VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 20, 2006

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 06-335
NLOS/GDM R1
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE REQUEST FOR REINSTATEMENT OF PREVIOUS REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS, LTOPS SETPOINT, AND LTOPS ENABLE TEMPERATURE BASIS

Virginia Electric and Power Company (Dominion) submitted a proposed Technical Specifications (TS) change request in a letter dated December 17, 2004 (Serial No. 04-755) to revise the Reactor Coolant System (RCS) Pressure/Temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoint, and LTOPS enable temperature (T_{enable}) basis for cumulative core burnups up to 47.6 EFPY and 48.1 EFPY (corresponding to the period of the renewed licenses) for Surry Power Station Units 1 and 2, respectively. The NRC approved the proposed TS change in Surry License Amendments 245/244 dated January 3, 2006. However, subsequent to NRC issue of the license amendments, Dominion discovered an error in the technical basis that supported the proposed TS change and was therefore relied upon by the NRC in their safety evaluation. Specifically, the revised RCS P/T limit curves did not consider the most limiting material property data at the ¾ thickness (3/4-T) reactor vessel thickness location [i.e., the 3/4-T reference nil-ductility temperature (RT_{NDT}) value at the corresponding location within the reactor vessel wall.] Therefore, the RCS P/T operating limits, LTOPS setpoint and LTOPS T_{enable} basis provided in the previous TS change request and approved by the NRC in License Amendments 245/244 are not valid for use throughout the renewed license periods of Surry Units 1 and 2.

Although the NRC approved the revised RCS P/T operating limits and LTOPS setpoint on January 3, 2006, Dominion was granted 180 days from the date of issuance for implementation (i.e., July 2, 2006). As a result, the operating limits/setpoint approved in License Amendments 245/244 have not, and will not, be implemented at Surry Power Station. The RCS P/T operating limits and LTOPS setpoint currently in Surry TS (i.e., prior to issuance of License Amendments 245/244) remain valid until 2011 and 2012 for Surry Units 1 and 2; therefore, there is no safety issue involved in their continued use. Dominion proposes to restore the RCS P/T operating limits, LTOPS setpoint, and LTOPS T_{enable} basis to those in place prior to the approval of License Amendments

245/244, (i.e., as previously approved by the NRC in License Amendments 207/207 for Surry Units 1 and 2, respectively, on December 28, 1995.)

Therefore, pursuant to 10 CFR 50.90, Dominion requests amendments, in the form of changes to the Technical Specifications (TS) to Facility Operating Licenses Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The proposed TS change restores the RCS P/T operating limits, LTOPS setpoint, and LTOPS T_{enable} basis to those that were in place prior to the approval of License Amendments 245/244.

A separate TS change request will be submitted at a later date to provide revised RCS P/T operating limits, LTOPS setpoint, and LTOPS T_{enable} basis that will be effective through the end of the Surry Units 1 and 2 license renewal periods.

A discussion of the proposed TS change is provided in Attachment 1. The marked-up and proposed TS pages reflecting the proposed change are provided in Attachments 2 and 3, respectively.

We have evaluated the proposed TS change and have determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. We have also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure will occur. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9), and, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The bases for these two determinations are provided in Attachment 1.

The proposed TS change has been reviewed and approved by the Station Nuclear Safety and Operating Committee. Dominion requests NRC approval of the proposed TS change by June 23, 2006 to facilitate implementation prior to the July 2, 2006 implementation date (i.e., 180 days after approval) of License Amendments 245/244.

If you have any further questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Very truly yours,

E. S. Grecheck
Vice President – Nuclear Support Services

Attachments

Commitments made in this letter:

1. A separate TS change request will be submitted at a later date to provide revised RCS P/T operating limits, LTOPS setpoint, and LTOPS T_{enable} basis that will be effective through the end of the Surry Units 1 and 2 license renewal periods.

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is the Vice President – Nuclear Support Services of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the forgoing document in behalf of that Company, and that the statements in the document are true and correct to the best of his knowledge, information, and belief.

Acknowledged before me this 20TH day of April, 2006.

My Commission Expires: May 31, 2006

Vicki L. Hull
Notary Public

(SEAL)

ATTACHMENT 1

Discussion of Change

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

DISCUSSION OF CHANGE

1.0 Description

Virginia Electric and Power Company (Dominion) proposes a change to the Surry Power Station Units 1 and 2 Technical Specifications (TS) pursuant to 10CFR50.90. The proposed change is requested to restore the Reactor Coolant System (RCS) Pressure/Temperature (P/T) operating limits and Low Temperature Overpressure Protection System (LTOPS) setpoint that were recently approved (but not implemented) in License Amendments 245/244 for Surry Units 1 and 2, respectively, to the previous limits/setpoint that were contained in the TS prior to NRC approval of the license amendments. The associated TS Basis change will also be restored to its previous condition.

The proposed change has been reviewed, and it has been determined that no significant hazards consideration exists as defined in 10 CFR 50.92. In addition, it has been determined that the change qualifies for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9); therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed TS change.

2.0 Proposed Change

The following specific changes to the Surry Units 1 and 2 TS are proposed:

- TS 3.1.B - Replace Figures 3.1-1 and 3.1-2 and the cumulative core burnup limits with the Figures 3.1-1 and 3.1-2, Reactor Coolant System Heatup and Cooldown Limitations previously approved by the NRC in License Amendments 207/207 for Surry Units 1 and 2, respectively (Reference 8.1). The proposed curves will be valid for both Surry Units 1 and 2. Associated TS Basis changes are also being implemented to return to the previous TS Basis text.

- TS 3.1.G.1.c(4) - This specification has been revised to reflect the proposed maximum allowable LTOPS setpoint value of 390 psig as previously approved by the NRC in License Amendments 207/207 for Surry Units 1 and 2, respectively.

3.0 Background

Dominion submitted a proposed TS change request to the NRC on December 17, 2004 (Reference 8.2) to revise the Reactor Coolant System (RCS) Pressure/Temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoint and the associated TS Basis for cumulative core burnups up to 47.6 EFPY and 48.1 EFPY (corresponding to the period of the renewed licenses) for Surry Units 1 and 2, respectively. The NRC approved the proposed TS change in Surry License Amendments 245/244 (Reference 8.3). However, subsequent to NRC issue of the license amendments, a discrepancy in the technical basis that supported the proposed TS change was identified. The NRC also relied upon this discrepant information in their safety evaluation supporting the Surry license amendments. Although the revised RCS P/T operating limits and LTOPS setpoint were approved by the NRC for implementation, Dominion was granted 180 days from the date of issuance for implementation (i.e., July 2, 2006). As a result, the operating limits and setpoint approved in License Amendments 245/244 have not, and will not, be implemented at Surry Power Station. Consequently, a TS change is necessary to restore the RCS P/T operating limits and LTOPS setpoint that were recently approved (but not implemented) in License Amendments 245/244 for Surry Units 1 and 2, respectively, to the limits/setpoint that were previously approved by the NRC in Surry License Amendments 207/207 and are currently included in the TS since License Amendments 245/244 have yet to be implemented. The associated TS Basis discussion will also be restored to its previous condition.

4.0 Technical Analysis

The revised RCS P/T operating limits submitted in the proposed TS change request are based on Westinghouse WCAP-15130, "Surry Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," Revision 1, April 2001. Bounding reactor vessel fluence values were used corresponding to 47.6 Effective Full Power Years (EFPY) for Unit 1 and 48.1 EFPY for Unit 2. This WCAP determined RCS P/T heatup and cooldown limits by evaluating the following ¼ and ¾ thickness (1/4T and 3/4T, respectively) reference nil-ductility temperature (RT_{NDT}) values at the corresponding locations within the reactor vessel wall:

1/4-T RT_{NDT} = 238.2 °F (based on Unit 1 weld material SA-1526)
3/4-T RT_{NDT} = 183.9 °F (based on Unit 1 weld material SA-1585)

These RT_{NDT} values were provided to Westinghouse based on a 1998 Dominion calculation. WCAP-15130, Rev. 1, was also cited by Dominion in support of the License Renewal application for Surry (Reference 8.4). In a letter dated November 19, 1999 (Reference 8.5), Dominion submitted revised material property data for Surry Unit 1 weld materials SA-1585 and SA-1650 based on testing of Arkansas Nuclear One (ANO)-1 capsule W-1, which contained the same weld wire heat as Surry weld materials SA-1585 and SA-1650. The November 1999 submittal revised the following material properties for Surry weld materials SA-1585 and SA-1650:

Chemistry factor decreased from 138.0 °F to 131.4 °F
Margin term increased from 48.3 °F to 68.5 °F

Use of these material properties with reactor vessel fluence estimates corresponding to 60-year operation, as provided in Reference 8.4, result in a 3/4-T RT_{NDT} = 197.3 °F. This exceeds the value used in WCAP-15130, Rev. 1. In a letter dated March 27, 2003 (Reference 8.6), Dominion submitted revised material property data for Surry Unit 2 weld material R3008 based on testing of Surry Unit 2 surveillance capsule Y. The March 2003 submittal revised the following material property for Surry Unit 2 weld material R3008:

Chemistry factor increased from 125.2 °F to 132.4 °F

Use of these material properties with reactor vessel fluence estimates corresponding to 60-year operation, as provided in Reference 8.4, result in a 3/4-T RT_{NDT} = 188.4 °F. This also exceeds the value used in WCAP-15130, Rev. 1. Therefore, the appropriate material properties (i.e., chemistry factors and margin terms), including evaluation of surveillance capsule test results for ANO-1 Capsule W-1 and Surry-2 Capsule Y, were not considered when the design work supporting the TS change request (Reference 8.2) was performed in 2004. This error invalidated the basis for the operating limits provided in the TS change request and approved in associated License Amendments 245/244 for Surry Units 1 and 2.

As noted above, Surry Units 1 and 2 License Amendments 245/244 have not been implemented. Dominion previously submitted the proposed RCS P/T operating limits and LTOPS setpoint contained in this proposed TS change request in a letter dated June 8, 1995 (Reference 8.7), and the NRC reviewed and approved the proposed TS change in License Amendments 207/207 for Surry Units 1 and 2, respectively, on December 28, 1995 (Reference 8.1). The RCS P/T limits and LTOPS setpoint approved in TS Amendments 207/207 were based on a 1/4-T RT_{NDT} value of 228.4 °F, and a 3/4-T RT_{NDT} value of 189.5 °F for Unit 1 Intermediate to Lower Shell Circumferential welds SA-1585 and SA-1650. These limiting Unit 1 RT_{NDT} values were based on peak reactor vessel inner surface fast neutron fluence (E>1 MeV) of 3.96 E19 n/cm^2 and a chemistry factor of 149.2 °F. The RCS P/T limits and LTOPS setpoint approved in TS Amendments 207/207 are valid to 28.8 EFPY for Surry Unit 1 and 29.4 EFPY for Surry Unit 2, which corresponded to the end of the original 40-year license periods at the time the P/T limits were developed. In Reference 8.4, Dominion stated that cumulative core burnups at the end of the original 40-year license period were 29.6 EFPY for Surry Unit 1 and 30.1 EFPY for Surry Unit 2. In Reference 8.6, Dominion provided the most recent update to the NRC's Reactor Vessel Integrity Database for Surry, utilizing 40-year fluence estimates corresponding to 29.6 EFPY for Surry Unit 1, and 30.1 EFPY for Surry Unit 2. Reference 8.6 reported a limiting 1/4-T RT_{NDT} value of 218.9 °F for Unit 1 Intermediate to Lower Shell Circumferential welds SA-1585 and SA-1650, based on a chemistry factor of 131.4 °F. The limiting 3/4-T RT_{NDT} value was 183.1 °F for Unit 1 welds SA-1585 and SA-1650; however, this value was not explicitly reported in Reference 8.6. Therefore, the RCS P/T limits and LTOPS setpoint approved in TS Amendments 207/207 are conservative with respect to the limiting RT_{NDT} values reported in Reference 8.6. Dominion evaluations to date have confirmed that the 1/4-T RT_{NDT} value of 228.4 °F, and the 3/4-T RT_{NDT} value of 189.5 °F remain bounding and conservative for the period of operation corresponding to the original 40-year license period. Consequently, no technical issues exist since the proposed TS change provided herein simply restores the RCS P/T operating limits, LTOPS setpoint, and associated TS Basis that were previously reviewed/approved by the NRC prior to the approval of License Amendments 245/244.

5.0 Regulatory Safety Analysis

Dominion notified the NRC about the TS discrepancy discussed above and entered the plant issue into the Surry Corrective Action System. During subsequent discussions with the NRC, it was determined that Dominion would prepare and submit a follow-up TS change request that would supercede the TS change approved in License Amendments 245/244 and restore the RCS P/T operating limits and LTOPS setpoint to the limits and setpoint that were included in the TS prior to the issuance of License Amendments 245/244. This approach is consistent with the manner of resolution used for similar TS implementation issues that occurred at Susquehanna Steam Electric Station, Unit 2 (References 8.8 through 8.10), and Ft. Calhoun Station Unit No. 1 (References 8.11 through 8.15).

The RCS P/T operating limits currently in Surry TS (i.e., prior to implementation of License Amendments 245/244) remain valid until 2011 and 2012 for Surry Units 1 and 2; therefore, there is no safety issue involved. A separate TS change request will be prepared and submitted to the NRC at a later date, but prior to expiration of the current limits/setpoint, to provide RCS P/T operating limits and an LTOPS setpoint that are valid through the end of the Surry Units 1 and 2 extended license periods.

6.0 Significant Hazards Consideration Determination

Virginia Electric and Power Company (Dominion) has reviewed the requirements of 10 CFR 50.92, relative to the proposed change to the Surry Units 1 and 2 Technical Specifications (TS), and determined that a Significant Hazards Consideration is not involved. The proposed change to the Surry Units 1 and 2 TS revises the Reactor Coolant System (RCS) pressure/temperature (P/T) limits and Low Temperature Overpressure Protection System (LTOPS) setpoint to restore them to the limits/setpoint that were previously approved by the NRC and that are currently implemented in the plant.

The following discussion is provided to support the conclusion that the proposed change does not create a significant hazards consideration:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not impact the condition or performance of any plant structure, system or component. The proposed change does not affect the initiators of any previously analyzed event or the assumed mitigation of accident or transient events since the plant will be operated in the same manner and within the same operating limits that are currently in place. The proposed change merely restores the RCS P/T limit curves and LTOPS setpoint that were approved by the NRC prior to the issue of License Amendments 245/244, and which are currently in effect. As a result, the proposed change to the Surry TS does not involve any increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated since neither accident probabilities nor consequences are being affected by this proposed change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any changes in station operation or physical modifications to the plant. In addition, no changes are being made in the methods used to respond to plant transients that have been previously analyzed. No changes are being made to plant parameters within which the plant is normally operated or in the setpoints, which initiate protective or mitigative actions, since the plant will be operated in the same manner and within the same operating limits that are currently in place. Since plant operation will not be affected by this change, no new failure modes are being introduced. Therefore, the proposed change to the Surry TS does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

3. Does the change involve a significant reduction in the margin of safety?

   The return to the previously approved RCS P/T operating limit curves and LTOPS setpoint does not involve a significant reduction in the margin of safety. The proposed change does not impact station operation or any plant structure, system or component that is relied upon for accident mitigation. Furthermore, the margin of safety assumed in the plant safety analysis is not affected in any way by the proposed change since the plant will be operated in the same manner and within the same operating limits and setpoints that are currently in place. Therefore, the proposed change to the Surry Technical Specifications does not involve any reduction in a margin of safety.

7.0 Environmental Consideration

The proposed change to the Surry Units 1 and 2 Technical Specifications restores the Reactor Coolant System (RCS) pressure/temperature (P/T) limit curves, LTOPS setpoint, and the TS Basis that were approved by the NRC prior to the issue of License Amendments 245/244 and which are currently in effect. The proposed TS change meets the eligibility criteria for categorical exclusion from an environmental assessment set forth in 10 CFR 51.22(c)(9), as discussed below:

(i) The license condition involves no significant hazards consideration.

As discussed in the evaluation of the Significant Hazards Consideration above, the proposed change to the RCS P/T limit curves, LTOPS setpoint, and associated TS Basis for Surry Power Station will not involve a significant increase in the probability or consequences of an accident previously evaluated. The possibility of a new or different kind of accident from any accident previously evaluated is also not created, and the proposed change does not involve a significant reduction in a margin of safety. Therefore, the proposed change to the RCS P/T limit curves, LTOPS setpoint, and TS Basis meet the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are identical to those currently in effect. No changes to plant systems, structures or components are proposed, and no new operating modes are established. Therefore, the proposed change to the RCS P/T limit curves, LTOPS setpoint, and TS Basis will not significantly change the types, or significantly increase the amounts, of effluents that may be released offsite.

(iii)There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change restores the Surry Units 1 and 2 RCS P/T limit curves, LTOPS setpoint, and TS Basis to those previously approved by the NRC prior to the approval of License Amendments 245/244. The allowable operating pressures and temperatures under the proposed RCS P/T limit curves are identical to those allowed under the existing TS RCS P/T limits. No changes to plant systems, structures or components are proposed, and no new operating modes are established. The proposed change maintains acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions. Therefore, the proposed change will not increase radiation levels compared to the existing TS P/T limits, LTOPS setpoint, and TS Basis as they are identical, and consequently, individual and cumulative occupational exposures are unchanged.

Based on the above, the proposed change does not have a significant effect on the environment, and meets the criteria of 10 CFR 51.22(c)(9). Therefore, the proposed TS change qualifies for a categorical exclusion from a specific environmental review by the Commission, as described in 10 CFR 51.22.

8.0 References

8.1 Letter from the USNRC to Virginia Electric and Power Company dated December 28, 1995 (Serial No. 96-020), "Surry Units 1 and 2 – Issuance of Amendments Re: Surry, Units 1 and 2 Reactor Vessel Heatup and Cooldown Curves (TAC Nos. M92537 and M97538)."

8.2 Letter from Virginia Electric and Power Company to the USNRC dated December 17, 2004 (Serial No. 04-755), "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed Technical Specifications Change Request for Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoint, and LTOPS Enable Temperature with Exemption Request for Alternate Material Properties Basis Per 10 CFR 50.60(b)."

8.3 Letter from the USNRC to Virginia Electric and Power Company dated January 3, 2006 (Serial No. 06-017), "Surry Power Station, Unit Nos. 1 and 2 – Issuance of Amendments on Reactor Coolant System Pressure and Temperature Limits (TAC Nos. MC5509 and MC5510)."

8.4 Letter from Virginia Electric and Power Company to the USNRC dated October 15, 2002 (Serial No. 02-601), "Virginia Electric and Power Company (Dominion), Surry and North Anna Power Stations Units 1 and 2, Response to Request for Supplemental Information, License Renewal Applications."

8.5 Letter from Virginia Electric and Power Company to the USNRC dated November 19, 1999 (Serial No. 99-452A), "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Evaluation of Reactor Vessel Materials Surveillance Data."

8.6 Letter from Virginia Electric and Power Company to the USNRC dated March 27, 2003 (Serial No. 03-195), "Virginia Electric and Power Company, Surry Power Station Unit 2, Evaluation of Surry Unit 2 Capsule Y Data."

8.7 Letter from Virginia Electric and Power Company to the USNRC dated June 8, 1995 (Serial No. 95-197), "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Request for Exemption – ASME Code Case N-514, Proposed Technical Specifications Change, Revised Pressure/Temperature Limits and LTOPS Setpoint."

8.8 Letter from the USNRC to Pennsylvania Power & Light Company dated June 5, 1987, "Technical Specification Revisions Regarding Drywell Cooling System (TAC No. 61098)." (ADAMS Accession #ML010180253)

8.9 Letter from Pennsylvania Power & Light Company to the USNRC dated April 8, 1988, "Susquehanna Steam Electric Station, Proposed Amendment 62 to License No. NPF-22: Exigent Request Due to Non-Installation of Drywell Cooling Mods."

8.10 Letter from the USNRC to Pennsylvania Power & Light Company dated May 17, 1988, "Subject: Technical Specification Changes Reflecting Cancellation of Drywell Fan Modifications (TAC No. 67901) Re: Susquehanna Steam Electric Station, Unit 2." (ADAMS Accession #ML010160144)

8.11 Letter from the NRC to Omaha Public Power District dated January 16, 2004, "Ft. Calhoun Station, Unit No. 1 – Issuance of Amendment (TAC No. MC0029)." (ADAMS Accession #ML040200757)

8.12 Letter from Omaha Public Power District to the USNRC dated February 6, 2004 (Serial No. LIC-04-0017), "Subject: Ft. Calhoun Station Unit No. 1 License Amendment Request, 'Extension of Implementation Period for License Amendment 224.'" (ADAMS Accession #ML040420160)

8.13 Letter from the USNRC to Omaha Public Power District dated February 13, 2004, "Ft. Calhoun Station, Unit No. 1 – Issuance of Amendment (TAC NO. MC1949)." (ADAMS Accession #ML040490383)

8.14 Letter from Omaha Public Power District to the USNRC dated May 5, 2004 (Serial No. LIC-04-0061), "Subject: Ft. Calhoun Station Unit No. 1 - Notification of Technical Issues Associated with License Amendment No. 225 (TAC No. MC1949 and TAC No. MC0029.)" (ADAMS Accession #ML041280261)

8.15 Letter from Omaha Public Power District to the USNRC dated May 7, 2004 (Serial No. LIC-04-0062), "Subject: Ft. Calhoun Station Unit No. 1 Exigent License Amendment Request, 'Restoration of Previous Licensed Rated Power Limit.'" (ADAMS Accession #ML041320200)

ATTACHMENT 2

Marked-up Technical Specifications Pages

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 28.8 ~~47.6~~ Effective Full Power Years (EFPY) and 29.4 ~~48.1~~ EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (228.4 ~~238.2~~°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 28.8 ~~47.6~~ EFPY and 29.4 ~~48.1~~ EFPY for Units 1 and 2, respectively (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 28.8 ~~47.6~~ EFPY or 29.4 ~~48.1~~ EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, $K_{\cancel{Ic}\,IR}$, for the metal temperature at that time. $K_{\cancel{Ic}\,IR}$ is obtained from the reference fracture toughness curve, defined in ~~Section XI~~ Appendix G to the ASME Code. The $K_{\cancel{Ic}\,IR}$ curve is given by the equation:

$$\cancel{K_{IC} = 33.2 + 20.734 \exp[0.02(T - RT_{NDT})]} \qquad (1)$$

where $K_{\cancel{Ic}\,IR}$ is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \le K_{\cancel{Ic}\,IR} \qquad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

$$K_{IR} = 26.78 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)]$$

K_{It} is the stress intensity factor caused by the thermal gradients

$K_{\cancel{IC}}$ IR is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, $K_{\cancel{IC}}$ IR is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of ~~47.6~~ 28.8 EFPY and ~~48.1~~ 29.4 EFPY for Units 1 and 2, respectively. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of $\leq$ ~~395~~ 390 psig and verify each PORV block valve is open at least once per 72 hours,

or

(5) The RCS shall be vented through one open PORV or an equivalent size opening as specified below:

(a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or

(b) with the RCS vented through a locked open vent path verify the path is open at least once per 31 days.

2. The requirements of Specification 3.1.G.1.c.(4) may be modified as follows:

a One PORV may be inoperable in INTERMEDIATE SHUTDOWN with the RCS average temperature > 200°F but < 350°F for a period not to exceed 7 days. If the inoperable PORV is not restored to OPERABLE status within 7 days, then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within the next 8 hours.

b One PORV may be inoperable in COLD SHUTDOWN or REFUELING SHUTDOWN with the reactor vessel head bolted for a period not to exceed 24 hours. If the inoperable PORV is not restored to OPERABLE status within 24 hours then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within 8 hours.

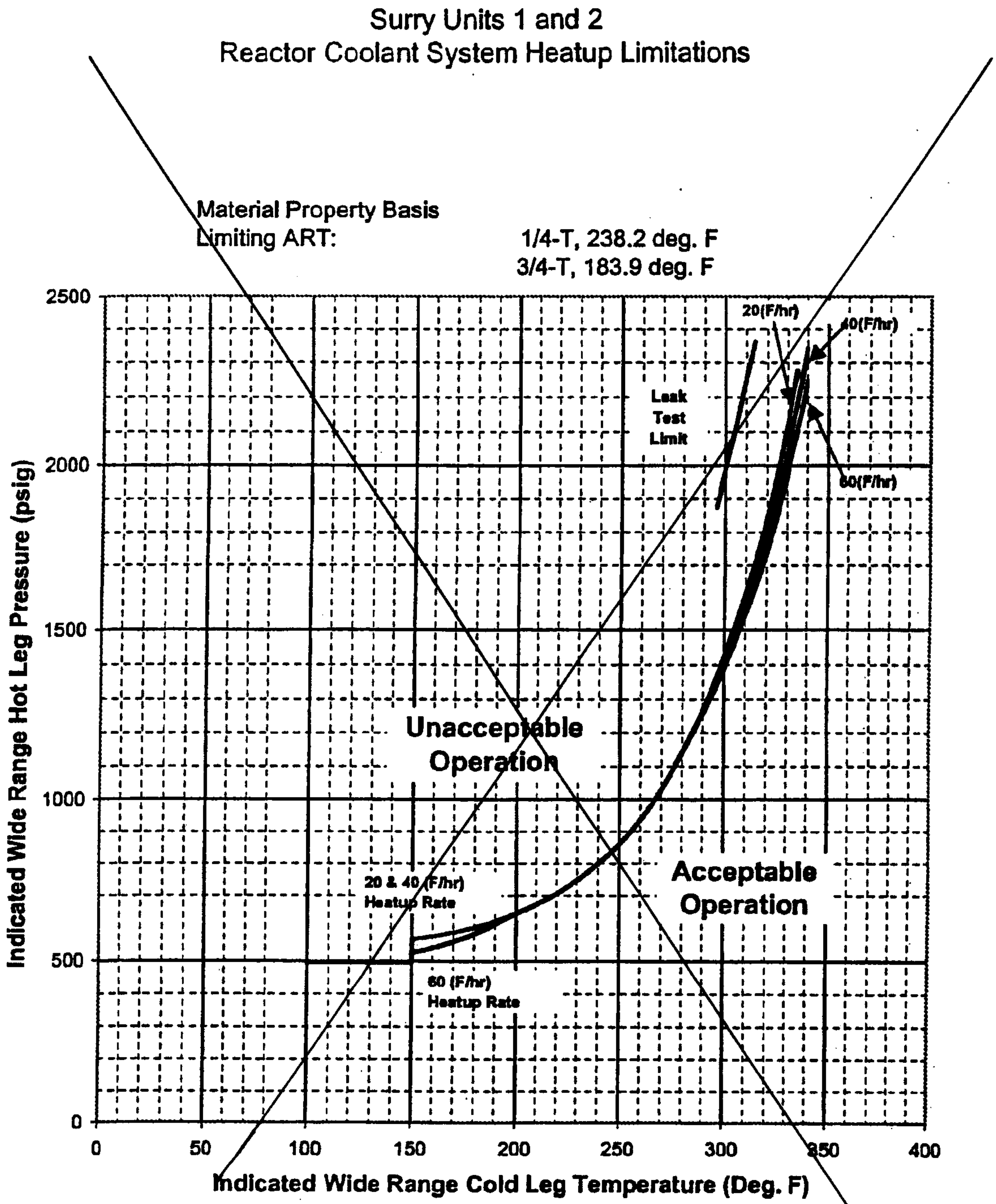
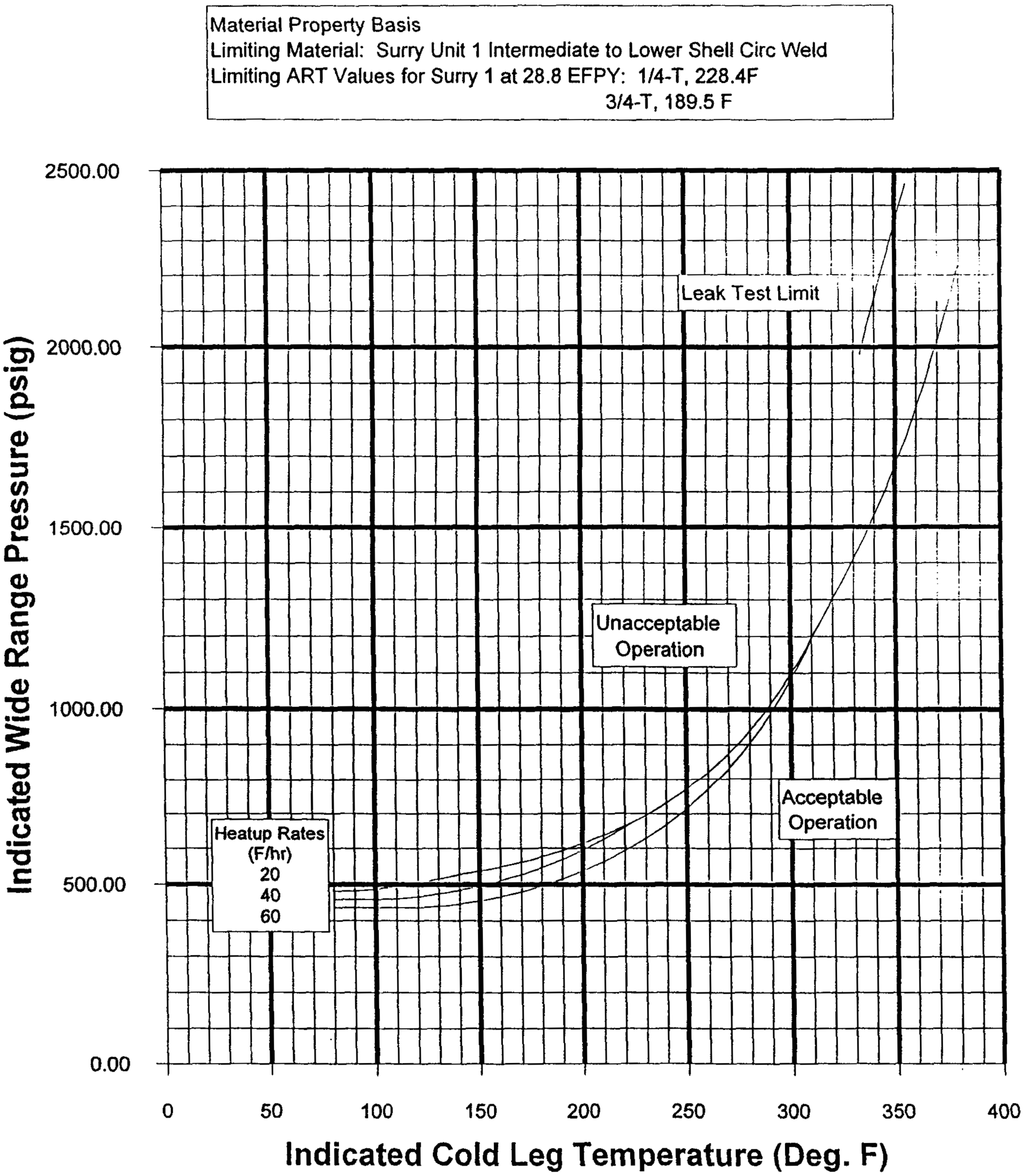


Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

Amendment Nos. ~~245/244~~

Replace with attached Figure 3.1-1

Surry Units 1 and 2
Reactor Coolant System Heatup Limitations



Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

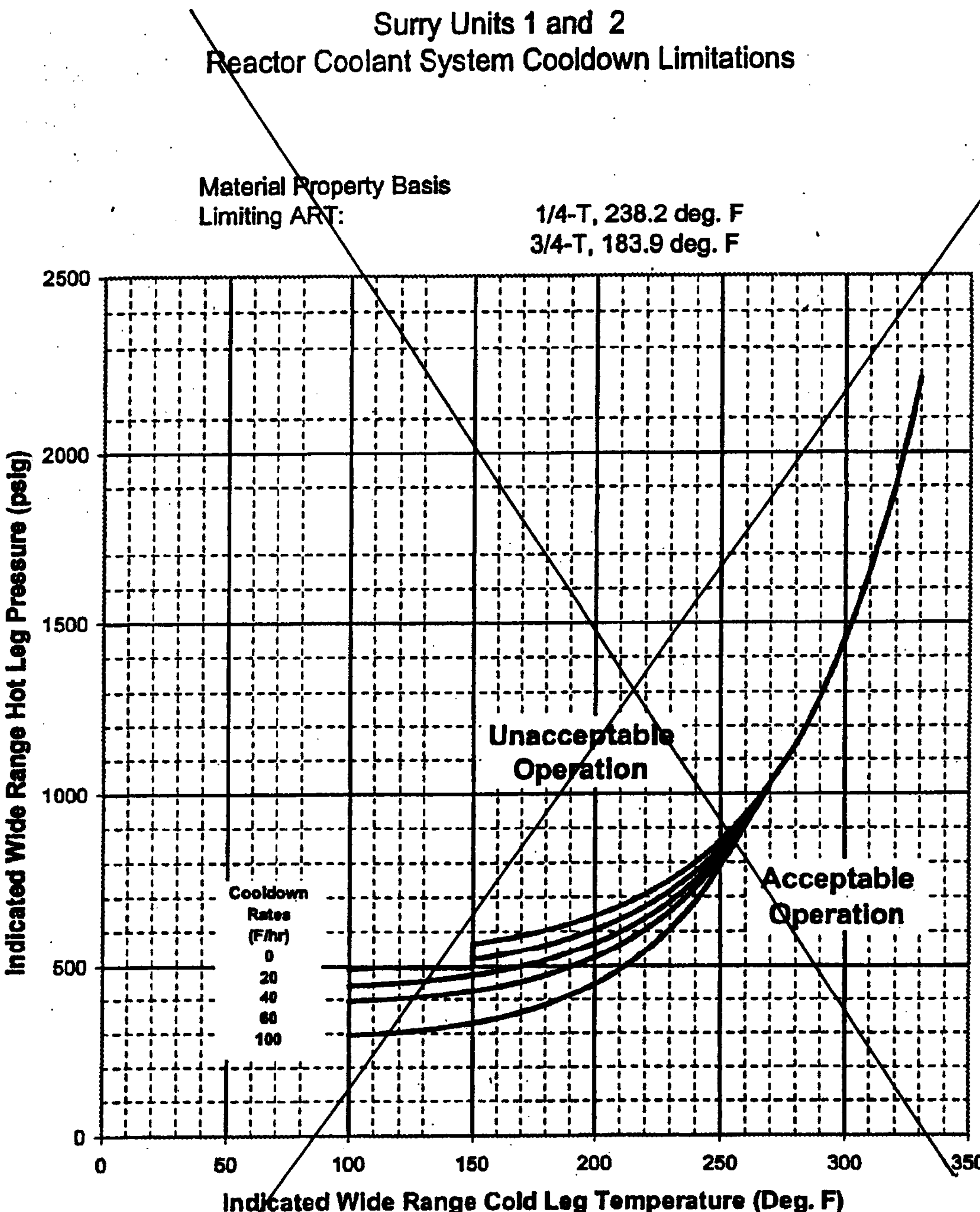
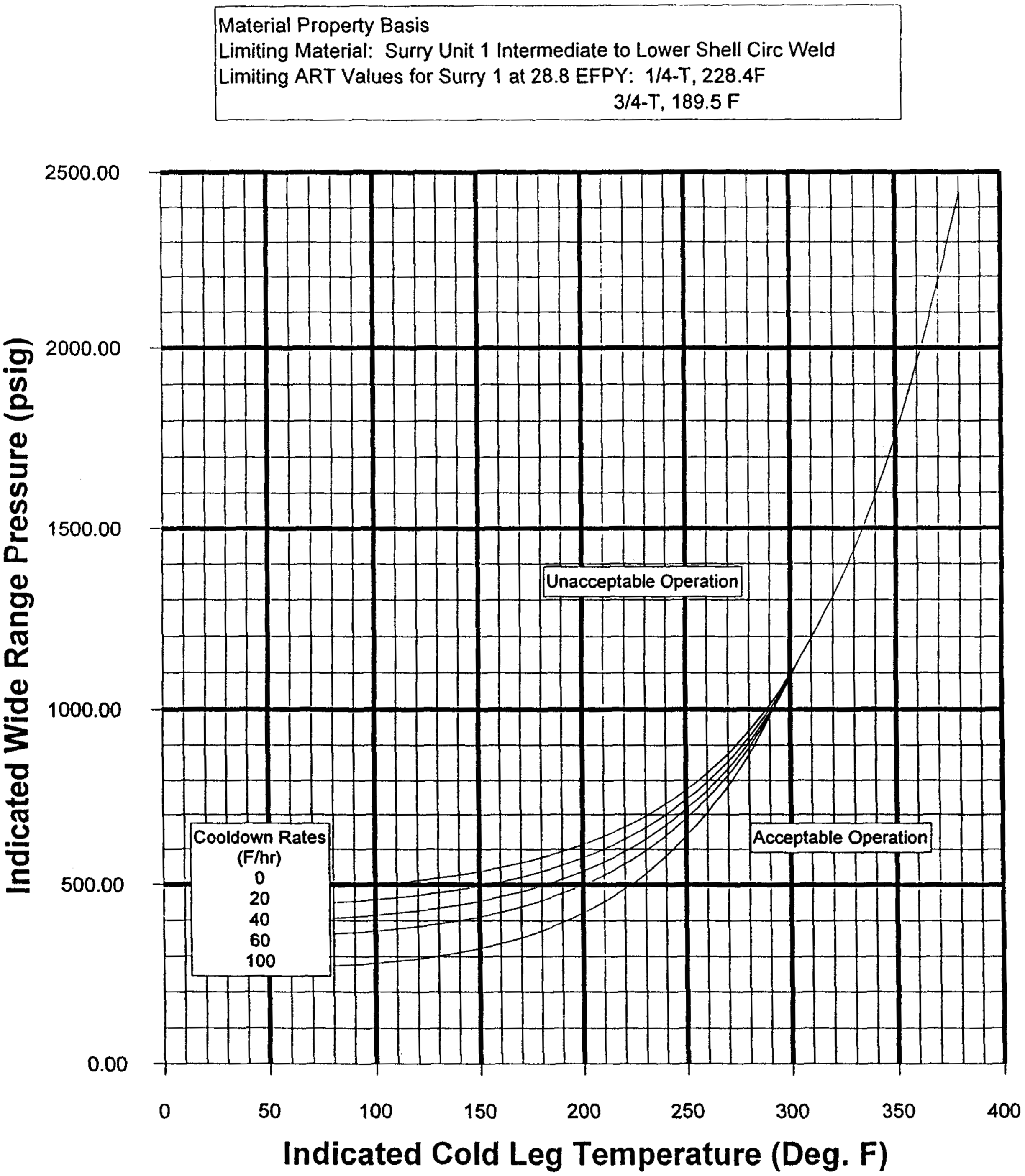


Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

Amendment Nos. ~~245/244~~

Replace with attached Figure 3.1-2

Surry Units 1 and 2
Reactor Coolant System Cooldown Limitations



Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

ATTACHMENT 3

<u>Proposed Technical Specifications Pages</u>

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 28.8 Effective Full Power Years (EFPY) and 29.4 EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (228.4°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 28.8 EFPY or 29.4 EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \quad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of $\leq$ 390 psig and verify each PORV block valve is open at least once per 72 hours,

or

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(a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or

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b One PORV may be inoperable in COLD SHUTDOWN or REFUELING SHUTDOWN with the reactor vessel head bolted for a period not to exceed 24 hours. If the inoperable PORV is not restored to OPERABLE status within 24 hours then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within 8 hours.

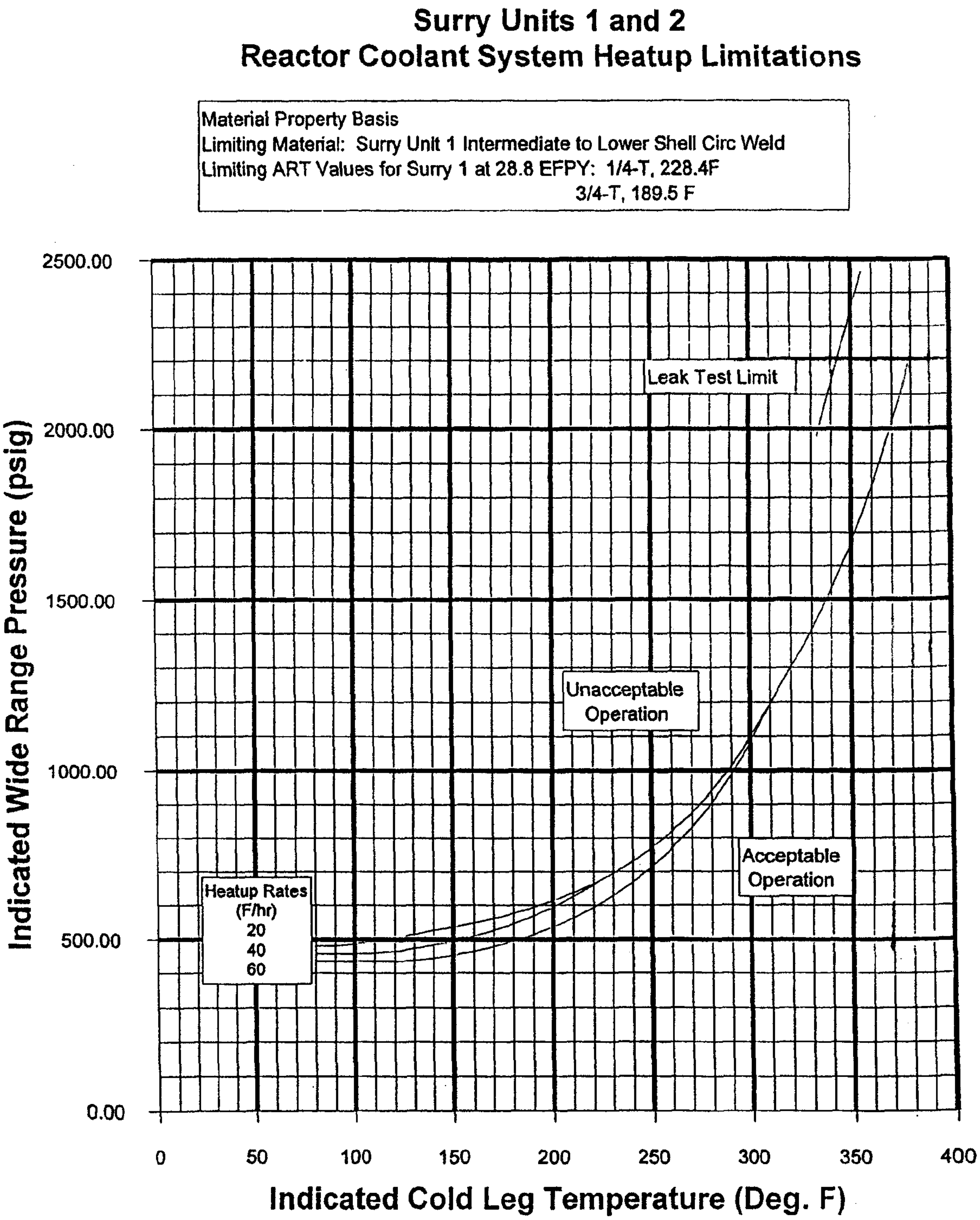

Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for the first 28.8 EFPY for Surry Unit 1 and the first 29.4 EFPY for Surry Unit 2

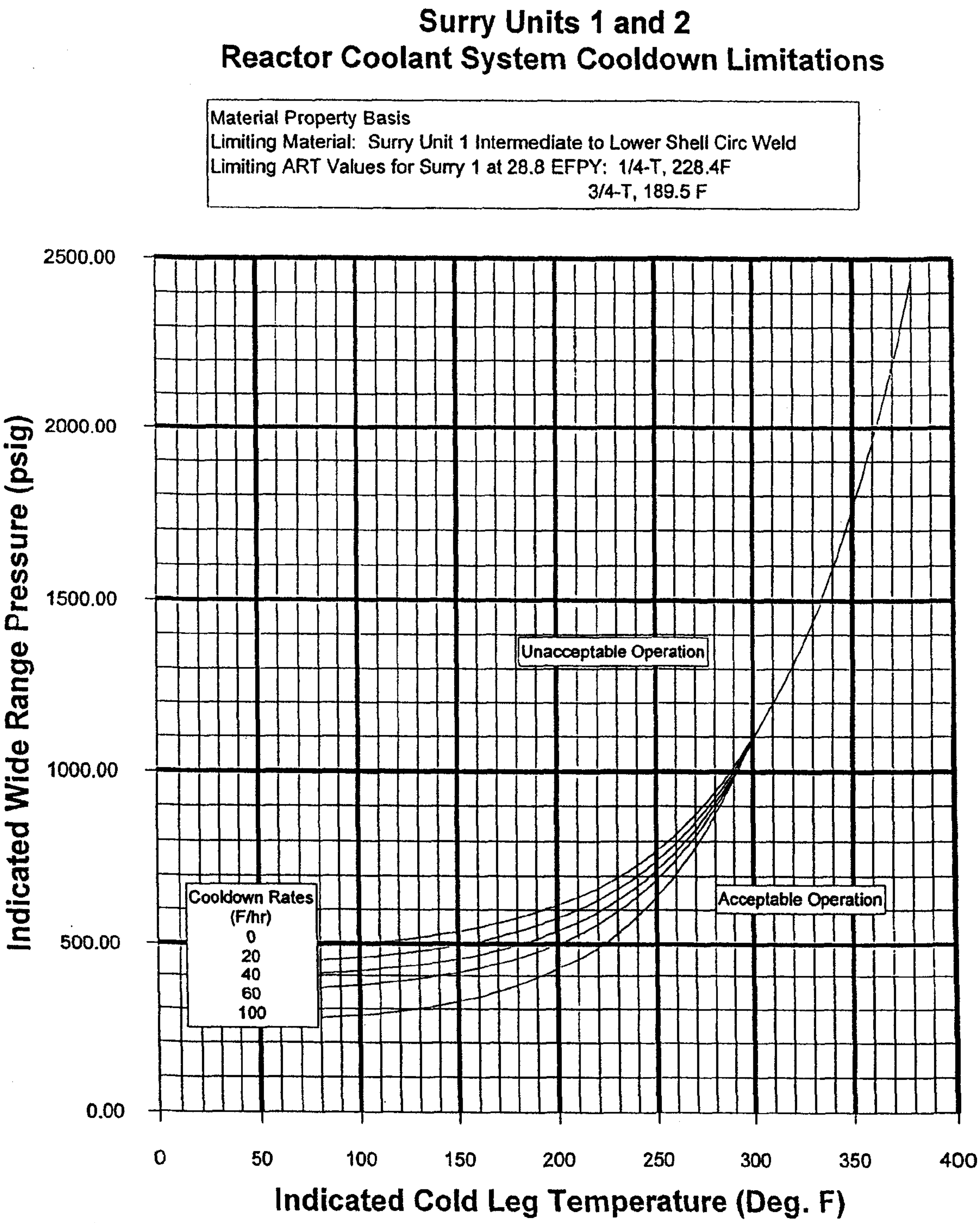

Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the first 28.8 EFPY for Surry Unit 1 and the first 29.4 EFPY for Surry Unit 2