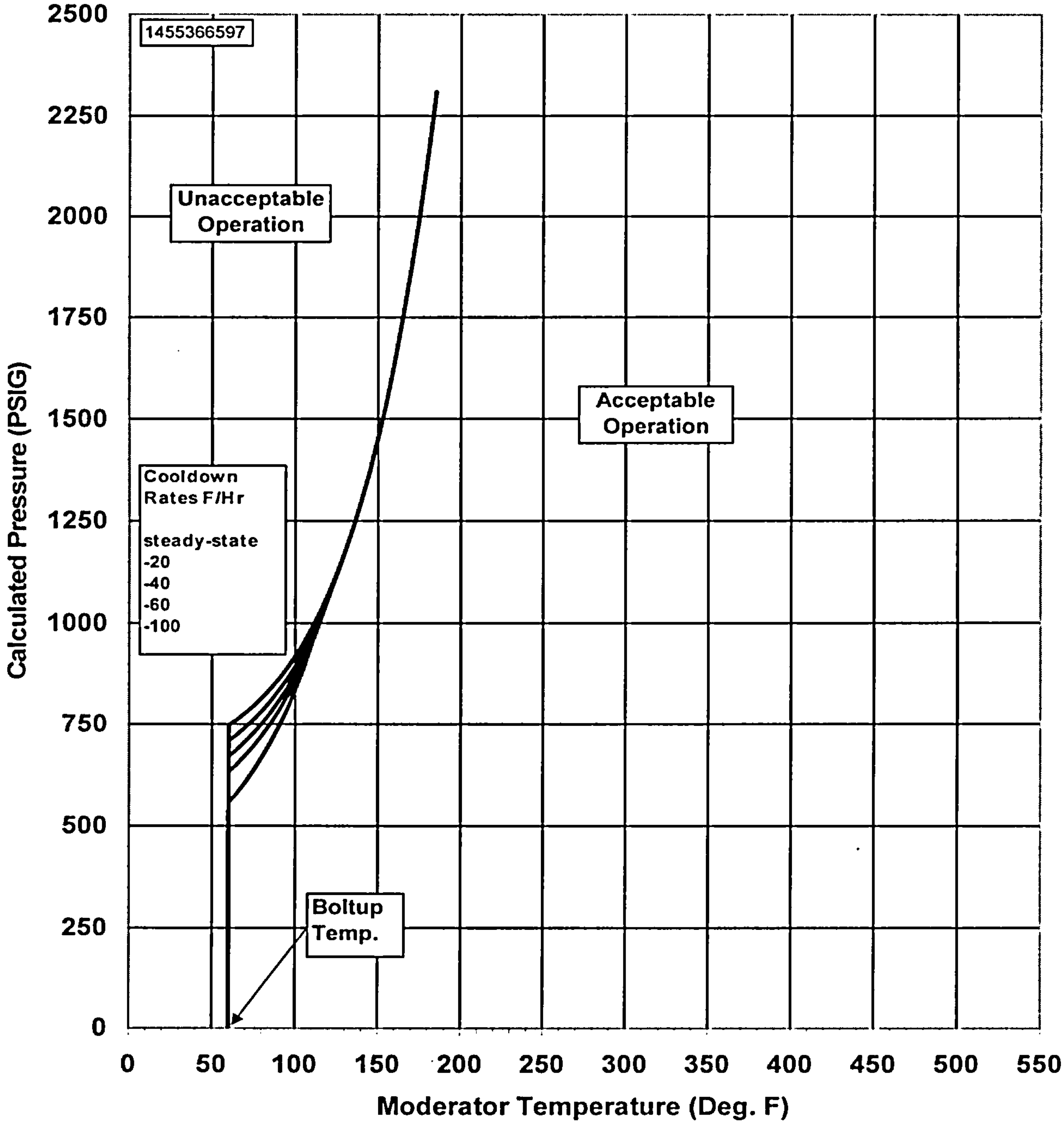

Figure 2-2 **Vogtle Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 36 EFPY (Without Margins for Instrumentation Error)** ***(Plotted Data provided on Table 2-2)***

Table 2-1
Vogtle Unit 1 Heatup Limits at 36 EFPY
(Without Uncertainties for Instrumentation Errors)

60°F/hr Heatup		60°F/hr Heatup Criticality Limit		100°F/hr Heatup		100°F/hr Heatup Criticality Limit		Leak Test Limit	
T	P	T	P	T	P	T	P	T	P
60	0	170	0	60	0	170	0	153	2000
60	747	170	760	60	730	170	730	170	2485
65	760	170	760	65	730	170	730		
70	760	170	760	70	730	170	730		
75	760	170	760	75	730	170	730		
80	760	170	763	80	730	170	730		
85	763	170	770	85	730	170	730		
90	770	170	782	90	730	170	730		
95	782	170	796	95	730	170	733		
100	796	170	815	100	733	170	739		
105	815	170	836	105	739	170	747		
110	836	170	862	110	747	170	759		
115	862	170	891	115	759	170	774		
120	891	170	925	120	774	170	791		
125	925	170	962	125	791	170	812		
130	962	175	1005	130	812	175	837		
135	1005	180	1052	135	837	180	865		
140	1052	185	1105	140	865	185	897		
145	1105	190	1163	145	897	190	933		
150	1163	195	1228	150	933	195	974		
155	1228	200	1300	155	974	200	1020		
160	1300	205	1380	160	1020	205	1071		
165	1380	210	1468	165	1071	210	1128		
170	1468	215	1566	170	1128	215	1191		
175	1566	220	1674	175	1191	220	1261		
180	1674	225	1793	180	1261	225	1339		
185	1793	230	1925	185	1339	230	1426		
190	1925	235	2070	190	1426	235	1521		
195	2070	240	2231	195	1521	240	1627		
200	2231	245	2408	200	1627	245	1743		
205	2408			205	1743	250	1872		
				210	1872	255	2014		
				215	2014	260	2171		
				220	2171	265	2344		
				225	2344				

Table 2-2

Vogtle Unit 1 Cooldown Limits at 36 EFPY
(Without Uncertainties for Instrumentation Errors)

Steady State		20°F/hr		40°F/hr		60°F/hr		100°F/hr	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	747	60	709	60	670	60	633	60	559
65	762	65	725	65	688	65	652	65	582
70	778	70	742	70	707	70	673	70	608
75	796	75	762	75	728	75	696	75	637
80	816	80	783	80	752	80	722	80	668
85	838	85	807	85	778	85	751	85	704
90	862	90	834	90	807	90	783	90	743
95	889	95	863	95	840	95	819	95	787
100	918	100	895	100	875	100	858	100	835
105	951	105	931	105	915	105	902	105	889
110	987	110	971	110	959	110	950	110	948
115	1027	115	1015	115	1007	115	1004		
120	1071	120	1063	120	1061				
125	1120	125	1117						
130	1173								
135	1233								
140	1299								
145	1371								
150	1452								
155	1541								
160	1639								
165	1747								
170	1867								
175	2000								
180	2146								
185	2308								

Table 3-1
Vogtle Unit 1 Data Points for the Maximum Allowable Nominal COPS PORV Setpoints

Temperature (Deg.F)	PORV Setpoint (psig)
70	612
90	612
140	642
201	760
202	760
350	760

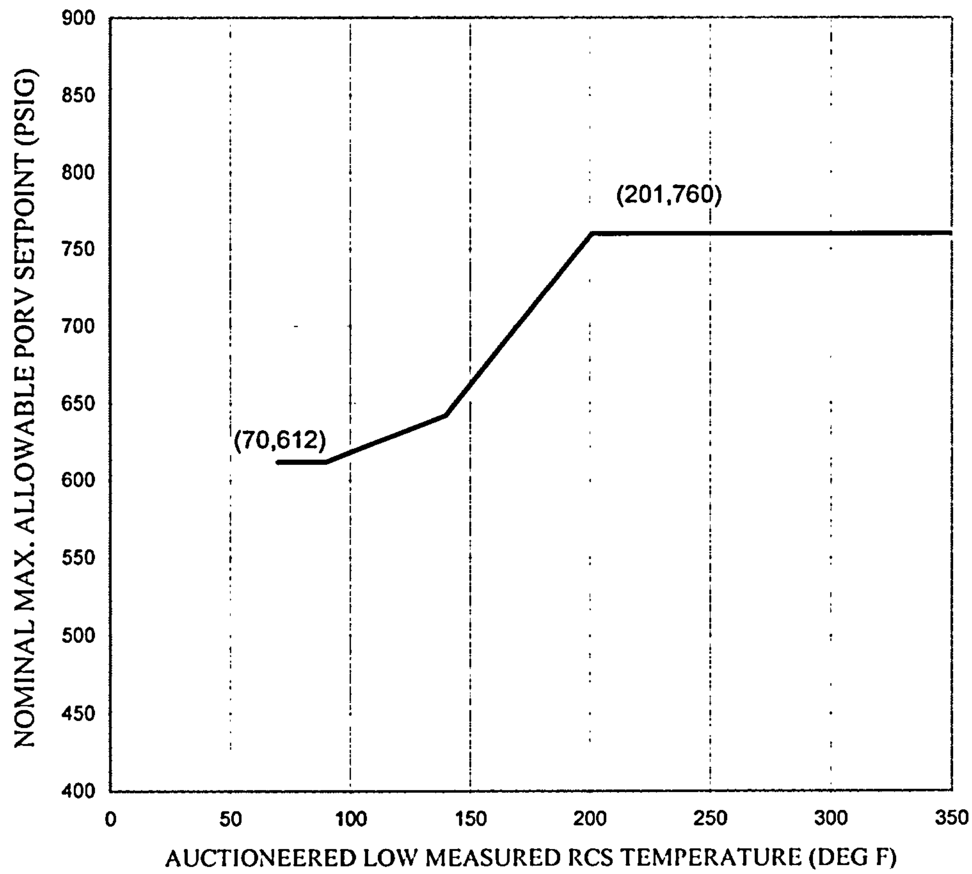

Figure 3-1: Vogtle Unit 1 Maximum Allowable Nominal PORV Setpoints for COPS

Table 5-1
Comparison of the Vogtle Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence ($x\ 10^{19}\ n/cm^2$)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) (a)	Measured (°F) (b)	Predicted (%) (a)	Measured (%)(c)
Intermediate Shell Plate B8805-3(e) (Longitudinal)	U	0.3691	27.8	13.6	15	0
	Y	1.276	41.0	31.9	20	0
	V	2.178	46.5	42.7	23	3
Intermediate Shell Plate B8805-3(e) (Transverse)	U	0.3691	27.8	0(d)	15	0
	Y	1.276	41.0	15.2	20	0
	V	2.178	46.5	33.8	23	2
Weld Metal(f)	U	0.3691	25.0	25.0	15	0
	Y	1.276	36.8	7.7	20	1
	V	2.178	41.8	0(d)	23	2
HAZ Metal	U	0.3691	--(e)	0(d)	--	5
	Y	1.276	--(e)	20.8	--	9
	V	2.178	--(e)	42.1	--	11

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1[11].

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82[8].

(d) The actual measured value of ΔRT_{NDT} for the intermediate shell plate (capsule U) is -9.58, the actual measured value of ΔRT_{NDT} for the weld metal (capsule V) is -1.34 and the actual measured value of ΔRT_{NDT} for the HAZ metal (capsule U) is -19.35. This physically should not occur, therefore for conservatism a value of zero will be reported (i.e. No Change in T_{30}).

(e) The heat number for lower shell plate B8805-3 is C-0623-1.

(f) The Surveillance weld was fabricated from Wire Heat No. 83653, Flux Type Linde 0091, Flux Lot No. 3536.

Table 5-2
Calculation of Chemistry Factors using Vogtle Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell	U	0.3691	0.725	13.6	9.9	0.526
Plate B8805-3[f]	Y	1.276	1.068	31.9	34.1	1.141
(Longitudinal)	V	2.178	1.211	42.7	51.7	1.467
Intermediate Shell	U	0.3691	0.725	0[e]	0.0	0.526
Plate B8805-3[f]	Y	1.276	1.068	15.2	16.2	1.141
(Transverse)	V	2.178	1.211	33.8	40.9	1.467
				SUM	152.8	6.268
	$CF_{B8805\text{-}3} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (152.8) \div (6.268) = 24.4°F$					
Surveillance Weld	U	0.3691	0.725	25.5 (25.0)[d]	18.5	0.526
Metal[g]	Y	1.276	1.068	7.9 (7.7)[d]	8.4	1.141
	V	2.178	1.211	0[e]	0.0	1.467
				SUM	26.9	3.134
	$CF_{Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (26.9) \div (3.134) = 8.6°F$					

Notes:

(a) f = Calculated fluence from capsule V dosimetry analysis results [10], (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from App. C of Ref. 10, rounded to one decimal point.

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 1.02.

(e) Actual values for ΔRT_{NDT} are -9.58 (Plate) and -1.34 (Weld). This physically should not occur, therefore for conservatism a value of zero will be used for this calculation..

(f) The heat number for lower shell plate B8805-3 is C-0623-1.

(g) Surveillance Weld was fabricated from Wire Heat No.83653, Flux Type Linde 0091, Flux Lot No. 3536.

Table 5-3
Reactor Vessel Beltline Material Unirradiated Toughness Properties for Vogtle Unit 1

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange B8801-1 (*Heat # 123J173VA1*)	- -	0.70	20°F
Vessel Flange B8802-1 (*Heat # 123H402VA1*)	- -	0.71	0°F
Intermediate Shell Plate B8805-1 (*Heat # C-0613-1*)	0.083	0.597	0°F
Intermediate Shell Plate B8805-2 (*Heat # C-0613-2*)	0.083	0.61	20°F
Intermediate Shell Plate B8805-3 (*Heat # C-0623-1*)	0.062	0.598	30°F
Lower Shell Plate B8606-1 (*Heat # C-2146-1*)	0.053	0.593	20°F
Lower Shell Plate B8606-2 (*Heat # C-2146-2*)	0.057	0.60	20°F
Lower Shell Plate B8606-3 (*Heat # C-2085-2*)	0.067	0.623	10°F
Intermediate Shell Longitudinal Welds, 101-124A, B & C[b]	0.042	0.102	-80°F
Lower Shell Longitudinal Welds, 101-142A, B & C[b]	0.042	0.102	-80°F
Circumferential Weld 101-171[b]	0.042	0.102	-80°F
Surveillance Program Weld Metal[b]	0.040	0.102	- -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) All welds, including the surveillance weld, were fabricated with weld wire heat number 83653, Linde 0091 Flux, Lot No. 3536. Per Regulatory Guide 1.99, Revision 2, "weight percent copper " and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld."

Table 5-4
Peak Calculated Neutron Fluence Projections at Key Azimuthal Locations on the Reactor Vessel Clad/Base Metal Interface for Vogtle Unit 1 (10^{19} n/cm^2, E > 1.0 MeV)

EFPY	Azimuthal Location					
	0°	15°	12.5° NP 30°	20° NP 30°	22.5° NP 30°	45°
8.57	0.290	0.433	0.533	0.349	0.289	0.523
16	0.527	0.781	0.953	0.624	0.517	0.936
36	1.17	1.72	2.09	1.36	1.13	2.05
54	1.74	2.56	3.10	2.03	1.68	3.05

Table 5-5
Vogtle Unit 1 Calculation of the ART Values for the 1/4T Location @ 36 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a]	ΔRT_{NDT}[b]	Margin[c]	ART[d]
Intermediate Shell Plate B8805-1	Position 1.1	53.1	1.06	0	56.3	34	90
Intermediate Shell Plate B8805-2	Position 1.1	53.1	1.06	20	56.3	34	**110**
Intermediate Shell Plate B8805-3	Position 1.1	38.4	1.06	30	40.7	34	105
	Position 2.1	24.4	1.06	30	26.0	17[e]	73
Lower Shell Plate B8606-1	Position 1.1	32.8	1.06	20	34.8	34	89
Lower Shell Plate B8606-2	Position 1.1	35.2	1.06	20	37.3	34	91
Lower Shell Plate B8606-3	Position 1.1	41.9	1.06	10	44.4	34	88
Inter. Shell Longitudinal Weld Seam 101-124A (*0° Azimuth*)	Position 1.1	34.5	0.899	-80	31.0	31	-18
	Position 2.1	8.6	0.899	-80	7.7	7.7[e]	-65
Inter. Shell Long. Weld Seams 101-124B,C (*120°, 240° Azimuth*)	Position 1.1	34.5	1.06	-80	36.6	36.6	-7
	Position 2.1	8.6	1.06	-80	9.1	9.1[e]	-62
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	34.5	1.06	-80	36.6	36.6	-7
	Position 2.1	8.6	1.06	-80	9.1	9.1[e]	-62
Lower Shell Long. Weld Seams 101-142A,C (*60°, 300° Azimuth*)	Position 1.1	34.5	1.06	-80	36.6	36.6	-7
	Position 2.1	8.6	1.06	-80	9.1	9.1[e]	-62
Lower Shell Long. Weld Seam 101-142B (*180° Azimuth*)	Position 1.1	34.5	0.899	-80	31.0	31	-18
	Position 2.1	8.6	0.899	-80	7.7	7.7[e]	-65

Notes:

(a) Initial RT_{NDT} values are measured values.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F); (Rounded per ASTM E29, using the "Rounding Method").
(e) Data deemed credible per Reference 10.
(f) Neutron Fluence value used for all material is the highest value (@ 30°) from Table 5-4 for 36 EFPY with exception to intermediate shell longitudinal weld 101-124A and lower shell longitudinal weld 101-142B which used the fluence at 0° from Table 5-4 for 36 EFPY.

Table 5-6
Vogtle Unit 1 Calculation of the ART Values for the 3/4T Location @ 36 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a]	ΔRT_{NDT}[b]	Margin[c]	ART[d]
Intermediate Shell Plate B8805-1	Position 1.1	53.1	0.773	0	41.0	34	75
Intermediate Shell Plate B8805-2	Position 1.1	53.1	0.773	20	41.0	34	95
Intermediate Shell Plate B8805-3	Position 1.1	38.4	0.773	30	29.7	29.7	89
	Position 2.1	24.4	0.773	30	18.9	17[e]	66
Lower Shell Plate B8606-1	Position 1.1	32.8	0.773	20	25.4	25.4	71
Lower Shell Plate B8606-2	Position 1.1	35.2	0.773	20	27.2	27.2	74
Lower Shell Plate B8606-3	Position 1.1	41.9	0.773	10	32.4	32.4	75
Inter. Shell Longitudinal Weld Seam 101-124A (*0° Azimuth*)	Position 1.1	34.5	0.622	-80	21.5	21.5	-37
	Position 2.1	8.6	0.622	-80	5.3	5.3[e]	-69
Inter. Shell Long. Weld Seams 101-124B,C (*120°, 240° Azimuth*)	Position 1.1	34.5	0.773	-80	26.7	26.7	-27
	Position 2.1	8.6	0.773	-80	6.6	6.6[e]	-67
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	34.5	0.773	-80	26.7	26.7	-27
	Position 2.1	8.6	0.773	-80	6.6	6.6[e]	-67
Lower Shell Long. Weld Seams 101-142A,C (*60°, 300° Azimuth*)	Position 1.1	34.5	0.773	-80	26.7	26.7	-27
	Position 2.1	8.6	0.773	-80	6.6	6.6[e]	-67
Lower Shell Long. Weld Seam 101-142B (*180° Azimuth*)	Position 1.1	34.5	0.622	-80	21.5	21.5	-37
	Position 2.1	8.6	0.622	-80	5.3	5.3[e]	-69

Notes:

(a) Initial RT_{NDT} values are measured values.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F); (Rounded per ASTM E29, using the "Rounding Method").
(e) Data deemed credible per Reference 10.
(f) Neutron Fluence value used for all material is the highest value (@ 30°) from Table 5-4 for 36 EFPY with exception to intermediate shell longitudinal weld 101-124A and lower shell longitudinal weld 101-142B which used the fluence at 0° from Table 5-4 for 36 EFPY.

Table 5-7
Summary of the Vogtle Unit 1 Reactor Vessel Beltline Material ART Values

Material	RG 1.99 R2 Method	¼ ART (°F)	3/4 ART (°F)
Intermediate Shell Plate B8805-1	Position 1.1	90	75
Intermediate Shell Plate B8805-2	Position 1.1	110	95
Intermediate Shell Plate B8805-3	Position 1.1	105	89
	Position 2.1	73	66
Lower Shell Plate B8606-1	Position 1.1	89	71
Lower Shell Plate B8606-2	Position 1.1	91	74
Lower Shell Plate B8606-3	Position 1.1	88	75
Inter. Shell Longitudinal Weld Seam 101-124A (*0° Azimuth*)	Position 1.1	-18	-37
	Position 2.1	-65	-69
Inter. Shell Long. Weld Seams 101-124B,C (*120°, 240° Azimuth*)	Position 1.1	-7	-27
	Position 2.1	-62	-67
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	-7	-27
	Position 2.1	-62	-67
Lower Shell Long. Weld Seams 101-142A,C (*60°, 300° Azimuth*)	Position 1.1	-7	-27
	Position 2.1	-62	-67
Lower Shell Long. Weld Seam 101-142B (*180° Azimuth*)	Position 1.1	-18	-37
	Position 2.1	-65	-69

Table 5-8

RT_{PTS} Calculations for Vogtle Unit 1 Beltline Region Materials at 36 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	$IRT_{NDT(U)}$[a]	ΔRT_{PTS}[b]	Margin[c]	RT_{PTS}[d]
Intermediate Shell Plate B8805-1	Position 1.1	53.1	1.20	0	63.7	34	98
Intermediate Shell Plate B8805-2	Position 1.1	53.1	1.20	20	63.7	34	118
Intermediate Shell Plate B8805-3	Position 1.1	38.4	1.20	30	46.1	34	110
	Position 2.1	24.4	1.20	30	29.4	17[e]	76
Lower Shell Plate B8606-1	Position 1.1	32.8	1.20	20	39.4	34	93
Lower Shell Plate B8606-2	Position 1.1	35.2	1.20	20	42.2	34	96
Lower Shell Plate B8606-3	Position 1.1	41.9	1.20	10	50.3	34	94
Inter. Shell Longitudinal Weld Seam 101-124A (*0° Azimuth*)	Position 1.1	34.5	1.04	-80	35.9	35.9	-8
	Position 2.1	8.6	1.04	-80	8.9	8.9[e]	-62
Inter. Shell Long. Weld Seams 101-124B, C (*120°, 240° Azimuth*)	Position 1.1	34.5	1.20	-80	41.4	41.4	3
	Position 2.1	8.6	1.20	-80	10.3	10.3[e]	-59
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	34.5	1.20	-80	41.4	41.4	3
	Position 2.1	8.6	1.20	-80	10.3	10.3[e]	-59
Lower Shell Long. Weld Seams 101-142A, C (*60°, 300° Azimuth*)	Position 1.1	34.5	1.20	-80	41.4	41.4	3
	Position 2.1	8.6	1.20	-80	10.3	10.3[e]	-59
Lower Shell Long. Weld Seam 101-142B (*180° Azimuth*)	Position 1.1	34.5	1.04	-80	35.9	35.9	-8
	Position 2.1	8.6	1.04	-80	8.9	8.9[e]	-62

Notes:

(a) Initial RT_{NDT} values are measured values

(b) $\Delta RT_{PTS} = CF * FF$

(c) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

(d) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(e) Data deemed credible per Reference 10.

(f) Neutron Fluence value used for all material is the highest value (@ 30°) from Table 5-4 for 36 EFPY with exception to intermediate shell longitudinal weld 101-124A and lower shell longitudinal weld 101-142B which used the fluence at 0° from Table 5-4 for 36 EFPY.

Table 5-9

RT_{PTS} Calculations for Vogtle Unit 1 Beltline Region Materials at 54 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	$IRT_{NDT(U)}$[a]	ΔRT_{PTS}[b]	Margin[c]	RT_{PTS}[d]
Intermediate Shell Plate B8805-1	Position 1.1	53.1	1.30	0	69.0	34	103
Intermediate Shell Plate B8805-2	Position 1.1	53.1	1.30	20	69.0	34	123
Intermediate Shell Plate B8805-3	Position 1.1	38.4	1.30	30	49.9	34	114
	Position 2.1	24.4	1.30	30	31.7	17[e]	79
Lower Shell Plate B8606-1	Position 1.1	32.8	1.30	20	42.6	34	97
Lower Shell Plate B8606-2	Position 1.1	35.2	1.30	20	45.8	34	100
Lower Shell Plate B8606-3	Position 1.1	41.9	1.30	10	54.5	34	99
Inter. Shell Longitudinal Weld Seam 101-124A (*0° Azimuth*)	Position 1.1	34.5	1.15	-80	39.7	39.7	0
	Position 2.1	8.6	1.15	-80	9.9	9.9[e]	-60
Inter. Shell Long. Weld Seams 101-124B, C (*120°, 240° Azimuth*)	Position 1.1	34.5	1.30	-80	44.9	44.9	10
	Position 2.1	8.6	1.30	-80	11.2	11.2[e]	-58
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	34.5	1.30	-80	44.9	44.9	10
	Position 2.1	8.6	1.30	-80	11.2	11.2[e]	-58
Lower Shell Long. Weld Seams 101-142A, C (*60°, 300° Azimuth*)	Position 1.1	34.5	1.30	-80	44.9	44.9	10
	Position 2.1	8.6	1.30	-80	11.2	11.2[e]	-58
Lower Shell Long. Weld Seam 101-142B (*180° Azimuth*)	Position 1.1	34.5	1.15	-80	39.7	39.7	0
	Position 2.1	8.6	1.15	-80	9.9	9.9[e]	-60

Notes:

(a) Initial RT_{NDT} values are measured values

(b) $\Delta RT_{PTS} = CF * FF$

(c) $M = 2 * (\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

(d) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(e) Data deemed credible per Reference 10.

(f) Neutron Fluence value used for all material is the highest value (@ 30°) from Table 5-4 for 36 EFPY with exception to intermediate shell longitudinal weld 101-124A and lower shell longitudinal weld 101-142B which used the fluence at 0° from Table 5-4 for 36 EFPY.

6.0 References

1. WCAP-14040-NP-A, Revision 4, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al.

2. WCAP-16142-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2", Warren Bamford, et. al., February 2004.

3. Code of Federal Regulations, 10CFR50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C.

4. WCAP-11011, *Georgia Power Company Alvin W. Vogtle Unit No. 2 Reactor Vessel Radiation Surveillance Program*, L. R. Singer, February 1986.

5. ASTM E23 *Standard Test Method Notched Bar Impact Testing of Metallic Materials*, in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

6. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

7. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, *Fracture Toughness Criteria for Protection Against Failure.*

8. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.*

9. Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*, U.S. Nuclear Regulatory Commission, May 1988.

10. WCAP-15067, *Analysis of Capsule V From the Southern Nuclear Vogtle Electric Generating Plant Unit 1 Reactor Vessel Radiation Surveillance Program*, T.J. Laubham, et. al., Dated September 1998. [*Note that the Testing/Analysis reports for surveillance capsules U and Y from Vogtle Unit 1 were documented under WCAP-12256 and WCAP-13931, Rev. 1, respectively.*]

11. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.

Southern Nuclear Company

Vogtle Unit 2

Pressure Temperature Limits Report

Revision 2, February 2004

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List Of Tables

List Of Figures

1.0 RCS Pressure Temperature Limits Report (PTLR)

This PTLR for Vogtle Unit 2 has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6. The TS addressed in this report are listed below:

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

LCO 3.4.12 Cold Overpressure Protection Systems (COPS)

Revisions to the PTLR shall be provided to the NRC after issuance.

2.0 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

The limits for TS 3.4.3 are presented in the subsections which follow and were developed using the NRC approved methodology in WCAP-14040, Revision 4[1] with exception of WCAP-16142-P, Revision 1[2] (elimination of the flange requirement). The operability requirements associated with the COPS are specified in LCO 3.4.12 and were determined to adequately protect the RCS against brittle fracture in the event of a cold overpressure transient in accordance with the methodology specified in TS 5.6.6.

2.1 RCS P/T Limits (LCO 3.4.3)

2.1.1 The minimum boltup temperature is 60°F

2.1.2 The RCS temperature rate-of-change limits are:

a. A maximum heatup rate of 100°F in any one hour period.
b. A maximum cooldown rate of 100°F in any one hour period.
c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.3 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2-1 and 2-2.

3.0 Cold Overpressure Protection Systems (LCO 3.4.12)

The setpoints for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsections which follow. These setpoints have been developed using the NRC-approved methodology specified in TS 5.6.6.

3.1 Pressurizer PORV Setpoints

The pressurizer PORV setpoints are specified in Figure 3-1 and Table 3-1. The limits for the COPS setpoints are contained in the 36 EFPY steady-state curves (Table 2-2), which are beltline conditions and are not compensated for pressure differences between the pressurizer transmitter and the reactor midplane/beltline or for instrument inaccuracies. The pressure difference between the pressurizer transmitter and the reactor vessel midplane/beltline with four reactor coolant pumps in operation is 74 psi.

Note: These setpoints include an allowance for the 50°F thermal transport effect for heat injection transients. A calculation has been performed to confirm that the setpoints will maintain the system pressure within the established limits when the pressure difference between the pressure transmitter and reactor midplane and maximum temperature/pressure instrument uncertainties are applied to the setpoints.

4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in UFSAR Table 5.3.1-9. The results of these examinations shall be used to update Figures 2-1, 2-2, and 3-1.

The pressure vessel steel surveillance program (WCAP-11381[4]) is in compliance with Appendix H[3] to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature RT_{NDT}, which is determined in accordance with ASTM E23 [5]. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Code Case N-640[6] of Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"[7]. The surveillance capsule removal schedule meets the requirements of ASTM E185-82[8]. The removal schedule is provided in UFSAR Table 5.3.1-9.

5.0 Supplemental Data Tables

Table 5-1 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2[9], predictions.

Table 5-2 shows calculations of the surveillance material chemistry factors using surveillance capsule data. Note that in the calculation of the surveillance weld chemistry factor, the ratio procedure from Regulatory Guide 1.99, Revision 2 was followed. The ratio in question is equal to 1.19.

Table 5-3 provides the required Vogtle Unit 2 reactor vessel toughness data.

Table 5-4 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves and the PTS evaluation.

Table 5-5 and 5-6 show the calculation of the 1/4T and 3/4T adjusted reference temperature at 36 EFPY for each beltline material in the Vogtle Unit 2 reactor vessel. The limiting beltline material was the lower shell plate R8-1.

Table 5-7 provides a summary of the adjusted reference temperature (ART) values of the Vogtle Unit 2 reactor vessel beltline materials at the 1/4T and 3/4T locations for 36 EFPY.

Table 5-8 provides RT_{PTS} values for Vogtle Unit 2 at 36 EFPY.

Table 5-9 provides RT_{PTS} values for Vogtle Unit 2 at 54 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R8-1
LIMITING ART VALUES AT 36 EFPY: 1/4T, 120°F
3/4T, 107°F

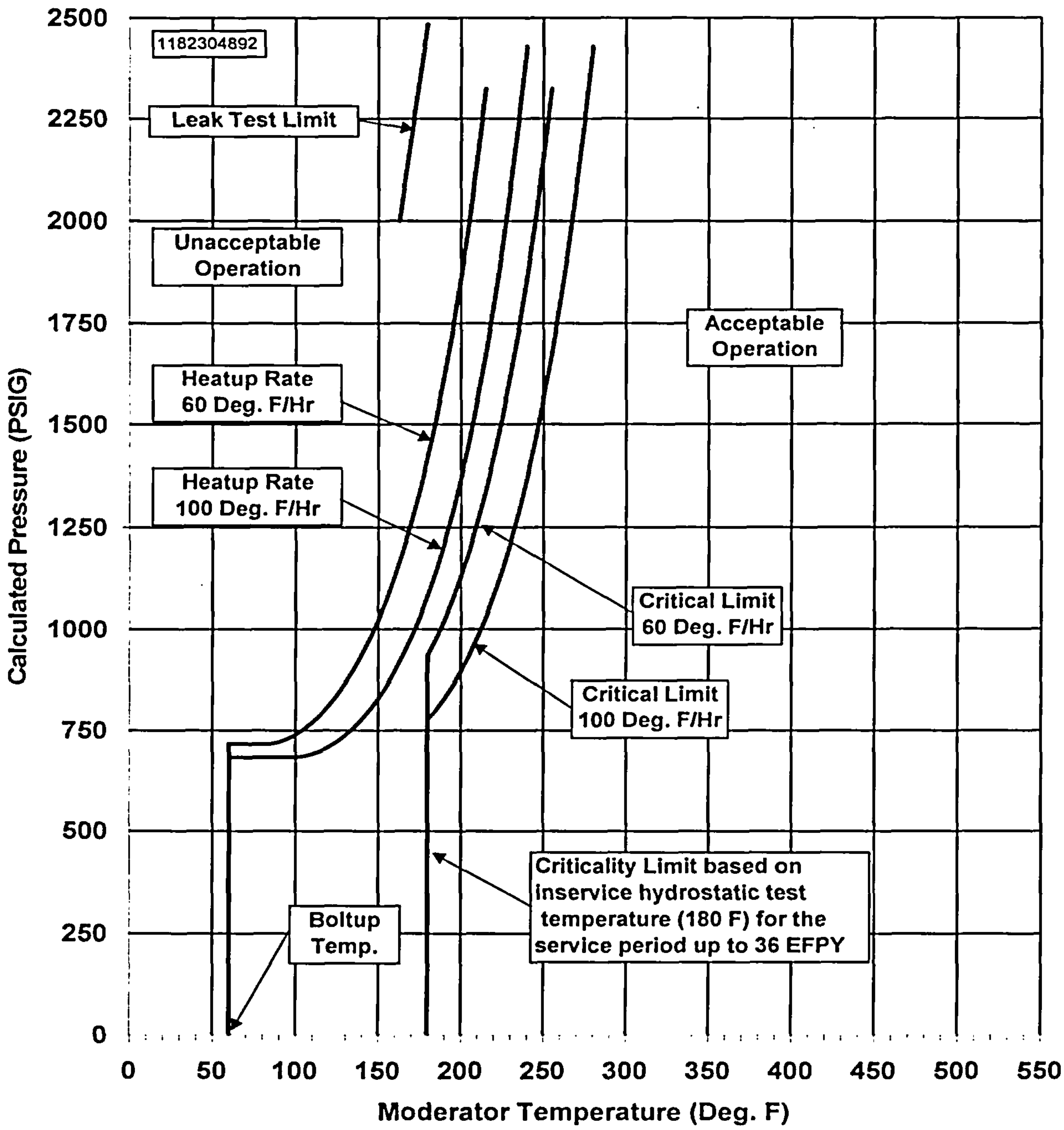

Figure 2-1 Vogtle Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable for the First 36 EFPY (Without Margins for Instrumentation Errors) *(Plotted Data provided on Table 2-1)*

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R8-1
LIMITING ART VALUES AT 36 EFPY: 1/4T, 120°F
3/4T, 107°F

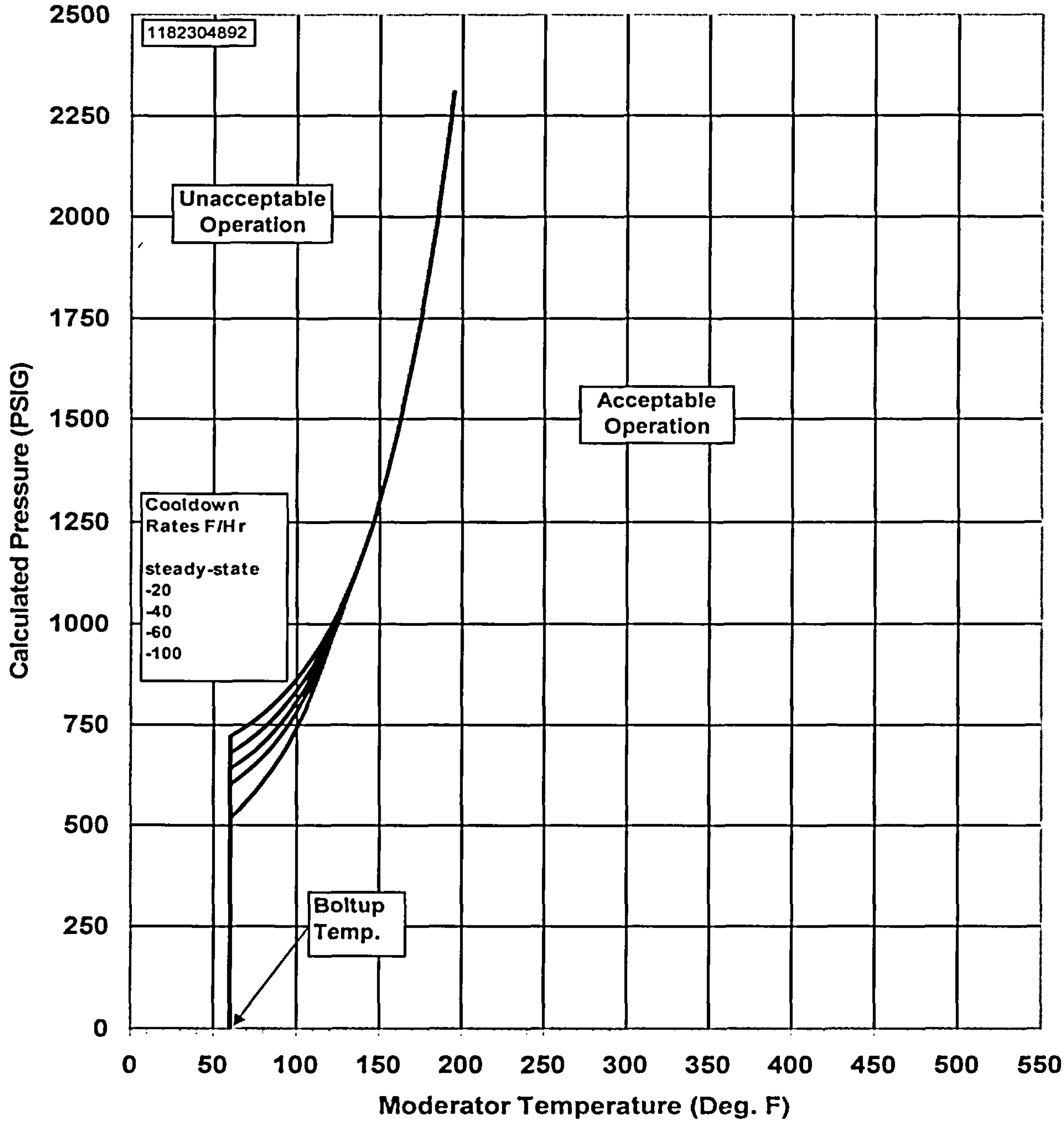

Figure 2-2 Vogtle Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 36 EFPY (Without Margins for Instrumentation Error) *(Plotted Data provided on Table 2-2)*

Table 2-1
Vogtle Unit 2 Heatup Limits at 36 EFPY
(Without Uncertainties for Instrumentation Errors)

60°F/hr Heatup		60°F/hr Heatup Criticality Limit		100°F/hr Heatup		100°F/hr Heatup Criticality Limit		Leak Test Limit	
T	P	T	P	T	P	T	P	T	P
60	0	180	0	60	0	180	0	163	2000
60	717	180	734	60	684	180	684	180	2485
65	717	180	727	65	684	180	684		
70	717	180	720	70	684	180	684		
75	717	180	717	75	684	180	684		
80	717	180	717	80	684	180	684		
85	717	180	721	85	684	180	684		
90	721	180	729	90	684	180	684		
95	729	180	739	95	684	180	684		
100	739	180	753	100	684	180	686		
105	753	180	769	105	686	180	691		
110	769	180	788	110	691	180	699		
115	788	180	811	115	699	180	709		
120	811	180	837	120	709	180	722		
125	837	180	866	125	722	180	737		
130	866	180	899	130	737	180	756		
135	899	180	936	135	756	180	777		
140	936	185	977	140	777	185	801		
145	977	190	1023	145	801	190	829		
150	1023	195	1074	150	829	195	860		
155	1074	200	1130	155	860	200	896		
160	1130	205	1193	160	896	205	935		
165	1193	210	1262	165	935	210	979		
170	1262	215	1339	170	979	215	1029		
175	1339	220	1424	175	1029	220	1084		
180	1424	225	1517	180	1084	225	1144		
185	1517	230	1621	185	1144	230	1212		
190	1621	235	1735	190	1212	235	1287		
195	1735	240	1861	195	1287	240	1369		
200	1861	245	2000	200	1369	245	1461		
205	2000	250	2154	205	1461	250	1562		
210	2154	255	2324	210	1562	255	1673		
215	2324			215	1673	260	1796		
				220	1796	265	1932		
				225	1932	270	2082		
				230	2082	275	2248		
				235	2248	280	2430		
				240	2430				

Table 2-2

Vogtle Unit 2 Cooldown Limits at 36 EFPY
(Without Uncertainties for Instrumentation Errors)

Steady State		20°F/hr		40°F/hr		60°F/hr		100°F/hr	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	722	60	681	60	640	60	599	60	518
65	734	65	694	65	654	65	614	65	537
70	747	70	708	70	670	70	632	70	557
75	762	75	724	75	687	75	651	75	581
80	778	80	742	80	706	80	672	80	606
85	796	85	761	85	728	85	695	85	635
90	816	90	783	90	752	90	721	90	667
95	838	95	807	95	778	95	750	95	703
100	862	100	833	100	807	100	782	100	742
105	889	105	863	105	839	105	818	105	786
110	918	110	895	110	875	110	858	110	834
115	951	115	931	115	914	115	901	115	888
120	987	120	971	120	958	120	950	120	948
125	1027	125	1015	125	1007	125	1003		
130	1071	130	1063	130	1060				
135	1120	135	1117						
140	1173								
145	1233								
150	1299								
155	1371								
160	1452								
165	1541								
170	1639								
175	1747								
180	1867								
185	2000								
190	2146								
195	2308								

Table 3-1
Vogtle Unit 2 Data Points for the Maximum Allowable Nominal COPS PORV Setpoints

Temperature (Deg.F)	PORV Setpoint (psig)
70	580
90	580
140	612
201	760
202	760
350	760

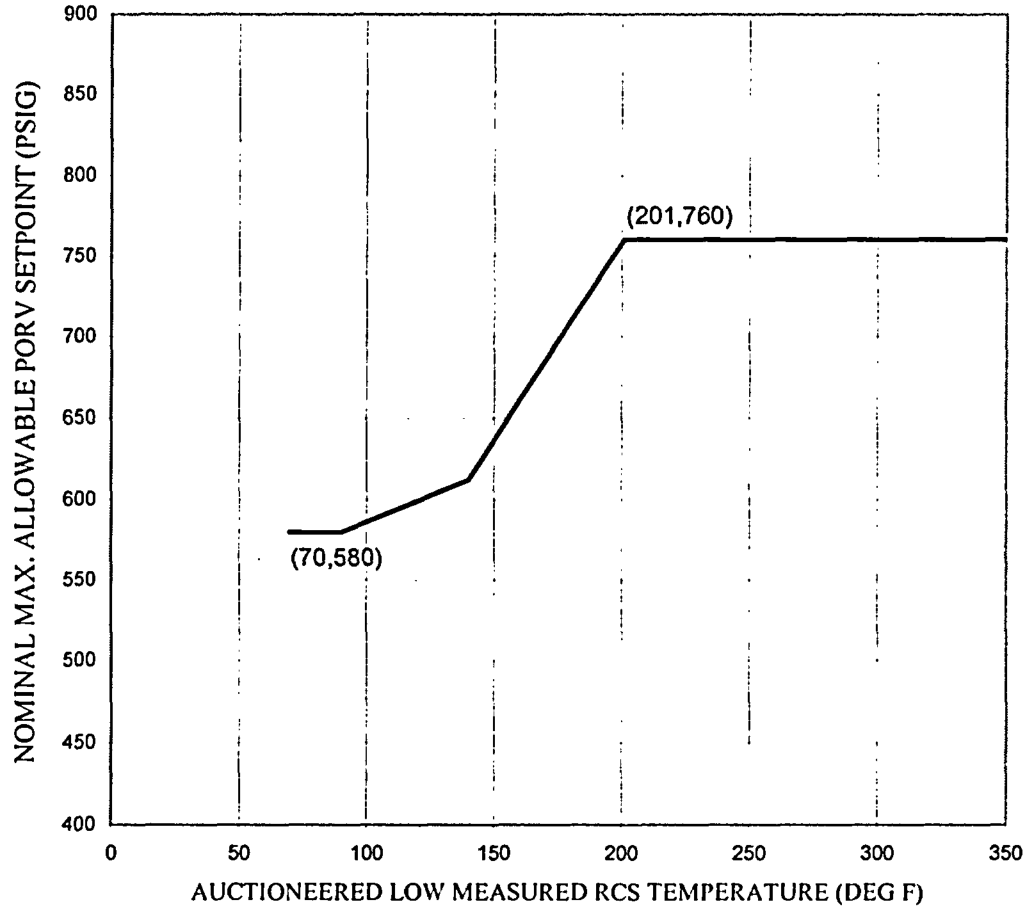

Figure 3-1: Vogtle Unit 2 Maximum Allowable Nominal PORV Setpoints for COPS

Table 5-1
Comparison of the Vogtle Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence ($x 10^{19}$ n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%)[c]
Lower Shell[e]	U	0.397	23.06	2.12	15	0
Plate B8628-1	Y	1.27	33.17	5.76	20	0
(Longitudinal)	X	2.01	36.89	29.35	22	3
Lower Shell[e]	U	0.397	23.06	0.00[d]	15	0
Plate B8628-1	Y	1.27	33.17	1.93	20	0
(Transverse)	X	2.01	36.89	29.72	22	7
Weld Metal[f]	U	0.397	27.08	0.00[d]	15	0
	Y	1.27	38.95	18.59	20	7
	X	2.01	43.32	20.07	22	5
HAZ Metal	U	0.397	---	0.00[d]	---	0
	Y	1.27	---	0.00[d]	---	0
	X	2.01	---	0.00[d]	---	7

Notes:

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1[11].

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82[8].

(d) Actual values for ΔRT_{NDT} are -7.14 (Plate), -17.49 (Weld), -24.05 (HAZ Cap. U), -9.86 (HAZ Cap. Y) and -2.1 (HAZ Cap. X). This physically should not occur, therefore for conservatism a value of zero will be reported (i.e. No Change in T_{30}).

(e) The heat number for lower shell plate B8628-1 is C-3500-2.

(f) The Surveillance weld was fabricated from Wire Heat No. 87005, Flux Type Linde 124, Flux Lot No. 1061.

Table 5-2
Calculation of Chemistry Factors using Vogtle Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Plate B8628-1[f] (Longitudinal)	U	0.397	0.744	2.1	1.6	0.554
	Y	1.27	1.07	5.8	6.2	1.14
	X	2.01	1.19	29.4	35.0	1.42
Lower Shell Plate B8628-1[f] (Transverse)	U	0.397	0.744	0.0[e]	0	0.554
	Y	1.27	1.07	1.9	2.0	1.14
	X	2.01	1.19	29.7	35.3	1.42
				SUM:	80.1	6.228
	$CF_{B8628-1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (80.1) \div (6.228) = 12.9°F$					
Surveillance Weld Material[g]	U	0.397	0.744	0.0[e]	0	0.554
	Y	1.27	1.07	22.1(18.6)[d]	23.6	1.14
	X	2.01	1.19	23.9(20.1)[d]	28.4	1.42
				SUM:	52.0	3.114
	$CF_{Surv. Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (52.0) \div (3.114) = 16.7°F$					

Notes:

(a) f = Calculated fluence from capsule X dosimetry analysis results [10], (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from App. B of Ref. 10, rounded to one decimal point.

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 1.19.

(e) Actual values for ΔRT_{NDT} are -7.14 (Plate) and -17.49 (Weld). This physically should not occur; therefore for conservatism a value of zero will be used.

(f) The heat number for lower shell plate B8628-1 is C-3500-2.

(g) Surveillance Weld was fabricated from Wire Heat No. 87005, Flux Type Linde 124, Flux Lot No. 1061.

Table 5-3
Reactor Vessel Beltline Material Unirradiated Toughness Properties for Vogtle Unit 2

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT} [a]
Closure Head Flange R7-1 (*Heat # 125L630VA1*)	---	0.72	10°F
Vessel Flange R1-1	---	0.87	-60°F
Intermediate Shell Plate R4-1 (*Heat # C-3527-1*)	0.07	0.63	10°F
Intermediate Shell Plate R4-2 (*Heat # C-3527-2*)	0.06	0.61	10°F
Intermediate Shell Plate R4-3 (*Heat # C-3552-1*)	0.05	0.60	30°F
Lower Shell Plate B8825-1 (*Heat # C-3500-1*)	0.06	0.62	40°F
Lower Shell Plate R8-1 (*Heat # C-4304-1*)	0.07	0.63	40°F
Lower Shell Plate B8628-1 (*Heat # C-3500-2*)	0.05	0.59	50°F
Intermediate Shell Longitudinal Weld Seams 101-124A, B & C	0.05	0.15	-10°F
Lower Shell Longitudinal Weld Seams 101-142A, B & C	0.05	0.15	-10°F
Intermediate to Lower Shell Plate Circumferential Weld Seam 101-171	0.05	0.15	-30°F
Surveillance Weld[b]	0.04	0.13	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) The weld material in the Vogtle Unit 2 surveillance program was made of the same wire and flux as the reactor vessel intermediate to lower shell girth seam weld (101-171). These welds were fabricated using weld wire heat no. 87005, Linde 124 Flux, lot no. 1061. The intermediate shell longitudinal weld seams (101-124A,B,C) and the lower shell longitudinal weld seams (101-142A,B,C) were fabricated using weld wire heat no. 87005, Linde 0091 Flux, lot no. 0145. Hence the surveillance weld is representative of all beltline welds.

Table 5-4
Peak Calculated Neutron Fluence Projections at Key Azimuthal Locations on the Reactor Vessel Clad/Base Metal Interface for Vogtle Unit 2 (10^{19} n/cm^2, E > 1.0 MeV)

EFPY	Azimuthal Location			
	0°	15°	30°	45°
7.62	0.263	0.381	0.456	0.452
16	0.525	0.763	0.914	0.908
36	1.15	1.67	2.01	2.01
54	1.71	2.49	2.99	2.99

Table 5-5
Vogtle Unit 2 Calculation of the ART Values for the 1/4T Location @ 36 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a]	ΔRT_{NDT}[b]	Margin[c]	ART[d]
Intermediate Shell Plate R4-1	Position 1.1	44.0	1.051	10	46.2	34	90
Intermediate Shell Plate R4-2	Position 1.1	37.0	1.051	10	38.9	34	83
Intermediate Shell Plate R4-3	Position 1.1	31.0	1.051	30	32.6	32.6	95
Lower Shell Plate B8825-1	Position 1.1	37.0	1.051	40	38.9	34	113
Lower Shell Plate R8-1	Position 1.1	44.0	1.051	40	46.2	34	**120**
Lower Shell Plate B8628-1	Position 1.1	31.0	1.051	50	32.6	32.6	115
	Position 2.1	12.9	1.051	50	13.6	13.6[e]	77
Intermediate Shell Longitudinal	Position 1.1	43.3	1.051	-10	45.5	45.5	81
Weld Seams 101-124A, B, C	Position 2.1	16.7	1.051	-10	17.6	17.6[e]	25
Lower Shell Longitudinal	Position 1.1	43.3	1.051	-10	45.5	45.5	81
Weld Seams 101-142A, B, C	Position 2.1	16.7	1.051	-10	17.6	17.6[e]	25
Intermediate to Lower Shell	Position 1.1	43.3	1.051	-30	45.5	45.5	61
Circ. Weld Seam 101-171	Position 2.1	16.7	1.051	-30	17.6	17.6[e]	5

Notes:

(a) Initial RT_{NDT} values are measured values.

(b) $\Delta RT_{NDT} = CF * FF$

(c) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F); (Rounded per ASTM E29, using the "Rounding Method").

(e) Data deemed credible per Reference 10.

(f) Neutron Fluence value used for all material is the highest value from Table 5-4 for 36 EFPY.

Table 5-6
Vogtle Unit 2 Calculation of the ART Values for the 3/4T Location @ 36 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a]	ΔRT_{NDT}[b]	Margin[c]	ART[d]
Intermediate Shell Plate R4-1	Position 1.1	44.0	0.763	10	33.6	33.6	77
Intermediate Shell Plate R4-2	Position 1.1	37.0	0.763	10	28.2	28.2	66
Intermediate Shell Plate R4-3	Position 1.1	31.0	0.763	30	23.7	23.7	77
Lower Shell Plate B8825-1	Position 1.1	37.0	0.763	40	28.2	28.2	96
Lower Shell Plate R8-1	Position 1.1	44.0	0.763	40	33.6	33.6	**107**
Lower Shell Plate B8628-1	Position 1.1	31.0	0.763	50	23.7	23.7	97
	Position 2.1	12.9	0.763	50	9.8	9.8[e]	70
Intermediate Shell Longitudinal	Position 1.1	43.3	0.763	-10	33.0	33.0	56
Weld Seams 101-124A, B, C	Position 2.1	16.7	0.763	-10	12.7	12.7[e]	15
Lower Shell Longitudinal	Position 1.1	43.3	0.763	-10	33.0	33.0	56
Weld Seams 101-142A, B, C	Position 2.1	16.7	0.763	-10	12.7	12.7[e]	15
Intermediate to Lower Shell	Position 1.1	43.3	0.763	-30	33.0	33.0	36
Circ. Weld Seam 101-171	Position 2.1	16.7	0.763	-30	12.7	12.7[e]	-5

Notes:

(a) Initial RT_{NDT} values are measured values.
(b) $\Delta RT_{NDT} = CF * FF$
(c) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F); (Rounded per ASTM E29, using the "Rounding Method").
(e) Data deemed credible per Reference 10.
(f) Neutron Fluence value used for all material is the highest value from Table 5-4 for 32 EFPY.

Table 5-7
Summary of the Vogtle Unit 2 Reactor Vessel Beltline Material ART Values

Material	RG 1.99 R2 Method	1/4 ART (°F)	3/4 ART (°F)
Intermediate Shell Plate R4-1	Position 1.1	90	77
Intermediate Shell Plate R4-2	Position 1.1	83	66
Intermediate Shell Plate R4-3	Position 1.1	95	77
Lower Shell Plate B8825-1	Position 1.1	113	96
Lower Shell Plate R8-1	Position 1.1	**120**	**107**
Lower Shell Plate B8628-1	Position 1.1	115	97
	Position 2.1	77	70
Intermediate Shell Longitudinal Weld Seams 101-124A, B, C	Position 1.1	81	56
	Position 2.1	25	15
Lower Shell Longitudinal Weld Seams 101-142A, B, C	Position 1.1	81	56
	Position 2.1	25	15
Intermediate to Lower Shell Circ. Weld Seam 101-171	Position 1.1	61	36
	Position 2.1	5	-5

Table 5-8
RT_{PTS} Calculations for Vogtle Unit 2 Beltline Region Materials at 36 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	$IRT_{NDT(U)}$[a]	ΔRT_{PTS}[b]	Margin[c]	RT_{PTS}[d]
Intermediate Shell Plate R4-1	Position 1.1	44.0	1.19	10	52.4	34	96
Intermediate Shell Plate R4-2	Position 1.1	37.0	1.19	10	44.0	34	88
Intermediate Shell Plate R4-3	Position 1.1	31.0	1.19	30	36.9	34	101
Lower Shell Plate B8825-1	Position 1.1	37.0	1.19	40	44.0	34	118
Lower Shell Plate R8-1	Position 1.1	44.0	1.19	40	52.4	34	126
Lower Shell Plate B8628-1	Position 1.1	31.0	1.19	50	36.9	34	121
	Position 2.1	12.9	1.19	50	15.4	15.4[e]	81
Inter. Shell Longitudinal Weld Seam 101-124A (0° Azimuth)	Position 1.1	43.3	1.04	-10	45.0	45.0	80
	Position 2.1	16.7	1.04	-10	17.4	17.4[e]	25
Inter. Shell Long. Weld Seams 101-124B, C (120°, 240° Azimuth)	Position 1.1	43.3	1.19	-10	51.5	51.5	93
	Position 2.1	16.7	1.19	-10	19.9	19.9[e]	30
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	43.3	1.19	-30	51.5	51.5	73
	Position 2.1	16.7	1.19	-30	19.9	19.9[e]	10
Lower Shell Long. Weld Seams 101-142B, C (210°, 330° Azimuth)	Position 1.1	43.3	1.19	-10	51.5	51.5	93
	Position 2.1	16.7	1.19	-10	19.9	19.9[e]	30
Lower Shell Long. Weld Seam 101-142A (90° Azimuth)	Position 1.1	43.3	1.04	-10	45.0	45.0	80
	Position 2.1	16.7	1.04	-10	17.4	17.4[e]	25

Notes:
(a) Initial RT_{NDT} values are measured values
(b) $\Delta RT_{PTS} = CF * FF$
(c) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$
(d) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
(e) Data deemed credible per Reference 10.
(f) Neutron Fluence value used for all material is the highest value from Table 5-4 for 36 EFPY.

Table 5-9

RT_{PTS} Calculations for Vogtle Unit 2 Beltline Region Materials at 54 EFPY[f]

Material	RG 1.99 R2 Method	CF (°F)	FF	$IRT_{NDT(U)}$[a]	ΔRT_{PTS}[b]	Margin[c]	RT_{PTS}[d]
Intermediate Shell Plate R4-1	Position 1.1	44.0	1.29	10	56.8	34	101
Intermediate Shell Plate R4-2	Position 1.1	37.0	1.29	10	47.7	34	92
Intermediate Shell Plate R4-3	Position 1.1	31.0	1.29	30	40.0	34	104
Lower Shell Plate B8825-1	Position 1.1	37.0	1.29	40	47.7	34	122
Lower Shell Plate R8-1	Position 1.1	44.0	1.29	40	56.8	34	131
Lower Shell Plate B8628-1	Position 1.1	31.0	1.29	50	40.0	34	124
	Position 2.1	12.9	1.29	50	16.6	16.6[e]	83
Inter. Shell Longitudinal Weld Seam 101-124A (0° Azimuth)	Position 1.1	43.3	1.15	-10	49.8	49.8	90
	Position 2.1	16.7	1.15	-10	19.2	19.2[e]	28
Inter. Shell Long. Weld Seams 101-124B, C (120°, 240° Azimuth)	Position 1.1	43.3	1.29	-10	55.9	55.9	102
	Position 2.1	16.7	1.29	-10	21.5	21.5[e]	33
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	43.3	1.29	-30	55.9	55.9	82
	Position 2.1	16.7	1.29	-30	21.5	21.5[e]	13
Lower Shell Long. Weld Seams 101-142B, C (210°, 330° Azimuth)	Position 1.1	43.3	1.29	-10	55.9	55.9	102
	Position 2.1	16.7	1.29	-10	21.5	21.5[e]	33
Lower Shell Long. Weld Seam 101-142A (90° Azimuth)	Position 1.1	43.3	1.15	-10	49.8	49.8	90
	Position 2.1	16.7	1.15	-10	19.2	19.2[e]	28

Notes:

(a) Initial RT_{NDT} values are measured values

(b) $\Delta RT_{PTS} = CF * FF$

(b) $M = 2 *(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

(c) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin (°F)}$

(d) Data deemed credible per Reference 10.

(e) Neutron Fluence value used for all material is the highest value from Table 5-4 for 54 EFPY.

6.0 References

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2. WCAP-16142-P, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Vogtle Units 1 and 2", Warren Bamford, et. al., February 2004.

3. Code of Federal Regulations, 10CFR50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, U.S. Nuclear Regulatory Commission, Washington, D.C.

4. WCAP-11381, *Georgia Power Company Alvin W. Vogtle Unit No. 2 Reactor Vessel Radiation Surveillance Program*, L. R. Singer, April 1986.

5. ASTM E23 *Standard Test Method Notched Bar Impact Testing of Metallic Materials*, in ASTM Standards, American Society for Testing and Materials, Philadelphia, PA.

6. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

7. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, *Fracture Toughness Criteria for Protection Against Failure.*

8. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.*

9. Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*, U.S. Nuclear Regulatory Commission, May 1988.

10. WCAP-15159, *Analysis of Capsule X From the Southern Nuclear Vogtle Electric Generating Plant Unit 2 Reactor Vessel Radiation Surveillance Program*, T.J. Laubham, et. al., Dated March 1999. [*Note that the Testing/Analysis reports for surveillance capsules U and Y from Vogtle Unit 2 were documented under WCAP-13007 and WCAP-14532, respectively.*]

11. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1999.