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January 25, 2002
L-02-004

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
License Amendment Request No. 297

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above license in the form of changes to the technical specifications. The proposed changes will change the Beaver Valley Power Station (BVPS) Unit 1 heatup/cooldown curves, the power-operated relief valve setpoint and the overpressure protection system enable temperature based upon use of the Master Curve Methodology. The heatup/cooldown curves are also changed to eliminate the effect of the reactor vessel flange. This License Amendment Request also contains a request for exemption from certain requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," 10 CFR 50.54, "Conditions of Licenses," and 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

This License Amendment Request contains five Attachments and five Enclosures. The proposed technical specification changes are presented in Attachment A. The safety analysis and no significant hazard evaluation is presented in Attachment B. The proposed pressure and temperature limits report (PTLR) changes, which are submitted for NRC review and approval, are provided in Attachment C. The changes proposed to the Technical Specification Bases are provided in Attachment D. The proposed Technical Specification Bases changes do not require NRC approval. The BVPS Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The bases changes are provided for information only. The applicable exemption request is included as Attachment E.

The enclosures are tabulated below.

Enclosure	Document Title
1	Proprietary Information Notice, Copyright Notice and a Westinghouse application for withholding proprietary information.
2	WCAP-15624, "Master Curve Fracture Toughness Application for BVPS-1," dated November 2001.
3	WCAP-15618, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Master Curve Applications," dated November 2001.
4	Proprietary (Class 2) version of Westinghouse Report, I&CE/PEOD(01)-206, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Master Curve," Revision 1, August 2001.
5	Non-proprietary (Class 3) version of Westinghouse Report, I&CE/PEOD(01)-206, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Master Curve," Revision 1, August 2001.

As Enclosure 4 contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, the owner of the information. This affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of Enclosure 4 report or the supporting Westinghouse Affidavit should reference CAW-02-1506 and should be addressed to Mr. H. A. Sepp, Manager Regulatory and Licensing Engineering, Westinghouse Electric Company, LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

FENOC letter L-01-027, dated March 7, 2001 transmitted WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program." This letter requested NRC approval of a change to the Capsule X withdrawal schedule in accordance with 10 CFR 50 Appendix H, Section III.B.3. Subsequently, FENOC verbally requested the NRC to defer approval of this withdrawal schedule, since the schedule would be further changed by the Master Curve Methodology. WCAP-15624

contains the Capsule X withdrawal schedule applicable to the Master Curve Methodology which supersedes the withdrawal schedule submitted by L-01-027. FENOC, therefore, requests NRC approval of the Capsule X withdrawal schedule shown in Table 5-1 of WCAP-15624 in accordance with 10 CFR 50 Appendix H, Section III.B.3. Following NRC approval of WCAP-15624, Table 4.5-3 of the BVPS Unit 1 Updated Final Safety Analysis Report will be revised to reflect the Capsule X withdrawal schedule. This approval is requested by January 30, 2003.

This License Amendment Request makes reference to a reference temperature pressurized thermal shock (RT_{PTS}) value for end-of-life extension (EOLE) and provides projected vessel material properties for EOLE. This information is provided for information only. NRC approval of the EOLE values, or operation beyond the current license period, is not being requested by this submittal.

This change has been reviewed by the Beaver Valley review committees. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis and no significant hazard evaluation. NRC approval of License Amendment Request No. 297 is requested by January 30, 2003. An implementation period of up to 60 days is requested following the effective date of this amendment.

If there are any questions concerning this matter, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Corrective Actions at 724-682-5284.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 25, 2002.

Sincerely,

Lew Myers

Lew W. Myers

Attachments
Enclosures

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

ATTACHMENT A

Beaver Valley Power Station, Unit No. 1
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The following is a list of the affected pages:

Affected Page	Pending LAR
6-20	295

CORE OPERATING LIMITS REPORT (Continued)

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System" Revision 0, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

6.9.6 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. Reactor Coolant System pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, hydrostatic testing, Overpressure Protection System (OPPS) enable temperature, and Power Operated Relief Valve (PORV) lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Specification 3.4.9.1, "Reactor Coolant System Pressure/Temperature Limits", and

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PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Continued)

2. Specification 3.4.9.3, "Reactor Coolant System Overpressure Protection Systems".

b. 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. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with WCAP-14040-NP-A, Rev. 2, and

2. the OPPS limits, i.e., PORV pressure relief setpoint and OPPS enable temperature, were developed in accordance with WCAP-14040-NP-A, Rev. 2.

The methodology listed in WCAP-14040-NP-A was used with ~~two~~ five exceptions:

a) Use of ASME Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1," May 7, 1999,

b) Use of ASME Code Case N-631, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials Other Than Bolting for Class 1 Vessels, Section III, Division 1," September 24, 1999,

c) Use of ASME Code Case N-641, "Alternate Pressure-Temperature Relationship and Low temperature Overpressure Protection System Requirements," January 17, 2000,

d) Use of WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," October 1999, and

e) Use of ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

~~a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and~~

~~b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".~~

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

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6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

ATTACHMENT B

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 297
MASTER CURVE METHODOLOGY

A. DESCRIPTION OF AMENDMENT REQUEST

In accordance with 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) proposes to amend Operating License DPR-66 for Beaver Valley Power Station (BVPS) Unit 1 to allow a different method for determining the adjusted reference nil-ductility temperature (RT_{NDT}) of the limiting beltline region materials of the reactor vessel. This method, the Master Curve Methodology, as described in Reference 1, WCAP-15624, "Master Curve Fracture Toughness Application for BVPS-1", is used to generate the heatup and cooldown curves, and subsequently the power-operated relief valves (PORVs) setpoint, and overpressure protection system (OPPS) enable temperature. The proposed pressure-temperature (P/T) limits have been prepared using the NRC approved methodology described in Reference 2, WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with five exceptions. They include the use of:

a) ASME Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1," May 7, 1999,

b) ASME Code Case N-631, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials Other Than Bolting for Class 1 Vessels, Section III, Division 1," September 24, 1999,

c) ASME Code Case N-641, "Alternate Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000,

d) WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," October 1999, and

e) ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

Pursuant to 10 CFR 50.12 (a) (2) (iii), FirstEnergy Nuclear Operating Company (FENOC) requests exemption to 10 CFR 50.60 and 10 CFR 50 Appendix G, based on American Society of Mechanical Engineers (ASME) Code Cases N-629, N-631, N-641, and Reference 3, WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants". The exemption request is included as Attachment E. The proposed Unit 1 P/T limits incorporate the results from testing of Capsules W and Y described in Reference 4, WCAP-15571, "Analysis of Capsule Y from FENOC Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program", and Reference 5, WCAP-12005, "Analysis of Capsule W from the Duquesne Light Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program".

The proposed Technical Specification and Pressure and Temperature Limits Report (PTLR) changes, which are submitted for NRC review and approval, are provided in Attachments A and C, respectively. The changes proposed to the Technical Specification Bases are provided in Attachment D. The proposed Technical Specification Bases changes do not require NRC approval. The BVPS Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. They are provided for information only.

The proposed changes to the Technical Specifications, Bases and PTLR have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added.

The following provides a summary of the proposed changes:

1. Technical Specification 6.9.6, which provides the reporting requirements associated with the PTLR, is revised to include methods used to generate the proposed heatup and cooldown curves, and subsequently the PORV setpoint and the OPPS enable temperature.

2. PTLR Section 4.2.1 is revised to be consistent with Technical Specification 6.9.6.

3. PTLR Section 4.2.1.1 is revised by adding a discussion concerning the Master Curve Methodology, stating that the curves do not include the effect of the reactor vessel flange, and changing the applicable ASME Code Case to N-641.

4. PTLR Section 4.2.1.3 is revised to change the OPPS arming and enable temperatures to the values documented in Reference 6, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Master Curve".

5. PTLR Section 4.2.3 is revised to include a discussion of Codes Cases N-629 and N-631.

6. PTLR Section 4.2.4, is changed by replacing the WCAPs appearing in References 2 and 5 with WCAP-15618 and WCAP-15624, respectively.

7. PTLR Figures 4.2-1 and 4.2-2 and Tables 4.2-1 and 4.2-2 are revised to be consistent with the Master Curve Methodology.

8. PTLR Table 4.2-3 is revised to reflect the new OPPS enable temperature and PORV setpoint.

9. PTLR Supplemental Tables are revised to be consistent with the Master Curve Methodology.

10. The Bases associated with Technical Specification 3/4.4.9 is revised by referencing ASME Code Cases N-629, N-631 and N-641.

11. In accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", NRC approval of the change to the capsule removal schedule is requested to incorporate the schedule shown in Table 1 of WCAP-15624. This removal schedule change reflects use of the Master Curve methodology.

Several of the pages affected by this license amendment request contain changes that have been previously submitted. Since approvals of the previously submitted changes are expected prior to the approval of this license amendment request, the pages affected by this request include those previously submitted changes that are germane to this request. The cover page of Attachments A, C, and D lists the pages affected by this license amendment request. The applicable license amendment request (LAR) number identifies the pages being changed by other LARs. The previously submitted LARs are 292 and 295.

LAR 292 was submitted by FENOC letter L-01-087, dated June 29, 2001. The changes germane to this request are changes to the heatup/cooldown curves, the PORV setpoint and the OPPS enable temperature to reflect 22 Effective Full Power Years (EFPY), the methodology of WCAP-14040-NP-A, and the applicability of Code Case N-640. The applicable pages for this request therefore reflect these proposed changes.

LAR 295 was submitted by FENOC letter L-01-135 , dated October 31, 2001. The changes germane to this request are proposed changes consisting of creating a PTLR. The applicable pages for this request therefore reflect these proposed changes.

To meet format requirements, the Technical Specification Index and Bases, and PTLR pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

B. DESIGN BASES

The ability of the reactor vessel to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor vessel is the most critical region of the vessel because it is subjected to significant fast neutron irradiation. Generally, the overall effects of fast neutron irradiation on the mechanical properties of low alloy, ferritic pressure vessel steels, such as SA533 Grade B Class 1, which is the base metal of Beaver Valley Unit 1 reactor pressure vessel, show an increase in hardness and tensile properties and a decrease in ductility and fracture toughness during high-energy irradiation. The Beaver Valley Unit 1 Vessel Radiation Surveillance Program, designed by Westinghouse, is described in Reference 7, WCAP-8457, "Duquesne Light Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program". The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Vessels." It is through this surveillance program that the effects of high-energy irradiation on the reactor vessel material is evaluated.

The purpose of this license amendment request is to change the methodology used to determine the reactor coolant system Pressure/Temperature (P/T) limits so that the ability of the reactor vessel to resist fracture is maintained over the design life of the plant. The applicable design basis is therefore the P/T limit requirements of ASME Boiler and Pressure Vessel Code, Section XI, Appendix G (Reference 8). The new methodology, known as the Master Curve Methodology, is being

proposed as the methodology to be used to evaluate the effects of high-energy irradiation on the reactor vessel material. The results of this evaluation are used to determine the heatup/cooldown curves necessary to provide assurance that Appendix G limits are not exceeded. Once the heatup/cooldown curves are established, overpressure protection setpoints are determined such that assurance is provided that the reactor vessel, and inherently the reactor coolant pressure boundary, maintains the ability to resist brittle fracture.

Overpressure Protection System

Overpressure protection for the reactor coolant system (RCS) is achieved through the use of self-actuated code safety valves located high in the system on the steam space of the pressurizer. These valves have a set pressure based on the RCS design pressure of 2485 psig and are intended to protect the system against transients initiated in the plant when the RCS is operating near its normal temperature. Power operated relief valves (PORVs) are provided in addition to the code safety valves. These relief valves automatically open above the setpoint (2335 psig). To avoid brittle fractures at reactor vessel metal temperatures below the overpressure protection system enable temperature, the allowable system pressure is required to be substantially less than the normal system design pressure of 2485 psig. Therefore, overpressure mitigation provisions for the reactor vessel must be available when the RCS, and hence the reactor vessel, is at a temperature below the overpressure protection system enable temperature. The overpressure protection system (OPPS) limits the RCS pressure at low temperature so that the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the P/T limits.

Normally when the RCS is at a temperature below the enable temperature, the RCS is open to the Residual Heat Removal System (RHRS) for the purposes of removing residual heat from the core. This provides a path for letdown to the purification subsystem and to control the RCS pressure when the plant is operating in a water solid mode. The RHRS, a system of relatively low design pressure, is provided with self-actuated water relief valves to prevent overpressure caused either within the system itself or from transients transmitted from the RCS. The RHRS relief valves will mitigate pressure transients originated in the RCS to maximum pressure values determined by the relief valves set pressure plus a pressure accumulation above the set pressure dependent on the liquid volume magnitude of the transient. During reactor power operation at temperatures above 350°F, the low design pressure RHRS is normally isolated from the high design

pressure RCS by two isolation valves in series. These isolation valves can inadvertently isolate the RHRS from the RCS. This action would negate the ability of the RHRS to mitigate an overpressure condition in the RCS. In this case, the OPPS is utilized to provide overpressure mitigation for the RCS.

Supplemental Surveillance Capsule

FENOC recognizes that the current testing that has been completed to date requires extrapolation to higher fluences and may not be sufficient to properly project a reference temperature pressurized thermal shock (RT_{PTS}) value for end-of-life extension (EOLE) based on the ASTM E 1921-97 (Reference 9) transition temperature, T_o, and the ASME Code defined transition temperature as provided in WCAP-15624. Therefore, to meet the intent of current regulations using the Charpy V-notch based approach, a supplemental surveillance capsule has been fabricated and installed in BVPS Unit 2. This supplemental capsule contains all of the beltline materials in BVPS Unit 1, and the irradiation in BVPS Unit 2 will allow faster accumulation of fluence than that possible in BVPS Unit 1. The testing of the supplemental capsule as part of a revised surveillance program will allow direct measurement of fracture toughness at the fluence corresponding to EOLE, thus eliminating the need to extrapolate using lower fluence data. The requirements for implementation of an integrated surveillance program are delineated in Appendix H to 10 CFR 50, " Reactor Vessel Material Surveillance Program Requirements." Appendix H defines an integrated surveillance program as a reactor vessel material surveillance program where "the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features." BVPS Unit 1 is in compliance with those requirements. However, the program does change the proposed withdrawal schedule to optimize the management of reactor pressure vessel (RPV) radiation damage. Following NRC approval of WCAP-15624, Table 4.5-3 of the UFSAR will be revised to reflect the Surveillance Schedule as shown in Table 5-1 of WCAP-15624.

Master Curve Methodology

Application of direct measurements of the fracture toughness (Master Curve Methodology) to reactor pressure vessel analysis often requires a method for transforming the measured values to equivalent values at the fluence of interest. Currently regulations and guides (10 CFR 50.61 and NRC Regulatory Guide 1.99, Revision 2) employ trend equations for the Charpy transition temperature shifts to accomplish these transitions. However, there is not an equivalent trend curve for

the Master Curve measurements. In the submittal for the Kewaunee Reactor Pressure Vessel (RPV), Reference 10, the available data spanned the fluence of interest and the transformation required only small interpolation. However, application to the BVPS Unit 1 RPV requires an extrapolation to the EOLE fluence to demonstrate the ability to remain within the current regulatory screening limit, since the current available surveillance material is irradiated to a maximum fluence of 2.15 x 10^{19} n/cm^2. The EOL and EOLE fracture toughness trend must be inferred by fitting RT_{To} data from the two highest fluence capsules. These results will be confirmed by future fracture toughness surveillance testing, which includes the newly inserted capsule in BVPS Unit 2.

Application of the Master Curve technology also requires the development of a margin strategy identified in WCAP-15624. Although ASME Code Cases N-629 and N-631 provide RT_{To}, which can be used as an alternative to RT_{NDT}, they do not provide guidance on the margins (if any) required to determine the corresponding values of RT_{PTS} or Adjusted Reference Temperature (ART), defined as ART = initial RT_{NDT} + Margins for uncertainties + ΔRT_{NDT}. The methodology being applied for the BVPS Unit 1 RPV is similar in method and intent to that developed for the Kewaunee RPV (Reference 11, WCAP-15075, "Master Curve Strategies for RPV Assessment"). However, there is a significant difference between the two when applied to the specific RPV materials. The limiting material for the BVPS Unit 1 RPV is a plate material with lower copper content and less potential variability in chemical and irradiation embrittlement than the weld metal evaluated for the Kewaunee RPV. Thus, uncertainties in applying the Master Curve approach for BVPS Unit 1 will be less than those for Kewaunee and therefore the heatup and cooldown curves generated will provide greater assurance of the reactor vessel integrity.

C. JUSTIFICATION

The current heatup and cooldown curves are being revised by reflecting the testing of irradiated materials from Surveillance Capsules Y and W and applying the Master Curve Methodology. The curves have been developed in accordance with the methodologies provided in WCAP-14040-NP-A, WCAP-15315, ASME Code Cases N-629, N-631, N-641, and the 1996 Addendum to the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G.

Master Curve Methodology

The use of the Master Curve Methodology, which employs Code Cases N-629 and N-631, provides additional operating margin during heatup and cooldown activities. Operating margin increases lower the risk associated with inadvertent actuation of the low temperature overpressure protection system and minimize the need for operator work around due to the narrow margin between the RCP seal limit and Appendix G curves (i.e., potentially reduces plant startup time). Additionally, testing and evaluating the properties of the RPV materials using the Master Curve approach provides a technically superior method for assessing radiation damage when compared to the Charpy V-notch approach.

The use of Code Case N-641 allows the use of the K_{IC} curve rather than the K_{IR} curve, the use of a circumferential flaw in the girth weld of the beltline, and a new criteria for establishing low temperature overpressure protection settings. The use of Code Case N-641 also eliminates the need to use three separate Code Cases (N-640, N-588, and N-514). By employing these conditions, the operating margin relative to the pump seal requirements is made larger and the chances of damaging the RCP seals and initiating a small break LOCA, a potential pressurized thermal shock initiator, are reduced. The gain in operating margin results in an increase in the safety of operating plants, as the likelihood of pump seal failure will decrease.

Reactor Vessel Flange Region

During the development of the ASME Appendix G requirements in the 1970's, a concern about the fracture margin in the reactor vessel closure flange region was identified. To address this concern the NRC added a requirement to Appendix G of 10 CFR 50 that the material temperature in the flange region must exceed the unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure, i.e., 621 psig for Beaver Valley Unit 1.

Improved knowledge of the fracture toughness and other fracture tolerance advancements has lead to the conclusion that this requirement is not longer necessary. The evaluation of and presentation of these improvements is provided in WCAP 15315 (Reference 3), which was endorsed by the ASME Code and submitted to the NRC in November of 1999. The WCAP demonstrates that the flange region is tolerant of assumed flaws in excess of 1/4 thickness at room temperature. Additionally, since there is no known degradation mechanism for the region and the fatigue usage in this region is less than 0.1, it may be concluded that

flaws are unlikely to initiate in this region. Based upon the conclusions drawn in WCAP 15315, it is requested that the burden of the flange requirement in Appendix G of 10 CFR 50 be eliminated from the pressure-temperature curves for Beaver Valley Unit 1.

Overpressure Protection System

The PORV setpoint and OPPS enable temperature are being revised based upon WCAP-14040-NP-A, Revision 2, ASME Code Case N-641, and 1996 Addendum to the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. As stated in WCAP-14040, the OPPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Thus, according to WCAP-14040-NP-A, it is appropriate to utilize the steady-state Appendix G limit to establish the setpoint. Therefore, the proposed setpoint is based on isothermal conditions.

D. SAFETY ANALYSIS

An analysis was performed to develop curves acceptable for heatup and cooldown based on the testing of irradiated materials from Surveillance Capsules Y and W and application of the Master Curve Methodology. Subsequently, the PORV setpoint was generated applicable to the 22 EFPY heatup and cooldown curves developed as a result of the analyses of Surveillance Capsules Y and W.

Heatup and cooldown limit curves are generated using the adjusted RT_{NDT} of the limiting beltline region materials of the reactor vessel. The adjusted RT_{NDT} values of the limiting materials in the core region of the reactor vessel are traditionally determined by using the unirradiated reactor vessel material Charpy V-notch impact toughness properties, estimating the radiation-induced ΔRT_{NDT}, (which is usually the 30 ft-lb temperature shift) and adding margin. The unirradiated RT_{NDT} is the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits no less than 50 ft-lb of impact energy and 35-mil lateral expansion minus 60°F.

In the traditional approach, RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is increased by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a

method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The referenced regulation provides a calculation methodology for ART values (IRT + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

Master Curve Methodology

For the proposed heatup/cooldown curves, a method different than the Charpy V-notch impact toughness approach for determining the adjusted RT_{NDT} of the limiting plate material is introduced. WCAP-15624 describes the approaches taken for the Master Curve fracture toughness application and the traditional Charpy approach.

Pressure and temperature limit curves for normal heatup and cooldown of the primary RCS have been calculated using the methods outlined in WCAP-15618, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Master Curve Applications" (Reference 12). These limit curves are developed using (ASME) Code Cases N-629, N-631, N-641 and WCAP-15315. These curves reflect the latest, conservative projected neutron fluence estimates including the effects of the power uprates (1.4% implemented and 8.0% planned), and the removal of hafnium absorbers from the core in 1R14, which took place in the year 2001.

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rule for Inservice Inspection of Nuclear Power Plant Components," Appendix G, contains conservative methods of analysis. The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor for the metal temperature at that time. K_{IC} can replace K_{IR} as the reference stress intensity factor by following Code Case N-641 and is obtained from the reference fracture toughness curve defined in the latest Addenda to ASME Code, Section XI.

The proposed methodology used to examine the integrity of the BVPS Unit 1 is based on ASME Code Cases N-629 and N-631. The two ASME Code Cases provide an alternative method for indexing the ASME reference fracture curves, based directly on fracture toughness measurements. ASME Code Case N-629 provides a means of measuring RT_{To}, to be used in the place of RT_{NDT} as an indexing temperature for non-irradiated materials for the ASME reference toughness curves. Code Case N-631 is equivalent to Code Case N-629 and is applicable for irradiated and unirradiated materials. Since Code Case N-631 values of reference temperature are determined using direct measurements of fracture toughness, they provide more accurate values than the indirect estimation procedure using Charpy technology. However, reactor vessel integrity analyses require the evaluation of estimated RT_{NDT} values as a function of fluence for inside surface, 1/4T and 3/4T locations in the vessel wall at end-of-life (EOL) and EOLE, which can only be accomplished by fitting a curve to measured data. The evaluation documented in WCAP-15624 involves the extrapolation to EOLE and indicates the RPV limiting plate material has adequate toughness to EOLE and beyond.

Supplemental Surveillance Capsule

A supplemental surveillance program has been designed and implemented that includes not only the limiting plate material, but also all of the BVPS Unit 1 beltline materials for future evaluation using the Master Curve Methodology. The testing of this supplemental capsule at a fluence corresponding close to EOLE or greater will confirm the toughness condition for the BVPS Unit 1 materials near the time when current EOL is reached.

Reactor Vessel Flange Region

The discussion given in WCAP-15315, concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using K_{IC} toughness. Further, there are no known mechanisms of degradation for this region, other than fatigue. Since the calculation design fatigue usage for this region according to WCAP-15315, is less than 0.1, it may be concluded that flaws are unlikely to initiate in this region. Stresses in the RPV flange region from head boltup have been analyzed and determined to be within ASME Code limits. Therefore, no additional boltup requirements are necessary, and the closure head/vessel flange region requirement of 10 CFR Part 50, Appendix G, can be eliminated from the pressure-temperature curves for Beaver Valley Unit 1.

Heatup/Cooldown Curves

The pressure difference between the wide-range pressure transmitter and the limiting beltline region has not been accounted for in the P/T limits curves generated for normal operation. Those differences will be incorporated into the plant procedure used to implement the P/T limit requirements. This pressure difference, however, has been incorporated into the generation of the OPPS setpoint.

The heatup curve, Figure 4.2-1 of the PTLR, was generated using heatup rates as high as 100°F/hr applicable for the first 22 EFPY with no margins for instrumentation errors included. The cooldown curves, Figure 4.2-2 of the PTLR, are generated up to 100°F/hr applicable for the first 22 EFPY with no margins for instrumentation errors included. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines for the P/T curves. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until P/T combinations are to the right of the critical limit line of the heatup curve shown in Figure 4.2-1 of the PTLR. The straight-line portion of the criticality limit is at the minimum permissible temperature of the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code.

The criticality limit curve specifies P/T limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G. The P/T limits for core operation (except for low-power physics tests) are set such that the reactor vessel must be at a temperature equal to or higher than the minimum permissible temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curves for the heatup and cooldown calculated as described in WCAP-15618. For the heatup curve, the minimum temperature for the inservice hydrostatic leak tests for the Beaver Valley Unit 1 reactor vessel at 22 EFPY is 278°F. The vertical line drawn from this point on the P/T curve, intersecting a curve 40°F higher than the P/T limit curve, constitutes the limit for core operation for the reactor vessel. The P/T limit curve defines all the above limits for ensuring prevention of non-ductile failure for the Beaver Valley Unit 1 reactor vessel during normal heatup and cooldown.

Overpressure Protection System

Based on TS changes provided in LAR No. 292, TS 3.4.9.3 requires that the OPPS be operable with a maximum of one charging pump capable of injecting into the RCS, and the accumulator isolated and either:

a) Two PORVs with a lift setting less than or equal to 403 psig, or

b) The RCS depressurized and an RCS vent of greater than or equal to 2.07 square inches.

The requirements are applicable to Mode 4 when any RCS cold leg temperature is less than, or equal to, an enable temperature of 343°F, Mode 5, and Mode 6 when the reactor vessel head is on.

The PORV setpoint and OPPS enable temperature are being revised to be based on WCAP 14040-NP-A and WCAP-15315, ASME Code Cases N-629, N-631, N-641, and 1996 Addendum to the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G.

PORV Setpoint

The current Maximum Allowable Nominal PORV setpoint is based on isothermal conditions. The values of the PORV setting and the enable temperature are revised in this submittal to 452 psig and 330°F, respectively, according to Reference 6, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Master Curve". The PORV setpoint and enable temperature are applicable up to and including 30% steam generator tube plugging. The PORV setpoint conservatively accounts for instrument uncertainties associated with the wide range pressure transmitter (±70 psig), per Reference 6. The setpoint also accounts for the pressure difference between the wide-range pressure transmitter and the reactor vessel limiting beltline region, identified in Westinghouse Nuclear Safety Advisory letter NSAL-93-005A. The PORV setpoint has been selected such that the peak overpressure will not exceed the Appendix G limit or the 800-psig PORV piping limit without any restrictions placed on the number of RCPs in operation.

Thermal transport effects, which are applied to the heat injection transient results and account for the steam generator operation at a temperature 50°F higher than the reactor coolant temperature, have also been incorporated into the PORV

setpoint. This increase in the secondary-to-primary side temperature from 25°F to 50°F is contained within the pending Unit 1 LAR No. 292. The heat addition event has been analyzed for RCS initial temperatures between 70°F and 300°F. The influx of fluid into the relatively inelastic RCS during the mass injection event causes a rapid increase in system pressure. The mass injection event uses the same RCS temperature, 100°F, that was used for the previous OPPS analysis. Pressure overshoots during the design basis events are based on a pressurizer PORV stroke open/close time of 3.0 seconds. The PORV setpoint is selected so that RCS pressures will not exceed the 22 EFPY Appendix G pressure limits down to the reactor vessel boltup temperature of 60°F.

During the PORV setpoint selection process it is noted that there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters and results in a maximum system pressure somewhat higher than the set pressure. Similarly, there will be a pressure undershoot while the valve is relieving. This is due to the reset pressure being below the setpoint and due to the delay in stroking the valve closed. In order to preserve the single-failure criteria, the overshoots are calculated assuming the availability of one PORV during the design basis mass injection and heat addition events, when the RCS is water-solid, concurrent with a loss-of-letdown. For conservatism the second PORV is assumed to have failed. The maximum and minimum pressures reached in the transient are a function of the selected setpoint and should fall within an acceptable pressure range.

The upper pressure limit for OPPS is defined by Appendix G requirements or by PORV piping limitations, after consideration of all uncertainties and the ΔP between the wide range pressure transmitter and the reactor vessel limiting region. The lower limit on pressure during the design basis OPPS mass injection and heat injection transients is established based on operational consideration for the RCP No. 1 seal limit which requires a nominal differential pressure of 200 psid across the seal faces for proper film-riding performance. As part of the PORV setpoint evaluation, margin to the RCP No. 1 seal limit is evaluated.

As demonstrated in Reference 6, pressure undershoot below the PORV setpoint during a design basis mass injection or heat injection event could cause the differential pressure to be less than the minimum requirement of the RCP No. 1 seal. Therefore, there is the potential for RCS pressure to violate the RCP No. 1 seal limit at the lowest RCS temperatures. If this occurs, the method for choosing

the PORV setpoint, which is consistent with approved methodology, as defined in WCAP-14040-NP-A, is followed. Further, with the same setpoint for both PORVs, there is a potential for both valves to open simultaneously, then on closing cause an undershoot that could also violate the RCP No. 1 seal limit.

OPPS Enable Temperature

The OPPS arming temperature (when the OPPS should be switched on to be operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature should be set to less than or equal to 330°F for 22 EFPY.

A calculated enable temperature is based on either an RCS temperature of less than 200°F, or materials concerns (reactor vessel metal temperature less than RT_{NDT} + 50°F), whichever is greater. The calculated enable temperature does not address the piping limit issue attributable to water hammer event. The calculated enable temperature is 300°F for 22 EFPY. The higher of the arming temperature and the calculated enable temperature is chosen for the OPPS enable temperature to ensure a conservative value, such that both the vessel and the PORV discharge piping are protected. As the arming temperature is higher and therefore, more conservative than the calculated enable temperature, the OPPS enable temperature is set to equal the arming temperature of 330°F for 22 EFPY.

E. NO SIGNIFICANT HAZARDS EVALUATION

In accordance with 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) proposes to amend Operating License DPR-66 for Beaver Valley Power Station (BVPS) Unit 1 to allow a different method for determining the adjusted RT_{NDT} (reference nil-ductility temperature) of the limiting beltline region materials of the reactor vessel. This method, the Master Curve Methodology, as described in WCAP-15624, "Master Curve Fracture Toughness Application for BVPS-1", is used to generate the heatup and cooldown curves, power operated relief valve (PORV) setpoint, and overpressure protection system (OPPS) enable temperature. The proposed pressure/temperature (P/T) limits have been prepared using the NRC approved methodology described in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" , with five exceptions. They include the use of:

a) ASME Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1," May 7, 1999,

b) ASME Code Case N-631, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials Other Than Bolting for Class 1 Vessels, Section III, Division 1," September 24, 1999,

c) ASME Code Case N-641, "Alternate Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000,

d) WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," October 1999, and

e) ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

In conjunction with this proposed license amendment, FirstEnergy Nuclear Operating Company (FENOC) requests exemptions to 10 CFR 50.60 and 10 CFR 50 Appendix G, pursuant to 10 CFR 50.12 (a) (2) (iii) based on American Society of Mechanical Engineers (ASME) Code Cases N-629, N-631, N-641 and WCAP-15315.

The following provides a summary of the proposed changes:

1. Technical Specification 6.9.6 is revised to include methods used to generate the proposed pressure and temperature limits, the power operated relief valve (PORV) setpoint and the overpressure protection system (OPPS) enable temperature.

2. The Pressure and Temperature Limits Report (PTLR) is revised:

   a) to be consistent with Technical Specification 6.9.6,

   b) by adding a discussion concerning the Master Curve Methodology and references to the applicable Westinghouse Topical Reports,

   c) by stating that the curves do not include the effect of the reactor vessel flange,

d) by changing the applicable ASME Code Cases to those listed above,

e) by changing tables, figures and the OPPS arming and enable temperatures, to reflect the Master Curve Methodology.

To meet format requirements, the Technical Specification Index and Bases, and PTLR pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focused on the three standards set forth in 10 CFR 50.92(c), as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

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

   No. The proposed changes do not result in physical changes being made to structures, systems, or components (SSCs), or to event initiators or precursors. The revised curves and corresponding overpressure protection system (OPPS) enable temperature have been developed using accepted engineering practices, methods derived from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, criteria set forth in Nuclear Regulatory Commission (NRC) Regulatory Standard Review Plan 5.3.2, and 10 CFR 50.61. They provide an adequate margin of

safety to ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events. This change in the heatup and cooldown curves, power operated relief valve (PORV) setpoint, and OPPS enable temperature will not affect the ability of the OPPS to control the reactor coolant system (RCS) at low temperatures such that the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure and temperature limits.

Also, the proposed changes do not impact the design of plant systems such that SSCs would now be more likely to fail. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed. Thus, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes do not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequence of an accident described in the UFSAR. As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of the SSCs to limit the radiological consequences of previously evaluated accidents. Reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes do not involve physical changes to analyzed SSCs or changes to the modes of plant operation defined in the Technical Specifications. The proposed changes do not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor do they alter the design or operation of any plant systems. No new accident scenarios, accident or transient initiators or precursors,

failure mechanism, or limiting single failures are introduced as a result of the proposed changes.

The proposed changes do not cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new or different mode of failure has been created and no new or different equipment performance requirements are imposed for accident mitigation. As such, the proposed changes have not effect on previously evaluated accidents.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident for any previously evaluated.

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

No. The proposed 10 CFR 50 Appendix G pressure/temperature limitations and OPPS enable temperature were prepared using methods derived from the ASME Boiler and Pressure Vessel Code, including Code Cases N-629, N-631, and N-641, and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. These documents and practices along with calculational limitations specified in 10 CFR 50.61 provide for an adequate margin of safety and ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events as described in the UFSAR.

The proposed heatup and cooldown curves define the limits for ensuring prevention of non-ductile failure for the Beaver Valley Unit 1 reactor coolant pressure boundary, and do not significantly reduce the margin of safety for the plant. The use of Code Cases N-629, N-631, and N-641 and WCAP-15315 eliminates excessive conservatism based on inspection data and improved knowledge of fracture toughness.

The PORV setpoint and OPPS enable temperature will continue to ensure the RCS pressure boundary will be protected from pressure transients. These values were generated using the proposed heatup and cooldown curves as input. The OPPS setpoint includes allowance for instrument uncertainties. Therefore, the margin of safety is not reduced.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfy the requirements of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL CONSIDERATION

This license amendment request changes an evaluation methodology applicable to a component located within the restricted area as defined in 10 CFR Part 20. It has been determined that this license amendment request involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. This license amendment request does not change requirements with respect to installation or use of a facility component located within the restricted area or change an inspection or surveillance requirement. The category of this licensing action does not individually or cumulatively have a significant effect on the human environment. Accordingly, this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this license amendment request.

References

1. WCAP-15624, "Master Curve Fracture Toughness Application for BVPS-1", W. L. Server, B. N. Bugos, D. Weakland, Westinghouse Non-Proprietary Class 3, dated November 2001, transmitted by FENOC letter L-02-004.

2. WCAP 14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek et al., January 1996, transmitted by Westinghouse Owner's Group letter from R. A. Newton to NRC, dated December 20, 1994.

3. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," Warren Bamford, October 1999, transmitted by Westinghouse letter NSBU-NRC-99-5954, dated November 4, 1999.

4. WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," C. Brown, et. al., November 2000, transmitted by FENOC letter L-01-027, dated March 7, 2001.

5. WCAP-12005, "Analysis of Capsule W from the Duquesne Light Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, S. L. Anderson, L. Albertin, November 1998, transmitted by Duquesne Light Company letter from John D. Sieber dated January 24, 1989.

6. Westinghouse Report I&CE/PEOD(01)-206, Westinghouse Proprietary Class 2, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Master Curve", Revision 1, August 2001, transmitted by FENOC letter L-02-004.

7. WCAP-8457, "Duquesne Light Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," J. A. Davidson, October 1974.

8. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D. C., Federal Register, Volume 60, No. 243, December 19, 1995.

9. ASTM Standard E 1921-97, "Test Method for the Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range," 1998 Annual Book of ASTM Standards, Vol. 03.01, American Society for Testing and Materials, West Conshohocken, PA.

10. NRC letter issued February 21, 2001, "Kewaunee Nuclear Power Plant – Request for Exemption from the Requirements of 10CFR Part 50, Appendices G and H, and 10 CFR 50.61 (TAC NO. MA8585)".

11. WCAP-15075, "Master Curve Strategies for RPV Assessment," R. G. Lott, et al., Westinghouse Non-Proprietary Class 3, dated September 1998, transmitted by Kewaunee letter NRC-98-116, dated November 18, 1998.

12. WCAP-15618, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Master Curve Applications", Westinghouse Non-Proprietary Class 3, C. Brown, C. Kim., dated November 2001, transmitted by FENOC letter L-02-004.

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

ATTACHMENT C

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 297

The following is a list of affected PTLR pages.

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4.2-iv	295
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SECTION 4.2 PRESSURE AND TEMPERATURE LIMITS REPORT

List of Tables

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4.2 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Unit 1 has been prepared in accordance with the requirements of Technical Specification 6.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) addressed in, or make reference to, this report are listed below:

TS 3.1.2.2 Reactivity Control Systems – Flow Paths – Operating,
TS 3.1.2.4 Reactivity Control Systems – Charging Pumps – Operating,
TS 3.4.1.2 Reactor Coolant System – Hot Standby,
TS 3.4.1.3 Reactor Coolant System – Shutdown,
TS 3.4.9.1 Reactor Coolant System - Pressure/Temperature Limits,
TS 3.4.9.3 Overpressure Protection Systems,
TS 3.5.2 ECCS Subsystems – $T_{avg} \geq 350°F$, and
TS 3.5.3 ECCS Subsystems – $T_{avg} < 350°F$.

4.2.1 Operating Limits

The PTLR limits for Beaver Valley Power Station (BVPS) Unit 1 were developed using a methodology specified in the Technical Specifications. The methodology listed in Reference 1 was used with ~~two~~ five exceptions:

a) Use of ASME Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1," May 7, 1999,

b) Use of ASME Code Case N-631, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials Other Than Bolting for Class 1 Vessels, Section III, Division 1," September 24, 1999,

c) Use of ASME Code Case N-641, "Alternate Pressure-Temperature Relationship and Low temperature Overpressure Protection System Requirements," January 17, 2000,

d) Use of WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," October 1999, and

e) Use of ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

~~a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and~~

~~b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".~~

4.2.1.1 RCS Pressure and Temperature (P/T) Limits (TS 3.4.9.1)

The RCS temperature rate-of-change limits defined in Reference 2 are:

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period, and

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c. A maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic testing operations above system design pressure.

The RCS P/T limits for heatup, leak testing, and criticality are specified by Figure 4.2-1 and Table 4.2-1. The RCS P/T limits for cooldown are shown in Figure 4.2-2 and Table 4.2-2. These limits are defined in Reference 2. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 4.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 4.2-1 and 4.2-2 are provided without margins for instrument error. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G. The heatup and cooldown curves ~~also~~ do not include the effect of the reactor vessel flange.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT}. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal working direction) minus 60°, or determined using the Master Curve Methodology. For Lower Shell Plate B6903-1 RT_{NDT} calculations, the Master Curve Methodology was applied as documented in WCAP-15624. For calculation of the adjusted reference temperature WCAP-15618 was used.

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PRESSURE AND TEMPERATURE LIMITS REPORT

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

Pressure-temperature limit curves shown in Figure 4.2-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-~~640~~641.

4.2.1.2 Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

The power operated relief valves (PORVs) shall each have maximum lift setting and enable temperature in accordance with Table 4.2-3. The lift setting provided does not impose any reactor coolant pump restrictions.

The PORV setpoint is based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with the methodology described in Reference 1, including the exceptions noted in Section 4.2.1. The PORV lift setting shown in Table 4.2-3 accounts for appropriate instrument error.

4.2.1.3 OPPS Enable Temperature (TS 3.4.9.3)

Two different temperatures are used to determine the OPPS enable temperature, they are the arming temperature and the calculated enable temperature. The arming temperature (when the OPPS rendered operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature is ~~343~~330°F.

The calculated enable temperature is based on either a RCS temperature of less than 200°F or materials concerns (reactor vessel metal temperature less than RT_{NDT} + 50°F), whichever is greater. The calculated enable temperature does not address the piping limit attributed to a water hammer discharge. The calculated enable temperature is ~~308°F~~300°F.

As the arming temperature is higher and, therefore, more conservative than the calculated enable temperature, the OPPS enable temperature, as shown in Table 4.2-3, is set to equal the arming temperature.

The calculation method governing the heatup and cooldown of the RCS requires the arming of the OPPS at and below the OPPS enable temperature specified in Table 4.2-3, and disarming of the OPPS above this temperature. The OPPS is required to be enabled, i.e., OPERABLE, when any RCS cold leg temperature is less than or equal to this temperature.

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4.2.3 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 4.2-4, taken from Reference 5, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2-5, taken from References 2 and 5, provides the reactor vessel beltline material property table.

Table 4.2-6, taken from Reference 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 22 EFPY.

Table 4.2-7, taken from Reference 2, shows the calculation of ARTs for 22 EFPY.

Table 4.2-8 shows the Reactor Vessel Toughness Data (Unirradiated).

Table 4.2-9, taken from Reference 5, provides RT_{PTS} values for ~~28~~ 27.44 EFPY.

Table 4.2-10, taken from Reference 5, provides RT_{PTS} values for ~~45~~ 44.18 EFPY.

The pre-irradiation fracture-toughness properties of the Beaver Valley Unit 1 reactor vessel material are presented in the Table 4.2-8. The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan and ASME Code Cases N-629 and N-631. The post-irradiation fracture toughness properties of the reactor vessel beltline material, determined in accordance with 10 CFR 50, Appendix H, were obtained directly from the Beaver Valley Unit 1 Vessel Material Surveillance Program.

4.2.4 References

1. WCAP-14040-NP-A, Revision 2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” J. D. Andrachek, et al., January 1996.

2. WCAP-15618, Revision 0, “Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Master Curve Applications,” C. Brown, C. Kim, November 2001.~~WCAP-15570, Revision 2, “Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation,” T. J. Laubham, April 2001.~~

3. WCAP-15571, “Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program,” C. Brown, et. al., November 2000.

4. WCAP-8475, “Duquesne Light Company, Beaver Valley Unit No. 1 Reactor Vessel Radiation Surveillance Program,” J. A. Davidson, October 1974.

5. WCAP-15624, Revision 0, “Master Curve Fracture Toughness Application for BVPS-1,” W. L. Server, B. N. Bugos, D. Weakland., November 2001.~~WCAP-15569, “Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1,” C. Brown, et al., November 2000.~~

6. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” Federal Register, Volume 60, No. 243, December 19, 1995.

7. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” May 15, 1991. (PTS Rule)

8. Regulatory Guide 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials,” U.S. Nuclear Regulatory Commission, May 1988.

9. ~~9. Westinghouse Report, “Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company Overpressure Protection System Setpoints for Y-Capsule”, Revision 1, April 2001.~~Westinghouse Proprietary Class 2 Report, I&CE/PEOD(01)-206, “Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Master Curve”, Revision 1, August 2001.

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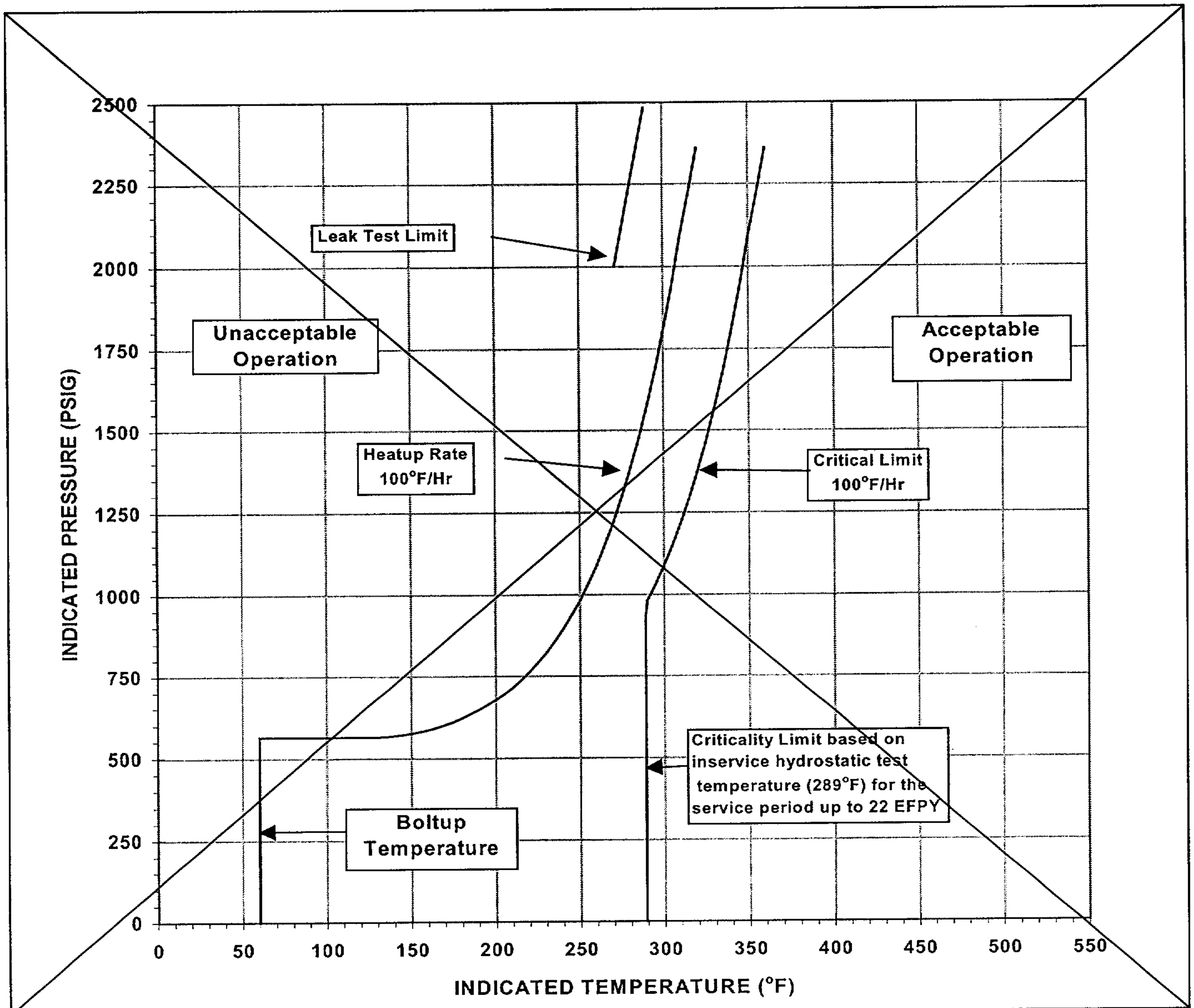
MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 22 EFPY: 1/4T, ~~233~~222°F

3/4T, 196°F



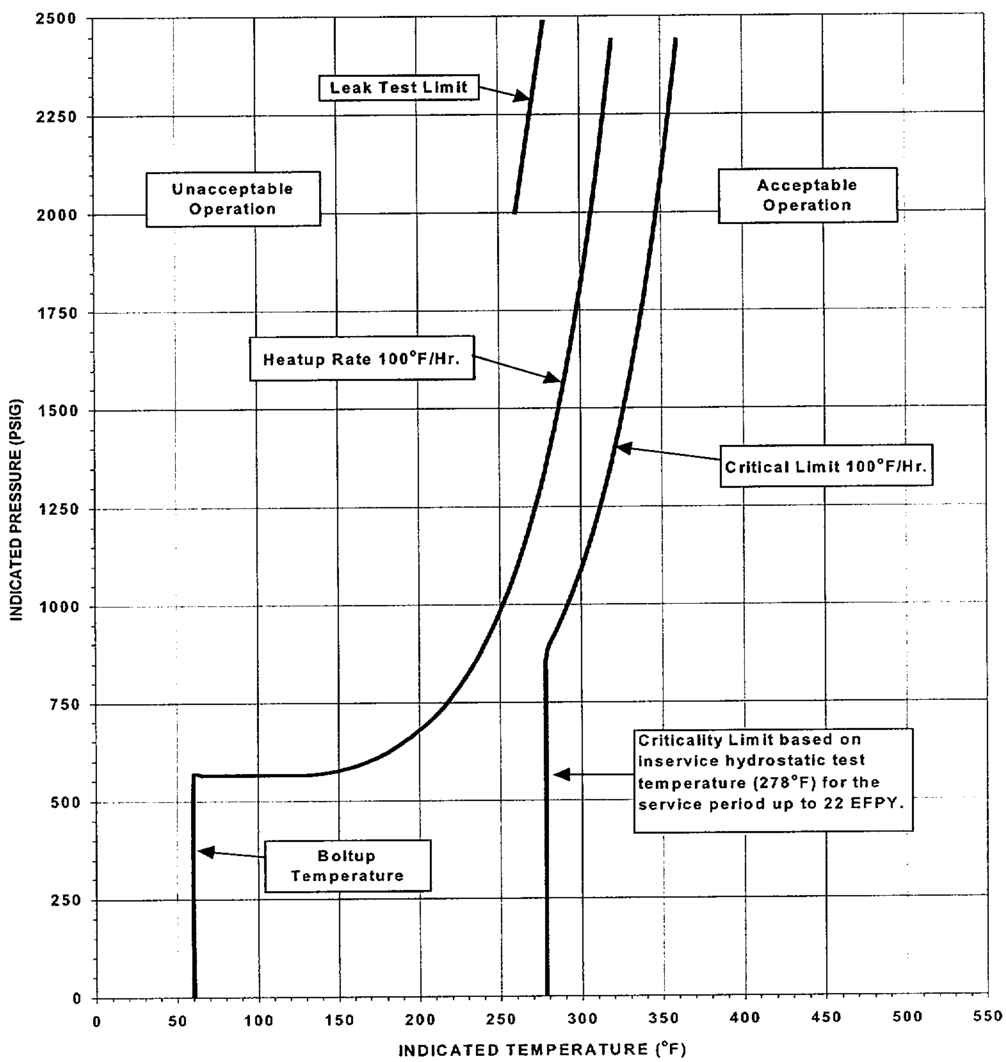
Replace with Insert A-1

Figure 4.2-1
Reactor Coolant System Heatup
Limitations Applicable for the First 22 EFPY (TS 3.4.9.1)

LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

Insert A-1



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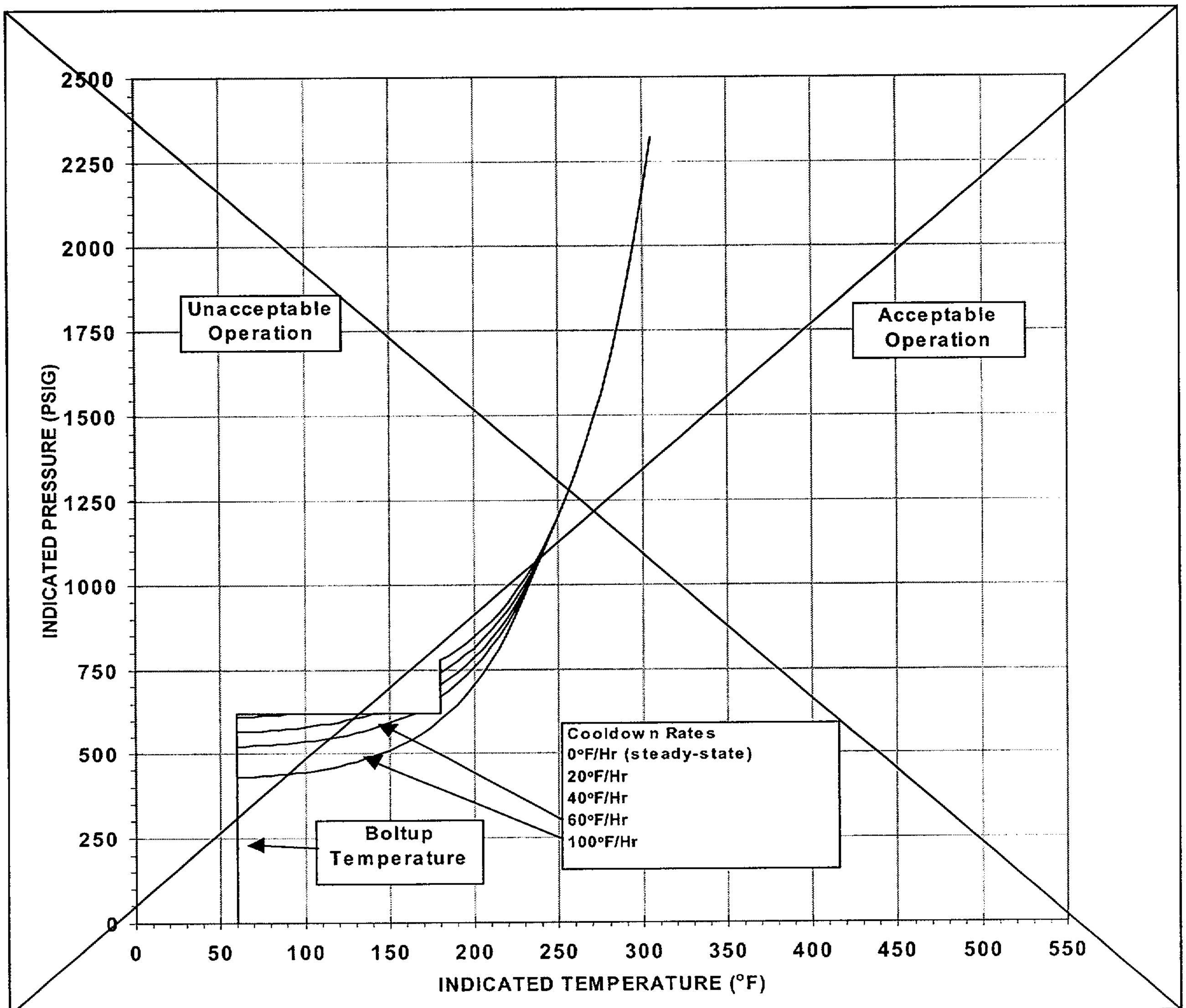
MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 22 EFPY: 1/4T, ~~233~~222°F

3/4T, 196°F



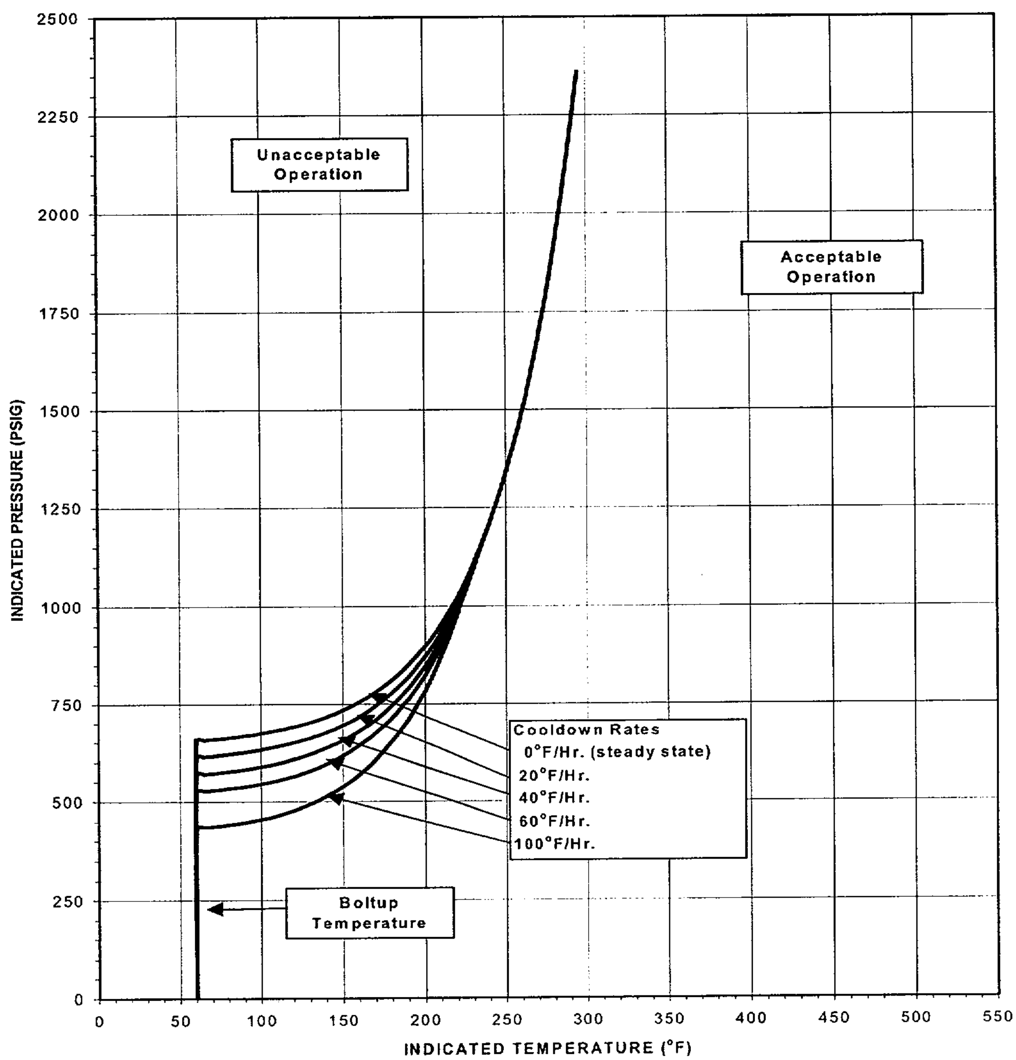
Replace with Insert A-2

Figure 4.2-2
Reactor Coolant System Cooldown
Limitations Applicable for the First 22 EFPY (TS 3.4.9.1)

LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

Insert A-2



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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-1
Heatup Curve Data Points for 22 EFPY (TS 3.4.9.1)

100°F/HR HEATUP				100°F/HR CRITICALITY				LEAK TEST LIMIT	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	210~~200~~	716~~677~~	~~289~~278	0	278~~289~~	739~~716~~	~~271~~260	2000
60	564	215~~205~~	739~~696~~	~~289~~278	564	278~~289~~	764~~739~~	~~289~~278	2485
65	564	220~~210~~	764~~716~~	~~289~~278	565	278~~289~~	792~~764~~		
70	564	225~~215~~	792~~739~~	~~289~~278	565	278~~289~~	822~~792~~		
75	564	230~~220~~	822~~764~~	~~289~~278	566	278~~289~~	856~~822~~		
80	564	235~~225~~	856~~792~~	~~289~~278	566	280~~289~~	894~~856~~		
85	564	240~~230~~	894~~822~~	~~289~~278	568	285~~289~~	936~~894~~		
90	564	245~~235~~	936~~856~~	~~289~~278	569	290~~289~~	982~~936~~		
95	564	250~~240~~	982~~894~~	~~289~~278	571	295~~290~~	1033~~982~~		
100	564	255~~245~~	1033~~936~~	~~289~~278	572	300~~295~~	1089~~1033~~		
105	564	260~~250~~	1089~~982~~	~~289~~278	575	305~~300~~	1151~~1089~~		
110	564	265~~255~~	1151~~1033~~	~~289~~278	577	310~~305~~	1219~~1151~~		
115	564	270~~260~~	1219~~1089~~	~~289~~278	580	315~~310~~	1294~~1219~~		
120	564	275~~265~~	1294~~1151~~	~~289~~278	583	320~~315~~	1378~~1294~~		
125	564	280~~270~~	1378~~1219~~	~~289~~278	586	325~~320~~	1470~~1378~~		
130	565	285~~275~~	1470~~1294~~	~~289~~278	591	330~~325~~	1571~~1470~~		
135	566	290~~280~~	1571~~1378~~	~~289~~278	593	335~~330~~	1682~~1571~~		
140	568	295~~285~~	1682~~1470~~	~~289~~278	600	340~~335~~	1806~~1682~~		
145	571	300~~290~~	1806~~1571~~	~~289~~278	601	345~~340~~	1941~~1806~~		
150	575	305~~295~~	1941~~1682~~	~~289~~278	611	350~~345~~	2091~~1941~~		
155	580	310~~300~~	2091~~1806~~	~~289~~278	612	355~~350~~	2256~~2091~~		
160	586	315~~305~~	2256~~1941~~	~~289~~278	621	360~~355~~	2438~~2222~~		
165	593	320~~310~~	2438~~2091~~	~~289~~278	~~621~~625	~~360~~	~~2361~~		
170	601	~~315~~	~~2222~~	~~289~~278	~~621~~633				
175	611	~~320~~	~~2361~~	~~289~~278	~~621~~641				
180	621			~~289~~278	~~621~~646				
~~180~~185	~~621~~633			~~289~~278	~~633~~659				
~~180~~190	~~621~~646			~~289~~278	~~646~~661				
~~185~~195	~~633~~661			~~289~~278	~~661~~677				
~~190~~200	~~646~~677			~~289~~278	~~677~~696				
~~195~~205	~~661~~696			~~289~~278	~~696~~716				

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-2 (Page 1 of 2)
Cooldown Curve Data Points for 22 EFPY (TS 3.4.9.1)

STEADY STATE		20°F/HR		40°F/HR.		60°F/HR.		100°F/HR.	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0	60	0
60	655~~21~~	60	613~~09~~	60	569~~6~~	60	525~~1~~	60	434~~0~~
65	657~~21~~	65	614~~11~~	65	571~~67~~	65	527~~3~~	65	436~~1~~
70	659~~21~~	70	616~~2~~	70	573~~68~~	70	5294	70	438~~2~~
75	661~~21~~	75	6184	75	575~~70~~	75	531~~25~~	75	440~~33~~
80	663~~21~~	80	620~~15~~	80	577~~2~~	80	533~~27~~	80	442~~35~~
85	665~~21~~	85	623~~17~~	85	5794	85	535~~29~~	85	445~~37~~
90	668~~21~~	90	626~~19~~	90	582~~76~~	90	538~~31~~	90	448~~39~~
95	671~~21~~	95	629~~21~~	95	585~~78~~	95	542~~34~~	95	451~~42~~
100	674~~21~~	100	632~~21~~	100	589~~1~~	100	545~~36~~	100	455~~45~~
105	678~~21~~	105	636~~21~~	105	593~~84~~	105	549~~40~~	105	460~~48~~
110	682~~21~~	110	640~~21~~	110	597~~87~~	110	554~~43~~	110	465~~52~~
115	686~~21~~	115	645~~21~~	115	602~~591~~	115	559~~47~~	115	471~~57~~
120	691~~21~~	120	650~~21~~	120	608~~596~~	120	565~~52~~	120	478~~62~~
125	697~~21~~	125	656~~21~~	125	614~~00~~	125	571~~57~~	125	485~~68~~
130	703~~621~~	130	662~~21~~	130	620~~06~~	130	579~~62~~	130	493~~74~~
135	710~~621~~	135	669~~21~~	135	628~~12~~	135	587~~69~~	135	503~~481~~
140	717~~621~~	140	677~~21~~	140	636~~18~~	140	596~~76~~	140	513~~490~~
145	725~~621~~	145	686~~21~~	145	646~~21~~	145	605~~584~~	145	525~~499~~
150	734~~621~~	150	695~~21~~	150	656~~21~~	150	617~~592~~	150	538~~09~~
155	744~~621~~	155	706~~621~~	155	667~~21~~	155	629~~02~~	155	552~~20~~
160	755~~621~~	160	718~~621~~	160	680~~21~~	160	643~~13~~	160	568~~33~~
165	767~~621~~	165	731~~621~~	165	694~~21~~	165	658~~21~~	165	586~~47~~
170	781~~621~~	170	745~~621~~	170	710~~621~~	170	675~~21~~	170	606~~563~~
175	795~~621~~	175	761~~621~~	175	727~~621~~	175	693~~21~~	175	629~~581~~
180	812~~621~~	180	779~~621~~	180	746~~621~~	180	714~~621~~	180	653~~00~~
185~~0~~	830~~621~~	185~~0~~	798~~621~~	185~~0~~	767~~621~~	185~~0~~	737~~621~~	185	681~~22~~
19~~8~~0	850~~778~~	19~~8~~0	820~~742~~	19~~8~~0	791~~06~~	19~~8~~0	763~~670~~	190	711~~647~~
19~~8~~5	872~~792~~	19~~8~~5	844~~757~~	19~~8~~5	817~~723~~	19~~8~~5	791~~689~~	195	745~~674~~

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Table 4.2-2 (Page 2 of 2)
Cooldown Curve Data Points for 22 EFPY (TS 3.4.9.1)

STEADY STATE		20°F/HR.		40°F/HR.		60°F/HR		100°F/HR	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
200	897	200	871	200	846	200	823	200	783
205	924	205	900	205	878	205	858	205	825
210	954~~892~~	210	932~~865~~	210	913~~839~~	210	896~~15~~	21~~2~~0	871~~61~~
215	987~~18~~	215	968~~894~~	215	952~~871~~	215	939~~850~~	21~~2~~5	923~~11~~
220	1023~~947~~	220	1008~~925~~	220	996~~05~~	220	986~~888~~	22~~3~~0	980~~67~~
225	1064~~980~~	225	1052~~961~~	225	1044~~944~~	225	1039~~930~~	22~~3~~5	104~~30~~
230	1108~~016~~	230	1101~~000~~	230	1097~~986~~	230	1097~~976~~	23~~4~~0	1108~~099~~
235	1157~~055~~	235	1154~~043~~	235	1156~~033~~	235	1157~~028~~	23~~4~~5	1157~~147~~
240	1212~~099~~	240	1212~~090~~	240	1212~~086~~	240	1212~~085~~	24~~5~~0	1212~~01~~
245	1272~~147~~	245	1272~~143~~	245	1272~~143~~	245	1272~~147~~	24~~5~~5	1272~~60~~
250	1339~~201~~	250	1339~~201~~	250	1339~~201~~	250	1339~~201~~	25~~6~~0	1339~~25~~
255	1412~~260~~	255	1412~~260~~	255	1412~~260~~	255	1412~~260~~	25~~6~~5	1412~~397~~
260	1494~~325~~	260	1494~~325~~	260	1494~~325~~	260	1494~~325~~	26~~7~~0	1494~~77~~
265	1583~~397~~	265	1583~~397~~	265	1583~~397~~	265	1583~~397~~	26~~7~~5	1583~~65~~
270	1683~~477~~	270	1683~~477~~	270	1683~~477~~	270	1683~~477~~	27~~8~~0	1683~~62~~
275	1792~~565~~	275	1792~~565~~	275	1792~~565~~	275	1792~~1565~~	27~~8~~5	1792~~70~~
280	1914~~662~~	280	1914~~662~~	280	1914~~662~~	280	1914~~1662~~	28~~9~~0	1914~~888~~
285	2048~~1770~~	285	2048~~1770~~	285	2048~~1770~~	285	2048~~1770~~	28~~9~~5	2048~~020~~
290	2196~~1888~~	290	2196~~1888~~	290	2196~~1888~~	290	2196~~1888~~	290~~300~~	2196~~65~~
295	2359~~020~~	295	2359~~020~~	295	2359~~020~~	295	2359~~2020~~	29~~30~~5	2359~~325~~

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-3

Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

FUNCTION	SETPOINT
OPPS Enable Temperature	~~343~~330°F
PORV Setpoint	~~403~~ 452 psig

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Table 4.2-4
Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF * ΔRT_{NDT}	FF^2
Lower Shell Plate B6903-1[d] (Longitudinal)	V	.323	.689	128.49	88.53	.475
	U	.646	.878	118.93	104.42	.771
	W	.986	.996	148.52	147.93	.992
	Y	2.15	1.208~~1~~	142.18	~~172.04~~171.75	1.4~~6~~459
Lower Shell Plate B6903-1[d] (Transverse)	V	.323	.689	137.81	94.95	.475
	U	.646	.878	131.84	115.76	.771
	W	.986	.996	179.99	179.27	.992
	Y	2.15	1.208~~1~~	166.93	201.~~99~~65	1.4~~64~~459
					SUM: 1104.~~89~~26	7.4~~04~~394
	$CF_{Plate\ B6903-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = (1104.~~89~~26) ÷ (7.4~~04~~394) = 149.24°F					
~~Beaver Valley~~	V	.323	.689	~~169.30~~159.72	1106.065	.475
~~Surv.~~ Weld ~~Material~~Metal (Heat 305424)[d]	U	.646	.878	~~176.30~~166.32	~~154.79~~146.03	.771
	W	.986	.996	~~198.99~~187.73	~~198.19~~186.98	.992
	Y	2.15	1.208~~1~~	~~189.41~~179.69	~~229.19~~217.07	1.4~~64~~459
					SUM: ~~698.82~~660.13	3.7~~02~~697
	$CF_{Weld\ Material} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = (66098.1382) ÷ (3.7~~02~~697) = 1788.68°F					

Notes:

(a) ~~Ff~~ f = Calculated fluence from ~~Beaver Valley Unit 1~~ capsule Y evaluation~~dosimetry analysis results,~~ (x 10^{19} n/cm², E > 1.0 ~~Mev~~MeV).

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

(c) ~~The surveillance weld metal~~ ΔRT_{NDT} values are the measured 30 ft-lb temperature shift values (°F)

~~have been adjusted by a ration factor of 1.06~~

~~Data not credible.~~

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Table 4.2-5

Reactor Vessel Beltline Material Properties

Material Description	Cu(%)	Ni(%)	Chemistry Factor (°F) [a]	Initial RT_{NDT}(°F) [b~~a~~]
Intermediate Shell Plate B6607-1	0.14	0.62	100.5	43
Intermediate Shell Plate B6607-2	0.14	0.62	100.5	73
Lower Shell Plate B6903-1[c]	0.205[d]~~1~~	0.5354[d]	~~147.2~~163.2[e]	-5~~27~~[e]
Lower Shell Plate B7203-2	0.14	0.57	98.7	20
Intermediate Shell Longitudinal ~~Shell~~ Welds ~~Seams (Heat 305424)~~ 19-714 A/~~&~~B[c]	0.282[d]~~8~~	0.~~63~~630[d]	~~191.7~~192.5	-56
Lower Shell Longitudinal Welds ~~Seams (Heat 305414)~~ 20-714 A/~~&~~B	0.3374	0.609~~1~~	~~210.5~~209.1	-56
Circumferential Weld 11-714	0.269	0.070	124.3	-56
~~Surveillance Weld (Heat 305424)~~	~~0.26~~	~~0.61~~	~~181.6~~	—

Note:

(a) Chemistry factor based on measured Cu and Ni chemistry and tables in References 2 and 5.

~~(a)~~(b) ~~The i~~Initial RT_{NDT} values for the plate materials ~~and~~ are ~~based on~~ measured values, while those for the ~~data while the~~ weld materials were not measured and generic values are used~~generic~~.

~~(b)~~(c) These materials are contained in the BVPS-1 surveillance program.

(d) Adjusted based on latest Capsule Y measurements.

(e) These values are based on Master Curve Methodology.

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-6

Summary of Adjusted Reference Temperature (ARTs) for 22 EFPY

MATERIAL DESCRIPTION	22 EFPY	
	1/4T ART(°F)[a]	3/4T ART(°F)[a]
Intermediate Shell Plate B6607-1	192~~3~~	166
Intermediate Shell Plate B6607-2	222~~3~~	196
Lower Shell Plate B7203-2	167~~8~~	~~141~~142
Lower Shell Plate B6903-1	214~~30~~	~~191~~172
~~Using S/C Data[b]~~	~~233~~	~~193~~
Intermediate Shell Longitudinal Weld 19-714A/B	145	~~102~~101
~~Using S/C Data[b]~~	~~143~~	~~100~~
Intermediate to Lower Shell Circ. Weld 11-714	152	~~119~~120
~~Using S/C Data[c]~~	~~86~~	~~63~~
Lower Shell Longitudinal Weld 20-714A/B	158~~9~~	~~111~~110
~~Using S/C Data[d]~~	~~168~~	~~117~~

Notes:

(a) ART = IRT + Bias + M + ΔRT.

IRT values are measured values. IRT values from Charpy approach are initial RT_{NDT} measurements, and IRT for the Master Curve results is $T_{o(u)} + 35°F$.

Bias = potential bias in Master Curve fracture toughness testing of Charpy V Notch size specimens in three-point bending (PCVN or RPCVN).

Margin = uncertainty in determining T_o is 12°F; $2\sigma T_o$=24°F; otherwise based on Charpy methodology.

ΔRT = CF * FF; CF from Master Curve method is based on the best fit through the irradiated ΔRT_{To} data using Charpy-based FF relation.

~~(a)ART = I + ΔRT_{NDT} + M.~~

~~(b)Based on Beaver Valley Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_Δ.)~~

~~(c)Based on credible St. Lucie Unit 1 surveillance data.~~

~~(d)Based on Fort Calhoun Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_Δ.)~~

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-7

Calculation of Adjusted Reference Temperatures (ARTs) for 22 EFPY

PARAMETER	VALUES	
Operating Time	22 EFPY	
Material	Plate B6607-2~~903-1~~	Plate B6607-2
Location	Intermediate~~Lower~~ Shell Plate 1/4T~~ ART(°F)~~	Intermediate Shell Plate 3/4T~~ ART(°F)~~
Chemistry Factor, CF (°F)	~~149.2~~100.5	100.5
Fluence (f), (x 10^{19} n/cm^2, (E >1.0 MeV~~v~~)[(a)]	1.72~~0 x 10^{19}~~	0.668~~6.62 x 10^{18}~~
Fluence Factor, FF[(b)]	1.14~~95~~	0.8874
ΔRT_{NDT} = CF * ~~x~~ FF, (°F)[~~(c)~~]	115~~71.56~~ [~~(c)~~]	898.184
~~Initial~~ RT$_{NDT}$, ~~I~~(°F)[(c~~a~~)]	~~27~~73	73
Margin, M(°F)	34 [~~(c)~~]	34
ART = ~~I+(CF*FF)+M, °F[(b)] per RG 1.99, Revision 2~~IRT + M + ΔART	~~233~~222	196

Notes:

(a) f(¼T) or f(¾T) = $f_{surf} * e^{-0.24x}$, where x = ¼T or x = ¾T. T is the vessel thickness of 7.88 inches

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

(c) IRT values (from Charpy approach) are initial RT_{NDT} measurements.

~~(a)Initial RT$_{NDT}$ values are measured values for plate material.~~

~~(b)This value was rounded per ASTM E29, using the "Rounding Method."~~

~~(c)Based on Beaver Valley Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_A.)~~

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Table 4.2-9

RT_{PTS} Calculation for Beltline Region Materials at EOL (~~28~~ 27.44 EFPY)

Material	Fluence ($x\ 10^{19}$ n/cm^2, E>1.0 MeV)	FF[(d)]	CF[(a)] (°F)	ΔRT_{PTS}[(ba)] (°F)	Margin (°F)	~~$RT_{NDT(U)}$~~ IRT[(b)] (°F)	~~RT_{PTS}~~ ART[(ce)] (°F)
Intermediate Shell Plate B6607-1	3.524	1.3289	100.5	133.56	34	43	2101
Intermediate Shell Plate B6607-2	3.524	1.3289	100.5	133.56	34	73	2401
Lower Shell Plate B7203-2	3.524	1.3289	~~98.7~~98.65	131.02	34	20	185
Lower Shell Plate B6903-1	3.524	1.3289	16347.2	225.0~~195.6~~	234	-527	24457
~~→ Using S/C Data[(e)]~~	~~3.54~~	~~1.329~~	~~149.2~~	~~198.3~~	~~34~~	~~27~~	~~259~~
Inter. Shell Long. Welds 19-714A/B	0.7048	0.9023	~~191.~~192.57	~~173.~~173.61	65.5	-56	1833
~~→ Using S/C Data[(e)]~~	~~0.708~~	~~0.903~~	~~188.8~~	~~170.5~~	~~65.5~~	~~-56~~	~~180~~
Lower Shell Long. Welds 20-714A/B	0.7048	0.9023	20910.15	18890.51	65.5	-56	198~~200~~
~~→ Using S/C Data[(f)]~~	~~0.708~~	~~0.903~~	~~223.9~~	~~202.2~~	~~65.5~~	~~-56~~	~~212~~
Circumferential Weld 11-714	3.523	1.3289	124.34	165.2	65.5	-56	175
~~→ Using S/C Data[(d)]~~	~~3.53~~	~~1.329~~	~~84.8~~	~~112.3~~	44	~~-56~~	~~101~~

Notes:

(a) For Master Curve method, CF is based on the measured irradiated ΔRT_{To} values. Calculations used three significant figures for chemistry values.

~~(a)~~(b) ΔRT_{PTS} = CF * FF. For Master Curve method, the Bias term is added to the multiplicative value for ΔRT_{PTS}.

~~(b) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.~~

(c) ~~RT_{PTS}~~ ART = ~~$RT_{NDT(U)}$~~ IRT + ΔRT_{PTS} + Margin ~~(°F)~~.

(d) Rounded-up numbers. For calculations refer to Reference 5.

~~(d) Based on credible St. Lucie Unit 1 surveillance data.~~

~~(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_A.~~

~~(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_A.~~

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Table 4.2-10

RT_{PTS} Calculation for Beltline Region Materials at Life extension (44.18~~5~~ EFPY)

Material	Fluence (x 10^{19} n/cm², E>1.0 MeV)	FF(d)	CF(a) (°F)	ΔRT_{PTS}(b~~e~~) (°F)	Margin (°F)	~~$RT_{NDT(U)}$~~ IRT~~(a)~~ (°F)	~~RT_{PTS}~~ ART(c~~b~~) (°F)
Intermediate Shell Plate B6607-1	5.87~~5~~	1.43	100.5	144~~3~~.0~~7~~	34	43	221
Intermediate Shell Plate B6607-2	5.87~~5~~	1.43	100.5	144~~3~~.0~~7~~	34	73	251
Lower Shell Plate B7203-2	5.87~~5~~	1.43	98.~~7~~65	141.3~~1~~	34	20	195
Lower Shell Plate B6903-1	5.87~~5~~	1.43	163~~47~~.2	~~2~~224~~210~~.0~~5~~	2~~3~~4	-5~~27~~	~~2~~261~~72~~
~~→ Using S/C Data(e)~~	~~5.85~~	~~1.43~~	~~149.2~~	~~213.4~~	~~34~~	~~27~~	~~274~~
Inter. Shell Long. Welds 19-714A/B	1.17~~43~~	1.04~~3~~	~~191.~~192.5~~7~~	~~197.~~201.1~~5~~	65.5	-56	~~207~~211
~~→ Using S/C Data(e)~~	~~1.13~~	~~1.03~~	~~188.8~~	~~194.5~~	~~65.5~~	~~-56~~	~~204~~
Lower Shell Long. Welds 20-714A/B	1.17~~43~~	1.04~~3~~	209~~10~~.1~~5~~ ~~223.9~~	218~~6~~.5~~8~~	65.5	-56	228~~6~~
~~→ Using S/C Data(f)~~	~~1.13~~	~~1.03~~		~~230.6~~	~~65.5~~	~~-56~~	~~240~~
Circumferential Weld 11-714	5.87~~2~~	1.43	124.4~~3~~	178~~7~~.2~~7~~	65.5	-56	188~~7~~
~~→ Using S/C Data(d)~~	~~5.82~~	~~1.43~~	~~84.8~~	~~121.3~~	44	~~-56~~	~~109~~

Notes:

(a) For Master Curve method, CF is based on the measured irradiated ΔRT_{To} values. Calculations used three significant figures for chemistry values.

(b) ΔRT_{PTS} = CF * FF. For Master Curve method, the Bias term is added to the multiplicative value for ΔRT_{PTS}.

(c) ART = IRT + ΔRT_{PTS} + Margin.

(d) Rounded-up numbers. For calculations refer to Reference 5.

~~(a) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.~~

~~(b) RT_{PTS} = $RT_{NDT(U)}$ + ΔRT_{PTS} + Margin (°F).~~

~~(c) ΔRT_{PTS} = CF * FF.~~

~~(d) Based on credible St. Lucie Unit 1 surveillance data.~~

~~(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_A.~~

~~(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_A.~~

ATTACHMENT D

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 297

The following is a list of the affected TS Bases pages.
These pages are included for information only.

Affected Page	Pending LAR
B 3/4 4-7	295
B 3/4 4-8	295

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The criticality limit curve includes the 10 CFR 50, Appendix G requirement that it be ≥ 40°F above the heatup curve and not less than the minimum permissible temperature for inservice hydrostatic testing. However, the PTLR criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.5, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E, provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

The pressure-temperature limit curves contained in the PTLR, are developed using ASME Code Cases N-629, N-631 and N-~~640~~641. One of the safety margins incorporated into the curves is the lower bound fracture toughness curve. The lower bound fracture toughness curves available in Appendix G to ASME Section XI use the reference stress intensity factor K_{IA}. The pressure-temperature limit curves based on Code Cases N-629, N-631 and N-~~640~~641 use the reference stress intensity factor K_{IC}. K_{IA} is a fracture toughness curve which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC} is a fracture toughness curve which is a lower bound on static fracture toughness only. The only change that is made when generating the revised pressure-temperature limits curve with K_{IC} is the lower bound fracture toughness curve selected. All other margins involved in the generation process remain unchanged. Since the heatup and cooldown process is a very slow one, with the fastest rate allowed being 100°F per hour, the rate of change of pressure and temperature is considered constant so that the stress is essentially constant. Both heatup and cooldown correspond to static loading, with regard to fracture toughness. The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{IA} should be used for the reactor pressure vessel is when a crack is running. This might happen during a pressurized thermal shock event, but not during heatup and cooldown. Therefore, the static toughness K_{IC} lower bound toughness is used to generate the pressure-temperature limit curves contained in the PTLR.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

This page was created by LAR 295.

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The overpressure protection system (OPPS) controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The maximum setpoint for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup meet the 10 CFR 50, Appendix G (including ASME Code Cases N-629, N-631 and N-~~640~~641) requirements during the OPPS MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only during shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires deactivating all but one charging pump and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the OPPS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve and, if needed, until the charging pump is actuated by SI.

ATTACHMENT E

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 297
Master Curve Methodology Exemption Requests

Requirement for Which Exemption is Requested

Pursuant to 10 CFR 50.12, "Specific Exemptions," the following is a request for exemption from certain requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," 10 CFR 50.54, "Conditions of Licenses," and 10 CFR 50, Appendix G, "Fracture Toughness Requirements." This exemption is requested to allow the application of ASME Code Case N-629, Code Case N-631 and Code Case N-641 and the use of WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", in determining the acceptable pressure and temperature limits, and overpressure protection system (OPPS) power operated relief valves (PORV) setpoints.

ASME Section XI Code Requirements

ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," Article A-2000, provides the service limits for pressure vessels and establishes the allowable vessel loading (internal pressure, external load, thermal stress) versus temperature. The code requirements are to maintain vessel operation conditions within Article A-2000 requirements.

Code Requirement from Which Exemption is Requested

Exemption is requested from 10 CFR 50, Appendix G, and ASME Section XI, Appendix G, requirements for reactor vessel pressure limits at low temperature.

Basis for Exemption Request

Current PORV setpoints produce operational constraints by limiting the pressure and temperature (P/T) range available to the operator to heatup and cooldown the plant. The "operating window" through which the operator must heat up and pressurize, or cool down and depressurize the reactor coolant system (RCS) is determined by Appendix G of Section XI, and the minimum required pressure for the reactor coolant pump (RCP) No. 1 seal, adjusted for OPPS overshoot and instrument uncertainties. Under the P/T requirements in Appendix G of ASME Section XI, OPPS can have significant impact on operation by limiting RCP operation at low temperatures. In addition, the operating pressure window imposed by OPPS becomes more and more restrictive with reactor

vessel service. Reducing this operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent OPPS actuation due to pressure surges associated with normal plant evolutions such as RCP starting and swapping operating charging pumps with the RCS in a water-solid condition.

The impact on the P/T limits and PORV setpoints has been evaluated due to increasing the service period to 22 Effective Full Power Years (EFPY) based on ASME Section XI, Appendix G, requirements. The results indicate that without the requested exemptions, OPPS would significantly restrict the ability to perform plant heatup and cooldown, create an unnecessary burden to plant operations, and challenge control of plant evolutions required with OPPS enabled.

Proposed Alternative

The use of ASME Code Cases N-629, N-631 and N-641 requirements, and WCAP-15315 allowance for reactor vessel pressure limits at low temperatures is proposed as an alternative to 10 CFR 50, Appendix G, and ASME Section XI, Appendix G, requirements.

Justification for Granting Relief

The application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. ASME Code Case N-641 recognizes the conservatism of the Appendix G curves and allows establishing a setpoint and enabling temperature which preserve an acceptable margin of safety while maintaining an adequate heatup and cooldown window. Increasing the heatup and cooldown "operating window" will reduce the likelihood of actuation of protection system pressure relieving devices and challenge to the RCP seals. The guidelines developed by FENOC for the OPPS enabling temperature and relief valve setpoint provide the same range of margin against vessel failure for conditions where experience indicates these events occur, as ASME Section III and Section XI, Appendix G provides for normal heatup and cooldown conditions. Codes Case N-641 preserves the current ASME Section XI, Appendix G approach for developing pressure/temperature limits by postulating circumferentially oriented reference flaw in a circumferential weld instead of an axially oriented reference flaw. These limits do not significantly change the likelihood of vessel failure associated with the normal heatup and cooldown. Additionally, applying ASME Code Case N-641, with the flange requirement of 10 CFR 50 Appendix G, would eliminate the benefit of ASME Code Case N-641 at temperatures below the Initial RT_{NDT} of the vessel and head flange, plus 120°F, since 10 CFR 50 Appendix G requires the vessel pressure to be 20% below

the hydrostatic-test pressure until the vessel is 120°F above the Initial RT_{NDT} of the flange materials. Therefore, establishing the OPPS enabling temperature and relief valve setpoint in accordance with ASME Code Case N-641 criteria and removal of the flange requirement, as documented in WCAP-15315, satisfy the underlying purpose of 10 CFR 50.60.

The underlying purpose of 10 CFR 50.60, Appendix G, is to establish fracture toughness requirements for ferritic material of pressure retaining components of the reactor coolant pressure boundary. These requirements provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences, to which the pressure boundary may be subjected to over its service lifetime. Section IV.A.2 of Appendix G requires that the reactor vessel be operated with P/T limits at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of ASME Section XI.

Appendix G of Section XI of the ASME Code requires the P/T limits be calculated: a) using a safety factor of 2 on the principal membrane (pressure) stresses; b) margin added to the reactor vessel RT_{NDT} in accordance with Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials;" c) assuming a flaw at the surface with a depth of 1/4 of the vessel wall thickness and a length of 6 times its depth; and d) using conservative fracture toughness curves that is based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the reactor vessel material.

The Pressure-Temperature Limit Curves determined using ASME Code Case N-641 are also less restrictive than the requirements of 10 CFR 50 Appendix G, Section IV.A.2.

Title 10, Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. The regulation states that the metal temperature of 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 preservice hydrostatic test pressure of 3106 psig. However, in accordance with WCAP-15315, this requirement is no longer necessary when using the methodology of Code Case N-641. Therefore, the BVPS Unit 1 P/T Limits Curves were generated without flange requirements included.

WCAP-15315 and Section XI of the ASME Code for developing Pressure-Temperature Limit Curves indicate that there is significant margin between the applied stress intensity factor and the fracture toughness at virtually all crack depths when using the K_{IC} toughness. 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 can be concluded that the integrity of the closure head/flange region is not a concern for any of the operating plants using the K_{IC} toughness.

Furthermore, there are no known mechanisms of degradation for this region, other than 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.

An exemption to 10 CFR 50.54 is requested to allow the use Code Cases N-629 and N-631. The use of the Master Curve methodology to determine the RT_{PTS} value during the expected licensed operating life of the plant is based on these two Code Cases. Since this fracture toughness methodology will be used for establishing RT_{PTS} and the P/T curves, future surveillance must be focused on measurement of fracture toughness instead of current Charpy V-notch surveillance capsule testing. The addition of a supplemental fracture toughness capsule in the BVPS Unit 2 reactor pressure vessel, containing BVPS Unit 1 reactor vessel beltline materials, will require approval of a type of integrated surveillance program per 10 CFR 50, Appendix H. The exemption to Appendix H allows the surveillance program to be modified during the current License period to ensure effective surveillance monitoring using fracture toughness specimens through end-of-life.

Based on aforementioned information, BVPS Unit 1 requests an exemption to 10 CFR 50.60 and 10 CFR 50.54, based on American Society of Mechanical Engineers (ASME) Code Case N 629, Code Case N-631, and Code Case N-641 and WCAP-15315.

L-02-004 Enclosure 1

Proprietary Information Notice, Copyright Notice and a Westinghouse application for withholding proprietary information.

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

January 7, 2002

CAW-02-1506

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Beaver Valley Unit 1 FirstEnergy Operating Company Overpressure Protection System Setpoints for Master Curve", Revision 1(Proprietary), August 2001

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-02-1506 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by FirstEnergy Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-01-1457 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: D. Holland, NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

CAW-02-1506

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 9th day
of January, 2002, ~~2001~~

Kay E. Gongaware
Notary Public

Notarial Seal
Kay E. Gongaware, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Feb. 7, 2005
Member, Pennsylvania Association of Notaries

KAY E. GONGAWARE NOTARY PUBLIC
COMMONWEALTH OF PENNSYLVANIA

/cm/0253S.doc

(1) I am Manager, Regulatory and Licensing Engineering, in Nuclear Services, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

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

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company Overpressure Protection System Setpoints for Master Curve", Revision 1 being transmitted by FENOC letter and Application for Withholding Proprietary Information from Public Disclosure, **to the Document Control Desk, Attention**

Mr. Samuel J. Collins. The proprietary information as submitted for use by FirstEnergy Nuclear Operating Company for the Beaver Valley Unit 1 is expected to be applicable in other licensee submittals in response to certain NRC requirements.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the methods for determination of equipment operability and acceptable calibration of the noted OPPS setpoint development.

(b) Provide the specific design information related to the parameters that are considered for the OPPS setpoint development.

(c) Assist the customer to obtain NRC approval

Further this information has substantial commercial value as follows:

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

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

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