Westinghouse Non-Proprietary Class 3

WCAP-14040-A
Revision 4

May 2004

Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves

Westinghouse

NRC SAFETY EVALUATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 27, 2004

Mr. Gordon Bischoff, Manager
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P.O. Box 355
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WOG PROJECT OFF

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES" (TAC NO. MB5754)

Dear Mr. Bischoff:

On May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," to the staff for review. On February 2, 2004, an NRC draft safety evaluation (SE) regarding our approval of WCAP-14040, Revision 3, was provided for your review and comments. By letter dated February 18, 2004, the WOG commented on the draft SE by indicating that the actual provision number of GL 96-03 should be provided in Sections 3.0 and 4.0 of the SE. In addition, minor editorial comments were provided by the WOG. The staff has incorporated the WOG's suggested comments into the final SE enclosed with this letter.

The staff has found that WCAP-14040, Revision 3, is acceptable for referencing in licensing applications for Westinghouse-designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the enclosed SE. The SE defines the basis for acceptance of the report.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the WOG publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, draft SE comments, and original report pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions in this letter, that the TR is acceptable, is invalidated, the WOG and/or the licensees referencing the TR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the TR without revision of the respective documentation.

Sincerely,

Robert A Gramm for/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
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Project No. 694

Enclosure: Safety Evaluation

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UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMIT CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P-T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted revisions to the TR for NRC staff review and approval by letter dated October 20, 2003.

On February 2, 2004, an NRC draft safety evaluation (SE) regarding our approval of WCAP-14040, Revision 3, was provided for your review and comments. By letter dated February 18, 2004, the WOG commented on the draft SE by indicating that the actual provision number of GL 96-03 should be provided in Sections 3.0 and 4.0 of the SE. In addition, minor editorial comments were provided by the WOG. The staff has incorporated the WOG's comments.

2.0 REGULATORY EVALUATION

Four specific topics are addressed in the context of the development of a PTLR methodology: (1) the calculation of neutron fluences for the RPV and RPV surveillance capsules; (2) the evaluation of RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these RPV material properties and the establishment of cold overpressure mitigating system (COMS) setpoints to protect the RPV from brittle failure; and (4) the development of an RPV material surveillance program to monitor changes in RPV material properties due to radiation. Regulatory requirements related to the four topics noted above are addressed in Appendices G and H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50). Appendix G to 10 CFR Part 50 provides requirements related to RPV P-T limit development and directly or indirectly addresses topics (1) through (3) above. Appendix H to 10 CFR Part 50 defines regulatory requirements related to RPV material surveillance programs and addresses topic (4) above.

For the staff's review of WCAP-14040, Revision 3, several additional guidance documents were used. NRC Standard Review Plan (SRP) Sections 5.2.2, "Overpressure Protection," 5.3.1, "Reactor Vessel Materials," and 5.3.2, "Pressure-Temperature Limits," provide specific review guidance related to RPV material property determination, P-T limit development, and COMS performance. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes analysis procedures acceptable to the NRC staff for the purpose of assessing RPV material property changes due to radiation. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," addresses NRC staff expectations for an acceptable fluence calculation methodology. American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides guidance on the establishment of RPV material surveillance programs and editions of ASTM E 185 are incorporated by reference into Appendix H to 10 CFR Part 50. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G provides specific requirements regarding the development of P-T limit curves.

Finally, specific guidance regarding topics and the level of detail to which they must be addressed as part of an acceptable PTLR methodology is given in GL 96-03.

3.0 TECHNICAL EVALUATION

The technical requirements to be addressed in an acceptable PTLR methodology are provided under the column heading "Minimum Requirements to be Included in Methodology" in the table entitled "Requirements for Methodology and PTLR" in Attachment 1 to GL 96-03. Summarized versions of the seven requirements are given below, along with the staff's technical evaluation of information in WCAP-14040, Revision 3, related to each requirement.

Requirement 1: Regarding the reactor vessel material surveillance program, the methodology should briefly describe the surveillance program. The methodology should clearly reference the requirements of Appendix H to 10 CFR Part 50.

The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.

Requirement 2: Regarding the calculation of RPV materials' adjusted reference temperatures (ART) values, the methodology should describe the method for calculating material ART values using RG 1.99, Revision 2.

Information regarding how material ARTs are to be determined within the WCAP-14040, Revision 3, PTLR methodology is provided in Sections 2.3 and 2.4 of the TR. In Section 2.3, the determination of initial, unirradiated material properties from Charpy V-notch impact tests and/or nil-ductility drop weight tests is clearly defined. The methodology specified in Section 2.3 accurately incorporates the guidance found in ASME Code Section III, paragraph NB-2331 and additional information in SRP Section 5.3.1.

In Section 2.4 of the TR, the determination of changes in material properties due to irradiation is addressed, along with the determination of margins necessary to account for uncertainties in initial properties and irradiation damage assessment. The methodology specified in Section 2.4 accurately incorporates the guidance found in RG 1.99, Revision 2.

The NRC staff, therefore, determined that the methodology described for determining material ART values in WCAP-14040, Revision 3, was consistent with the guidance provided in the ASME Code, SRP Section 5.3.1, and RG 1.99, Revision 2, and was, therefore, acceptable.

Requirement 3: Regarding the development of RPV P-T limit curves, the methodology should describe the application of fracture mechanics-based calculations in constructing P-T limit curves based on the provisions of Appendix G to Section XI of the ASME Code and SRP Section 5.3.2.

Basic and optional elements of the methodology for RPV P-T limit curve development in WCAP-14040, Revision 3, are given in Sections 2.5, 2.6, 2.7, 2.8, and Appendix A of the TR.

In Section 2.5 of the TR, the fracture toughness-based guidelines from Appendix G to Section XI of the ASME Code are specified (based on the 1995 Edition through 1996 Addenda of the ASME Code). Notably, specific reference is made to the use of: (1) the ASME Code lower bound dynamic crack initiation/crack arrest (K_{IA}) fracture toughness curve; (2) the use of a postulated flaw that has a depth of one-quarter of the wall thickness and a 6:1 aspect ratio; and (3) the use of a structural factor of 2 on primary membrane stress intensities (K_{IM}) when evaluating normal heatup and cooldown and a structural factor of 1.5 on K_{IM} when evaluating hydrostatic/leak test conditions.

Optional guidelines for P-T limit curve development are also addressed in WCAP-14040, Revision 3. The option of using the ASME Code static crack initiation fracture toughness curve (K_{IC}), as given in ASME Code Case N-640, is addressed in Sections 2.5 and 2.8. The option of using ASME Code Case N-588, which enables the postulation of a circumferentially-oriented flaw (with appropriate stress magnification factors) when evaluating a circumferential weld, is addressed in Section 2.8. WCAP-14040, Revision 3, notes, however, that licensee use of the provisions of either ASME Code Case N-640 or N-588 requires, in accordance with 10 CFR 50.60(b), an exemption if the provisions of the Code Case are not contained in the edition of the ASME Code included in a facility's licensing basis. Appendix A to WCAP-14040, Revision 3, provides additional details regarding the application of optional ASME Code Cases and includes copies of ASME Code Case N-588, N-640, and N-641 (which effectively combines the provisions of N-588 and N-640 into a single Code Case).

A detailed discussion of the calculational methodology for P-T limit curve generation is given in Section 2.6 of the TR. Specific equations are given for the determination of primary membrane stresses due to internal pressure and membrane and bending stresses due to thermal gradients. Equations related to the generation of P-T limit curves for steady-state conditions, finite heatup rates, finite cooldown rates, and hydrostatic/leak test conditions are given. The equations given in WCAP-14040, Revision 3, are equivalent to those provided in Section XI of the ASME Code and consistent with the guidance given in SRP Section 5.3.2.

Therefore, the NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P-T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P-T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

Requirement 4: Regarding the development of RPV P-T limit curves, the methodology should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied when constructing P-T limit curves.

Minimum temperature requirements regarding the material in the highly stressed region of the RPV flange are given in Appendix G to 10 CFR Part 50. Information provided in Sections 2.9 and 2.10 of the TR addresses the incorporation of minimum temperature requirements into the development of P-T limit curves. In Section 2.9, the 10 CFR Part 50, Appendix G requirements are cited. WCAP-14040, Revision 3, goes on to note that there is an effort underway to revise or eliminate these requirements based on information contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." However, WCAP-14040, Revision 3, states that until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

WCAP-14040, Revision 3, provides supplemental information in Section 2.10 regarding the establishment of RPV boltup temperature, specifically that the minimum boltup temperature should be 60 °F or equal to the highest material reference temperature in the highly stressed RPV flange region, whichever is higher (i.e., more conservative). Although no specific requirements related to boltup temperature are provided in Appendix G to 10 CFR Part 50, the information in WCAP-14040, Revision 3, is consistent with other, related requirements in Appendix G to 10 CFR Part 50 and in Appendix G to Section XI of the ASME Code.

The NRC staff concludes that the methodology specified in WCAP-14040, Revision 3, addresses RPV minimum temperature requirements in a way which is consistent with Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code and is, therefore, acceptable.

Requirement 5: Regarding the calculation of RPV materials' ARTs, the methodology should describe how the data from multiple surveillance capsules may be used in ART calculations.

Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.

The NRC staff has reviewed the information in Section 2.4 of the TR and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Requirement 6: Regarding the calculation of the neutron fluence, the methodololgy should describe how the neutron fluence is calculated.

Neutron Fluence Methodology

WCAP-14040, Revision 3, includes a revised Section 2.2. The revised section includes plant-specific transport calculations and the validity of the calculations. For the neutron transport calculations, the applicant is using the two-dimensional discrete ordinates code, DORT (Reference 1) with the BUGLE-96 cross section library (Reference 2). Approximations include a P_5 Legendre expansion for anisotropic scattering and a S_{16} order of angular quadrature. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Two dimensional flux solutions $\Phi(r, \theta, z)$ are constructed using (r,θ) and (r,z) distributions. Extreme cases, with respect to power distribution arising from part-length fuel assemblies, use the three-dimensional TORT Code (Reference 1) with the BUGLE-96 cross section library. Source distribution is obtained from a burn-up weighted average of the power distributions of individual fuel cycles. The method accounts for source energy spectral effects and neutrons/fission due to burnup by tracking the concentration of U-235, U-238, Pu-239, Pu-240, and Pu-241. Mesh spacing accounts for flux gradients and material interfaces.

The proposed methodology, as outlined above, adheres to the guidance of RG 1.190, and therefore, is acceptable.

Validation of Transport Calculations

The Westinghouse validation is structured in four parts:

- comparison to pool critical assembly (PCA) simulator results (Reference 3),
- comparison to calculations in the H. B. Robinson benchmark (Reference 4),
- comparison to a measurement database from pressurized water reactor (PWR) surveillance capsules, and
- an analytical sensitivity study addressing the uncertainty components of the transport calculations.

Comparisons of calculated results to the corresponding PCA measured quantities establish the adequacy of the basic transport calculation and the associated cross sections. Comparison to the H.B. Robinson benchmark addresses uncertainties related to the method and generally to the neutron exposure. Comparisons to the PWR database provide an indication of the presence of a bias and of the uncertainty of the calculated value with respect to the corresponding measured values. Finally, the analytical sensitivity study validates the overall uncertainties whether from the methodology or the lack of precise knowledge of the input parameters.

Comparison of the measured data to the calculations was performed on the basis of measured/calculated (M/C) ratios, and with best estimate values calculated using least squares adjusted measured values. The least squares adjustment is based on weighing individual measurements based on spectral coverage. Comparisons are done before and after spectral adjustments. This method is addressed in RG 1.190, as well as in the ASTM Standard E944-96.

The NRC staff requested that the WOG address the completeness of its database. By letter dated October 20, 2003, the WOG responded by indicating that all of the surveillance capsules analyzed with the proposed methodology (DORT and BUGLE-96) are included in the database. The NRC staff found the response acceptable.

The NRC staff concludes that the proposed benchmarking methodology adheres to the guidance in RG 1.190 and to ASTM standards, and therefore, is acceptable.

Requirement 7: Regarding the low temperature overpressure protection/cold overpressure mitigating system, the lift setting limits for the power operated relief valves should be developed using NRC-approved methodologies.

The method in this section is identical to the existing method in the approved Revision 2 of WCAP-14040. The thermal hydraulics analysis for the mass and heat input transients is using the same specialized version of LOFTRAN, which was approved in Revision 2.

The cold overpressure mitigating system is the same as in the approved version, and therefore, the NRC staff finds it acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the information provided in WCAP-14040, Revision 3, related to the requirements of GL 96-03, as cited in Section 3.0 of this SE, and finds WCAP-14040, Revision 3, to be acceptable for referencing as a PTLR methodology, subject to the following conditions:

a. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the RPV material surveillance program.

b. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

c. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

5.0 REFERENCES

1. "DOORS 3.1, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996.

2. "Bugle-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," RSICL Data Library Collection DLC-185, Oak Ridge National Laboratory, March 1996.

3. NUREG/CR-6454 (ORNL/TM-13205), "Pool Critical Assembly Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, July 1997.

4. NUREG/CR-6453 (ORNL/TM-13204), "H.B. Robinson Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, February 1998.

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Date: February 27, 2004

WCAP-14040-A
Revision 4

Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves

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1.0 INTRODUCTION

1.1 BACKGROUND

The concept of a Pressure and Temperature Limits Report (PTLR) was introduced into the Technical Specifications during the development of NUREG 1431[1], Standard Technical Specifications for Westinghouse PWRs and is consistent with the philosophy of NRC Generic Letter 88-16[2]. The PTLR is similar to the Core Operating Limits Report (COLR), which is currently licensed for several plants and also contained in NUREG 1431. The COLR contains core related limit values which may change from cycle to cycle as they are related to a cycle specific core design. In the same way, a PTLR contains reactor vessel material related limits which may change every fluence cycle as they are related to reactor vessel material and strength. Implementation of the PTLR will allow licensees to relocate their RCS heatup and cooldown curves and COMS setpoints currently contained in the Technical Specifications to the PTLR. Additionally, the Vessel Fluence and Materials tables contained in the Technical Specifications or Bases can be relocated to licensee controlled documents. This process will allow changes to these tables, figures and values to be made without making a License Amendment Request (LAR). These figures are typically revised due to changes in the nil ductility reference temperature (RT_{NDT}), regulations and surveillance capsule withdrawal.

1.2 PURPOSE OF TOPICAL REPORT

In order to implement the PTLR, the analytical methods used to develop the pressure and temperature limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the Technical Specifications. The purpose of this report is to provide the current Westinghouse methodology for developing the RCS heatup and cooldown curves and COMS setpoints. When approved by the NRC, this methodology may be referenced by licensees to implement the PTLR.

This topical report does not provide all of the methodologies which can be used to develop RCS heatup and cooldown curves and COMS setpoints, but rather methodologies that can be referenced by licensees when approved by the NRC to license the PTLR concept.

1.3 CONTENT OF TOPICAL REPORT

This report contains the methodology used to develop the RCS heatup and cooldown curves in Section 2.0 and the methodology used to develop the COMS setpoints in Section 3.0. The methodology used to develop the COMS enable temperature is also discussed in Section 3.0.

2.0 PRESSURE-TEMPERATURE LIMIT CURVES

2.1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the beltline region of the reactor vessel. The most limiting RT_{NDT} of the material in the core (beltline) region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}). The unirradiated RT_{NDT} is defined as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (both normal to the major working direction) minus 60ºF.

RT_{NDT} increases as the material is exposed to fast-neutron irradiation. 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 original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced 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)[3]. Regulatory Guide 1.99, Revision 2, is used for the calculation of adjusted reference temperature (ART) values (irradiated RT_{NDT} with margins for uncertainties) at 1/4t and 3/4t locations. "t" is the thickness of the vessel at the beltline region measured from the clad/base metal interface (Note, thickness of cladding is neglected as specified in the ASME Code, Section III, paragraph NB-3122.3). Using the adjusted reference temperature values, pressure-temperature limit curves are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by Appendix G, Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code[5]. The procedure for establishing the pressure-temperature limits is entirely deterministic. The conservatisms included in the limits are (but not limited to):

- An assumed flaw in the wall of the reactor vessel has a depth equal to 1/4 of the thickness of the vessel wall and a length equal to 1-1/2 times the vessel wall thickness,
- A factor of 2 is applied to the membrane stress intensity factor (K_{IM}),
- 2-sigma margins are applied in determining the adjusted reference temperature (ART), and
- The limiting toughness is based upon a reference value [K_{Ia}, which is a lower bound of the dynamic crack initiation or arrest toughnesses, or K_{Ic}, which is a lower bound of static feature toughness].

This section describes the methodology used by Westinghouse to develop the allowable pressure-temperature relationships for normal plant heatup and cooldown rates that are included in the Pressure-Temperature Limits Report (PTLR). First, the methodology describing how the neutron fluence is calculated for the reactor vessel beltline materials is provided. Next, sections describing fracture toughness properties, adjusted reference temperature calculation, criteria for allowable pressure-temperature relationships, and pressure-temperature curve generation are provided.

2.2 NEUTRON FLUENCE METHODOLOGY

The methodology used to provide neutron exposure evaluations for the reactor pressure vessel is based on the requirements provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."(6). The vessel exposure projections are based on the results of plant specific neutron transport calculations that are validated by benchmarking of the analytical approach, comparison with industry wide power reactor data bases, and finally, by comparison to plant specific surveillance capsule and reactor cavity dosimetry data. In the validation process, the measurement data are used solely to confirm the accuracy of the transport calculations. The measurements are not used in any way to modify the results of the transport calculations.

2.2.1 Plant Specific Transport Calculations

In the application of the methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, plant specific forward transport calculations are carried out on a fuel cycle specific basis using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)]/[\phi(r)]$$

where:

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

All of the transport calculations are carried out using the DORT discrete ordinates code Version 3.1(7) and the BUGLE-96 cross-section library[11]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering is treated with a P_5 legendre expansion and the angular discretization is modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures are treated on a fuel cycle specific basis. The synthesis procedure combining the $\phi(r,\theta)$, $\phi(r,z)$, and $\phi(r)$ transport solutions into the three-dimensional flux/fluence maps within the reactor geometry is accomplished by post-processing the output files generated by the $[r,\theta]$, $[r,z]$, and $[r]$ DORT calculations.

In some extreme cases where part length poisons or shielded fuel assemblies have been inserted into the reactor core to reduce the fluence locally in the vicinity of key vessel materials, the calculational approach may be modified to use either a multi-channel synthesis approach or a fully three-dimensional technique. For the full three-dimensional analysis, the TORT(7) three-dimensional discrete ordinates transport code is used in conjunction with either the BUGLE-96 ENDF/B-VI based library to provide a complete solution without recourse to the use of flux synthesis techniques.

In developing an analytical model of the reactor geometry, nominal design dimensions are normally employed for the various structural components. In some cases as-built dimensions are available; and, in those instances, the more accurate as-built data are used for model development. However, for the most part, as built dimensions of the components in the beltline region of the reactor are not available, thus, dictating the use of design dimensions. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor are normally taken to be representative of full power operating conditions. The reactor core itself is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The spatial mesh description used in the transport models depends on the overall size of the reactor and on the complexity required to model the core periphery, the in-vessel surveillance capsules, and the details of the reactor cavity. Mesh sizes are chosen to assure that proper convergence of the inner iterations is achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations is set at a value of 0.001.

The mesh selection process results in a smaller spatial mesh in regions exhibiting steep gradients, in material zones of high cross-section (Σ_t), and at material interfaces. In the modeling of in-vessel surveillance capsules, a minimum set of 3 radial by 3 azimuthal mesh are employed within the test specimen array to assure that sufficient information is produced for use in the assessment of fluence gradients within the materials test specimens, as well as in the determination of gradient corrections for neutron sensors. Additional radial and azimuthal mesh are employed to model the capsule structure surrounding the materials test specimen array. In modeling the stainless steel baffle region at the periphery of the core, a relatively fine spatial mesh is required to adequately describe this rectilinear component in r,θ geometry. In performing this x,y to r,θ transition, care is taken to preserve both the thickness and volume of the steel region in order to accurately address the shielding effectiveness of the component.

The spatial variation of the neutron source is generally obtained from a burnup weighted average of the respective power distributions from individual fuel cycles. These spatial distributions include pinwise gradients for all fuel assemblies located at the periphery of the core and typically include a uniform or flat distribution for fuel assemblies interior to the core. The spatial component of the neutron source is transposed from x,y to [r,θ], [r,z], and [r] geometry by overlaying the mesh schematic to be used in the transport calculation on the pin by pin array and then computing the appropriate relative source applicable to each spatial interval within the reactor core.

These x,y to [r,θ], [r,z], and [r] transpositions are accomplished by first defining a fine mesh working array. The sizes of the fine mesh are usually chosen so that there is at least a 10x10 array of fine mesh over the area of each fuel pin at the core periphery. The coordinates of the center of each fine mesh interval and its associated relative source strength are assigned to the fine mesh based on the pin that is coincident with the center of the fine mesh. In the limit as the sizes of the fine mesh approach zero, this technique becomes an exact transformation.

Each space mesh in the transport geometry is checked to determine if it lies totally within the area of a particular fine working mesh. If it does, the relative source of that fine mesh is assigned to the transport space mesh. If, on the other hand, the transport space mesh covers a part of one or more fine mesh, then the relative source assigned to the transport mesh is determined by an area weighting process as follows:

$$P_m = \frac{\sum_i A_i P_i}{\sum_i A_i}$$

where:

P_m = the relative source assigned to transport mesh m.

A_i = the area of fine working mesh i within transport mesh m.

P_i = the relative source within fine working mesh i.

The energy distribution of the source is determined on a fuel assembly specific basis by selecting a fuel assembly burnup representative of conditions averaged over each fuel cycle and an initial enrichment characteristic for each assembly. From this average burnup and initial enrichment, a fission split by isotope including ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu is derived; and, from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum are determined for each fuel assembly. These composite values are then combined with the spatial distribution to produce the overall absolute neutron source for use in the transport calculations.

2.2.2 Validation of the Transport Calculations

The validation of the methodology described in Section 2.2.1 is based on the guidance provided in Regulatory Guide 1.190. In particular, the validation consists of the following stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL)[(12)].

2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment[(22)].

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant specific transport calculations used in the exposure assessments.

4. Comparisons of calculations with a measurements data base obtained from a large number of surveillance capsules withdrawn from a variety of pressurized water reactors.

At each subsequent application of the methodology, comparisons are made with plant specific dosimetry results to demonstrate that the plant specific transport calculations are consistent with the uncertainties derived from the methods qualification.

The first stage of the methods validation addresses the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This stage, however, does not test the accuracy of commercial core neutron source calculations nor does it address uncertainties in operational or geometric variables that impact power reactor calculations. The second stage of the validation addresses uncertainties that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations. The third stage of the validation identifies the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant specific parameters. The overall calculational uncertainty is established from the results of these three stages of the validation process.

The following summarizes the uncertainties determined from the results of the first three stages of the validation process:

PCA Benchmark Comparisons	3%
H. B. Robinson Benchmark Comparisons	3%
Analytical Sensitivity Studies	11%
Internals Dimensions	3%
Vessel Inner Radius	5%
Water Temperature	4%
Peripheral Assembly Source Strength	5%
Axial Power Distribution	5%
Peripheral Assembly Burnup	2%
Spatial Distribution of the Source	4%
Other Factors	5%

The category designated "Other Factors" is intended to attribute an additional uncertainty to other geometrical or operational variables that individually have an insignificant impact on the overall uncertainty, but collectively should be accounted for in the assessment.

The uncertainty components tabulated above represent percent uncertainty at the 1σ level. In the tabulation, the net uncertainty of 11% from the analytical sensitivity studies has been broken down into its individual components. When the four uncertainty values listed above (3%, 3%, 11%, and 5%) are combined in quadrature, the resultant overall 1σ calculational uncertainty is estimated to be 13%.

To date the methodology described in Section 2.2.1 coupled with the BUGLE-96 cross-section library has been used in the evaluation of dosimetry sets from 82 surveillance capsules from 23 pressurized water reactors. These capsule withdrawals included 2-5 capsules from individual reactors. The comparisons of the plant specific calculations with the results of the capsule dosimetry are used to further validate the calculational methodology within the context of a 1σ calculational uncertainty of 13%.

This 82 capsule data base includes all surveillance capsule dosimetry sets analyzed by Westinghouse using the Bugle-96 cross-section library and the synthesis approach described in Section 2.2.1. No surveillance capsule dosimetry sets were excluded from the M/C data base. As additional capsules are analyzed using the synthesis approach with the BUGLE-96 cross-section library the M/C comparisons will be added to the database.

The comparisons between the plant specific calculations and the data base measurements are provided on two levels. In the first instance, measurement to calculation (M/C) ratios for each fast neutron sensor reaction rate from the surveillance capsule irradiations are listed. This tabulation provides a direct comparison, on an absolute basis, of measurement and calculation. The results of this comparison for the surveillance capsule data base are as follows:

REACTION	M/C	STD DEV
$^{63}Cu(n\alpha)^{60}Co$	1.09	7.9%
$^{54}Fe(n,p)^{54}Mn$	0.99	8.4%
$^{58}Ni(n,p)^{58}Co$	0.99	8.9%
$^{238}U(n,f)^{137}Cs$	1.01	11.8%
$^{237}Np(n,f)^{137}Cs$	1.06	11.3%
Linear Average	1.03	9.8%

These comparisons show that the calculations and measurements for the surveillance capsule data base fall well within the 13% calculational uncertainty for all of the fast neutron reactions.

The second comparison of calculations with the data base is based on the least squares adjustment of the individual surveillance capsule data sets. The least squares adjustment procedure provides a weighting of the individual sensor measurements based on spectral coverage and allows a comparison of the neutron flux (E > 1.0 MeV) before and after adjustment. The neutron flux/fluence (E > 1.0 MeV) is the primary parameter of interest in the overall pressure vessel exposure evaluations.

The least squares evaluations of the 82 surveillance capsule dosimetry sets followed the guidance provided in Section 1.4.2 of Regulatory Guide 1.190 and in ASTM Standard E944-96, “Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance.”

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

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

For the data base comparisons, the calculated neutron spectra were obtained from the results of plant specific neutron transport calculations applicable to each of the 82 surveillance capsules. The sensor reaction rates and dosimetry cross-sections were the same as those used in the direct M/C comparisons noted above.

The results of this latter comparison expressed in terms of the ratio of adjusted flux to calculated flux (A/C) are summarized as follows for the 82 capsule data base:

PARAMETER	A/C	STD DEV
$\phi(E > 1.0\ MeV)$	1.00	7.3%

As with the comparisons based on the linear average of reaction rate M/C ratios, the comparisons of the least squares adjusted results with the plant specific transport calculations demonstrate that the calculated results are essentially unbiased with an uncertainty well within the 20% criterion established in Regulatory Guide 1.190.

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50(4), as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code(8). These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")(9) of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness guidelines in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

2.4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The adjusted reference temperature (ART) for each material in the beltline region is calculated in accordance with Regulatory Guide 1.99, Revision 2[3]. The most limiting ART values (i.e., highest value at 1/4t and 3/4t locations) are used in determining the pressure-temperature limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (2.4\text{-}1)$$

IRT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8] and calculated per Section 2.3. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for that class of material can be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the shift in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF\ f^{\,(0.28 - 0.10 \log f)} \qquad (2.4\text{-}2)$$

CF (°F) is the chemistry factor and is a function of copper and nickel content. CF is given in Table 1 of Reference 3 for weld metal and in Table 2 in Reference 3 for base metal (Position 1.1 of Regulatory Guide 1.99, Revision 2). In Tables 1 and 2 of Reference 3 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in Regulatory Guide 1.99, Revision 2, Paragraph B.4) become available they may be used to calculate the chemistry factor per Position 2.1 of Regulatory Guide 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n} [A_i\ f_i^{\,(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n} [f_i^{\,(0.28-0.10\log f_i)}]^2} \qquad (2.4\text{-}3)$$

Where "n" is the number of surveillance data points, "A_i" is the measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, "f_i." Where "f_i" is the fluence for each surveillance data point.

If Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a higher value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a lower value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4t), the following formula is used to attenuate the fast neutron fluence (E> 1 MeV) at the specified depth.

$$f = f_{surface}\, e^{(-0.24x)} \quad (2.4\text{-}4)$$

where $f_{surface}$ 10^{19} n/cm^2, E > 1 MeV) is the value, calculated per Section 2.2, of the neutron fluence at the base metal surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (2.4-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by equation (2.4-2) plus two standard deviations ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be submitted for NRC review and approval to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by Appendix G to 10 CFR Part 50[(4)]. Margin is calculated by the following equation:

$$\text{Margin} = 2\,[(\sigma_I^2 + \sigma_\Delta^2)]^{0.5} \quad (2.4\text{-}5)$$

σ_I, is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_I, is estimated from the precision of the test method ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per Regulatory Guide 1.99, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

2.5 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME Code requirements[(5)] for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates specify that the total stress intensity factor, K_i, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, the fracture toughness for the metal temperature at that time. Two values of fracture toughness may be used, K_{Ia} or K_{Ic}.

K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G, to Section XI of the ASME Code (1995 Edition through the 1996 Addenda). (Note that in Appendix G, to Section III of the ASME Code, the reference fracture toughness is denoted as K_{IR}, whereas in Appendix G of Section XI, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.) The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \exp[0.0145\,(T - RT_{NDT} + 160)] \quad (2.5\text{-}1)$$

where,

K_{Ia} = lower bound of dynamic and crack arrest toughness as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RT_{NDT} is the adjusted reference temperature (ART) of Section 2.4.

K_{Ic} is also obtained from Section XI of the ASME Code, for example in Appendix A, and is a lower bound of static fracture toughness. Since heatup and cooldown is a slow process, static properties are appropriate. The K_{Ic} curve is given by the following expression:

$$K_{Ic} = 33.20 + 20.734 \exp [0.0200 (T - RT_{NDT})] \tag{2.5-2}$$

The use of the K_{Ic} curve (Section XI, Appendix A) as a basis for developing P-T limit curves is currently contained in ASME Code Case N640. Use of the K_{Ic} fracture toughness will yield less limiting P-T curves, which is clearly a benefit.

However, the use of Code Case 640 presently includes a restriction on the setpoints for the Cold Overpressure Mitigation System (COMS). This maximum pressure for the COMS system is 100% of the pressure allowed by the P-T limit curves. This essentially disallows the use of Code Case N514 in these circumstances, meaning that the COMS system must protect to the actual P-T limit curve, rather than 110 percent, as allowed by Code Case N514.

The governing equation for generating pressure-temperature limit curves is defined in Appendix G of the ASME Code[5] as follows:

$$C\, K_{IM} + K_{It} < \text{Reference Fracture Toughness} \tag{2.5-3}$$

where,

K_{IM} = stress intensity factor caused by membrane (pressure) stress,

K_{It} = stress intensity factor caused by the thermal gradients through the vessel wall,

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

C = 1.5 for hydrostatic and leak test conditions when the reactor core is not critical

Reference Fracture Toughness = K_{Ia} or K_{Ic}, as discussed above

(Note: K_{It} is set to zero for hydrostatic and leak test calculations since these tests are performed at isothermal conditions).

At specific times during the heatup or cooldown transient, the reference fracture toughness is determined by the metal temperature at the tip of the postulated flaw (the postulated flaw has a depth of one-fourth of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, paragraph G-2120), the appropriate value for RT_{NDT} at the same location, and the reference fracture toughness equation (2.5-1 or 2.5-2). The thermal stresses resulting from the temperature gradients through the vessel wall and the corresponding (thermal) stress intensity factor, K_{It}, for the reference flaw are calculated as described in Section 2.6. From Equation (2.5-3), the limiting pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated as described in Section 2.6.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference 1/4t (t = reactor vessel wall thickness) flaw of Appendix G, Section XI to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the vessel wall because the thermal gradients that increase with increasing cooldown rates produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Allowable pressure-temperature curves are generated for steady-state (zero rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4t vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the temperature difference across the wall developed during cooldown results in a higher value of reference fracture toughness at the 1/4t location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in reference fracture toughness exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4t location and, therefore, allowable pressures could be lower if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4t flaw at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the reference fracture toughness for the inside 1/4t flaw during heatup is lower than the reference fracture toughness for the same flaw during steady-state conditions at the same coolant temperature. However, conditions may exist so that the effects of compressive thermal stresses and lower reference fracture toughness do not offset each other and the pressure-temperature curve based on finite heatup rates could become limiting. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature, the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained for the inside 1/4t flaw.

The third portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case of a 1/4t outside surface flaw. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and coolant temperature during the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate is analyzed on an individual basis.

Following the generation of the three pressure-temperature curves, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state data and finite heatup rate data for both inside and outside surface flaws. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is not possible to predict which condition is most limiting because of local differences in irradiation (RT_{NDT}), metal temperature and thermal stresses. With the composite curve, the pressure limit is at all times based on analysis of the most critical situation.

Finally, the 1983 Amendment to 10CFR50(4) has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule 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 and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. In addition, when the core is critical, the pressure-temperature limits for core operation (except for low power physics tests) require that the reactor vessel 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 pressure-temperature curve for heatup and cooldown. These limits are incorporated into the pressure-temperature limit curves wherever applicable.

A petition for rulemaking to eliminate the flange requirement contained in 10CFR50 Appendix G was submitted to the NRC by Westinghouse in November 1999. Until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

Figure 2.2 shows an example of a heatup curve using a heatup rate of 60°F/Hr applicable for the first 16 EFPY. Figure 2.3 shows an example of cooldown curves using rates of 0°, 20°, 40°, 60°, and 100°F/Hr applicable for the first 16 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 2.2 and 2.3. Note that the step in these curves are due to the previously described flange requirements [4].

2.6 PRESSURE-TEMPERATURE CURVE GENERATION METHODOLOGY

2.6.1 Thermal and Stress Analyses

The time-dependent temperature solution utilized in both the heatup and cooldown analysis is based on the one-dimensional transient heat conduction equation:

$$\rho C \frac{\partial T}{\partial t} = K \left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r} \right] \tag{2.6.1-1}$$

with the following boundary conditions applied at the inner and outer radii of the reactor vessel,

$$\text{at } r = r_i, \quad -K\frac{\partial T}{\partial r} = h(T - T_c) \tag{2.6.1-2}$$

$$\text{at } r = r_o, \quad \frac{\partial T}{\partial r} = 0 \tag{2.6.1-3}$$

where,

r_i	=	reactor vessel inner radius
r_o	=	reactor vessel outer radius
ρ	=	material density
C	=	material specific heat
K	=	material thermal conductivity
T	=	local temperature
r	=	radial location
t	=	time
h	=	heat transfer coefficient between the coolant and the vessel wall
T_c	=	coolant temperature

These equations are solved numerically to generate the position and time-dependent temperature distributions, T(r,t), for all heatup and cooldown rates of interest.

With the results of the heat transfer analysis as input, position and time-dependent distributions of hoop thermal stress are calculated using the formula for the thermal stress in a hollow cylinder given by Timoshenko[14].

$$\sigma_\theta(r,t) = \frac{E\alpha}{1-v}\frac{1}{r^2}\left[\frac{r^2 + r_i^2}{r_o^2 - r_i^2}\int_{r_i}^{r_o} T(r,t)r\,dr + \int_{r_i}^{r} T(r,t)r\,dr - T(r,t)r^2\right] \tag{2.6.1-4}$$

where,

$\sigma_\theta(r,t)$	=	hoop stress at location and time t
E	=	modulus of elasticity
α	=	coefficient of linear expansion
V	=	Poisson's ratio

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α are evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r)r\,dr}{r_o^2 - r_i^2} \qquad (2.6.1\text{-}5)$$

E and α are calculated as a function of this equivalent temperature and the Eα product in equation (2.6.1-4) is treated as a constant in the computation of hoop thermal stress.

The linear bending (σ_b) and constant membrane (σ_m) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in Appendix A, to Section XI of the ASME Code[15]. These stress components are used for determining the thermal stress intensity factors, K_{It}, as described in subsections 2.6.3 and 2.6.4.

2.6.2 Steady-State Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at the 1/4t location where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius.

The allowable pressure $P(T_c)$ is a function of coolant temperature, and the pressure temperature curve is calculated for the steady state case at the assumed 1/4t inside surface flaw. First, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t}}{2.0} \qquad (2.6.2\text{-}1)$$

where,

$K_{Ia}(T\text{-}RT_{NDT})$ = allowable reference stress intensity factor as a function of T-RT_{NDT} at 1/4t. (See Sections 2.7 and 2.8 for the new approach using Code Cases N640 and N588.)

Next, the maximum allowable pressure stress is determined using an iterative process and the following three equations:

$$Q = \varphi^2 - 0.212\left(\frac{\sigma_P}{\sigma_y}\right)^2 \qquad (2.6.2\text{-}2)$$

$$\sigma_P = \frac{K_{IM(max)}}{1.1\, M_K \sqrt{\frac{\pi a}{Q}}} \tag{2.6.2-3}$$

$$K_{IP} = 1.1 M_K \sigma_P \sqrt{\frac{\pi a}{Q}} \tag{2.6.2-4}$$

where,

Q = flaw shape factor modified for plastic zone size[16],

ϕ = is the elliptical integral of the 2nd kind (ϕ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw)[16],

0.212 = plastic zone size correction factor[16],

σ_P = pressure stress,

σ_y = yield stress,

1.1 = correction factor for surface breaking flaws,

M_K = correction factor for constant membrane stress [16], M_K as function of relativeflaw depth (a/t) is shown in Figure 2.4,

a = crack depth of 1/4t,

K_{IP} = pressure stress intensity factor.

The maximum allowable pressure stress is determined by incrementing σ_p from an initial value of 0.0 psi until a pressure stress is found that computes a K_{IP} value within 1.0001 of the $K_{IM(max)}$ value. After the maximum allowable σ_p is found, the maximum allowable internal pressure is determined by

$$P(T_c) = \sigma_P \left[\frac{r_o^2 - r_i^2}{r_o^2 + r_i^2} \right] \tag{2.6.2-5}$$

where,

$P(T_c)$ = calculated allowable pressure as a function of coolant temperature.

2.6.3 Finite Cooldown Rate Analyses

For each cooldown rate the pressure-temperature curve is calculated at the inside 1/4t location. First, the thermal stress intensity factor is calculated for a coolant temperature at a given time using the following equation from the Welding Research Council[16]:

$$K_{It} = \left[\sigma_m \, 1.1 M_K + \sigma_b \, M_B\right] \sqrt{\frac{\pi a}{Q}} \qquad (2.6.3\text{-}1)$$

where,

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_K = correction factor for membrane stress[16] (see Figure 2.4),

M_B = correction factor for bending stress[16], M_B as a function of relative flaw depth (a/t) is shown in Figure 2.5.

The flaw shape factor Q in equation (2.6.2-6) is calculated from[16]

$$Q = \phi^2 - 0.212\left(\frac{\sigma_m + \sigma_b}{\sigma_y}\right)^2 \qquad (2.6.3\text{-}2)$$

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \qquad (2.6.3\text{-}3)$$

From $K_{IM(max)}$, the maximum allowable pressure is determined using the iterative process described above and equations (2.6.2-2) through (2.6.2-5).

The steady-state pressure-temperature curve of Section 2.6.2 is compared to the cooldown curves for the 1/4t inside surface flaw at each cooldown rate. At any time, the allowable pressure is the lesser of the two values, and the resulting curve is called the composite cooldown limit curve.

Finally, the 10 CFR Part 50[4] requirement for the closure flange region is incorporated into the cooldown composite curve as described in Section 2.5.

2.6.4 Finite Heatup Rate Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) or (2.5-2) at both the 1/4t and 3/4t locations where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius. At the 3/4t location, a 1/4t flaw is assumed to originate on the outside of the vessel.

For each heatup rate a pressure-temperature curve is calculated at the 1/4t and 3/4t locations. First, the thermal stress intensity factor is calculated at the 1/4t and 3/4t locations for a coolant temperature at a given time using Option 1 or 2 from Section 2.6.3.

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factors at the 1/4t and 3/4t locations are determined using the following equations:

$$\text{At 1/4t,} \quad K_{IM(max)1/4t} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \qquad (2.6.4\text{-}1)$$

$$\text{At 3/4t,} \quad K_{IM(max)3/4t} = \frac{K_I * (T - RT_{NDT})_{3/4t} - K_{It}(T_c)_{3/4t}}{2.0} \qquad (2.6.4\text{-}2)$$

From $K_{IM(max)1/4t}$ and $K_{IM(max)3/4t}$, the maximum allowable pressure at both the 1/4t and 3/4t locations is determined using the iterative process described in Section 2.6.2 and equations (2.6.2-2) through (2.6.2-5).

As was done with the cooldown case, the steady state pressure-temperature curve of Section 2.6.2 is compared with the 1/4t and 3/4t location heatup curves for each heatup rate, with the lowest of the three being used to generate the composite heatup limit curve. The composite curve is then adjusted for the 10 CFR Part 50[(4)] rule for closure flange requirements, as discussed in Section 2.5.

2.6.5 Hydrostatic and Leak Test Curve Analyses

The minimum inservice hydrostatic leak test curve is determined by calculating the minimum allowable temperature at two pressure values (pressure values of 2000 psig and 2485 psig, approximately 110% of operating pressure, are generally used). The curve is generated by drawing a line between the two pressure-temperature data points. The governing equation for generating the hydrostatic leak test pressure-temperature limit curve is defined in Appendix G, Section X1, of the ASME Code[(5)] as follows:

$$1.5\ K_{IM} < K_{Ia} \qquad (2.6.5\text{-}1)$$

where, K_{IM} is the stress intensity factor caused by the membrane (pressure) stress and K_{Ia} is the reference stress intensity factor as defined in equation (2.5-1). Note that the thermal stress intensity factor is neglected (i.e., $K_{It} = 0$) since the hydrostatic leak test is performed at isothermal conditions.

The pressure stress is determined by,

$$\sigma_p = \left[\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2}\right] P \tag{2.6.5-2}$$

where,

P = the input pressure (generally 2000 and 2485 psig)

Next, the pressure stress intensity factor is calculated for a 1/4t flaw by,

$$K_{IM} = \left[1.1 M_K \sqrt{\frac{\pi a}{Q}}\right] \sigma_p \tag{2.6.5-3}$$

The K_{IM} result is multiplied by the 1.5 factor of equation (2.5-2) and divided by 1000,

$$K_{HYD} = \frac{1.5 K_{IM}}{1000} \tag{2.6.5-4}$$

Finally, the minimum allowable temperature is determined by setting K_{HYD} to K_{Ia} in equation (2.5-1) and solving for temperature T:

$$T = \frac{\ell n \left[\frac{(K_{HYD} - 26.78)}{1.223}\right]}{0.0145} + RT_{NDT} - 160.0 \tag{2.6.5-5}$$

The 1983 Amendment to 10CFR50(4) has a rule which addresses the test temperature for hydrostatic pressure tests. This rule states that, when there is no fuel in the reactor vessel during hydrostatic pressure tests or leak tests, the minimum allowable test temperature must be 60°F above the adjusted reference temperature of the beltline region material that is controlling. If fuel is present in the reactor vessel during hydrostatic pressure tests or leak tests, the requirements of this section and Section 2.5 must be met.

2.7 1996 ADDENDA TO ASME SECTION XI, APPENDIX G METHODOLOGY

ASME Section XI, Appendix G was updated in 1996 to incorporate the most recent elastic solutions for K_I due to pressure and radial thermal gradients. The new solutions are based on finite element analyses for inside surface flaws performed at Oak Ridge National Laboratories and sponsored by the NRC, and work published for outside surface flaws. These solutions provide results that are very similar to those obtained by using solutions previously developed by Raju and Newman.

This revision provides consistent computational methods for pressure and thermal K_I, for thermal gradients through the vessel wall at any time during the transient. Consistent with the original version of

Appendix G, no contribution for crack face pressure is included in the K_I due to pressure, and cladding effects are neglected.

Using these elastic solutions in the low temperature region will provide some relief to restrictions associated with reactor operation at relatively low temperatures. Although the relief is relatively small in terms of the absolute allowable pressure, the benefits are substantial, because even a small increase in the allowable pressure can be a significant percentage increase in the operating window at relatively low temperatures. Implementing this revision results in a safety benefit (reduced likelihood of lifting COMS relief valves), with no reduction in vessel integrity.

The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension:

$$K_{Im} = M_m \times (pR_i / t) \tag{2.7-1}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,

M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,

M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 3.09 for $\sqrt{t} > 3.464$

where,

p = internal pressure,

Ri = vessel inner radius, and

t = vessel wall thickness.

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where $M_b = 0.667\ M_m$

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{It} = 0.953 \times 10^{-3}\ CR\ t^{2.5} \tag{2.7-2}$$

where:

CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{It} = 0.753 \times 10^{-3}\ HU\ t^{2.5} \qquad (2.7\text{-}3)$$

where:

HU = the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from ASME Section XI, Appendix G, Figure G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Section XI, Appendix G, Figure G-2214-2 for the maximum thermal K_I .

1. The maximum thermal K_I relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2) of Appendix G to ASME Section XI.

2. Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (2.7\text{-}4)$$

or similarly, K_{It} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \qquad (2.7\text{-}5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the equation:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.7\text{-}6)$$

where *x* is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and *a* is the maximum crack depth.

Once K_{Ia} (As calculated via Equation 2.5-1) is known, the pressure can be solved using Equation 2.5-3 with the newly calculated K_{It} and new equation for K_{IM}.

$$C * [M_m \times (pR_i / t)] + K_{It} < K_{Ia}$$

where:

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

C = 1.5 for hydrostatic and leak test conditions when the reactor core is not critical

This results in a pressure equation as follows:

$$p = \frac{[K_{It} - K_{Ia}]}{C * M_m * (R_i / t)} \tag{2.7-7}$$

Note that K_{It} is equal to zero for steady state and hydrostatic leak test conditions. In addition, K_{Ia} and K_{It} must be calculated individually for inside and outside flaw locations (i.e., the ¼T and ¾T wall locations) and the minimum pressure must be used from these two locations. [Note: K_{Ia} for ¼ T steady state is not the same as K_{Ia} for ¼T thermal conditions since the wall temperature is equal to the water temperature in steady state, but is not the case under thermal conditions.]

2.8 CODE CASES N-640 FOR KIC AND N-588 FOR CIRCUMFERENTIAL WELD FLAWS

2.8.1 ASME Code Case N-640

In February of 1999, the ASME Code approved Code Case N-640 which allows the use of the reference fracture toughness curve K_{Ic}, as found in Appendix A of Section XI, in lieu of Figure G-2110-1 in Appendix G for the development of pressure-temperature limit curves. (This is also described in Section 2.5 herein). Thus, when developing pressure-temperature limit curves, it is acceptable to calculate the reference stress intensity via Equation 2.5-2, in lieu of Equation 2.5-1. In addition, the K_{Ic} can be substituted for K_{Ia} in Equations 2.5-3, 2.6.2-1, 2.6.3-3, 2.6.4-1, 2.6.4-2, 2.6.5-1 and 2.7-7.

2.8.2 ASME Code Case N-588

In 1997, ASME Section XI, Appendix G was revised to add a methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The original ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds. In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Section XI, Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension…

The K_I corresponding to membrane tension for the postulated circumferential defect of G-2120 is

$$K_{IM} = M_m \text{ x } (PR_i/t)$$

Where, M_m for an inside surface flaw is given by:

$M_m \quad = \quad 0.89$ for $\sqrt{t} < 2$,

$M_m \quad = \quad 0.443 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

$M_m \quad = \quad 1.53$ for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

$M_m \quad = \quad 0.89$ for $\sqrt{t} < 2$,

$M_m \quad = \quad 0.443 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

$M_m \quad = \quad 1.53$ for $\sqrt{t} > 3.464$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circumferential weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology.

2.9 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule 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 pre-service hydrostatic test pressure (3106 psig), which is 621 psig for a typical Westinghouse reactor vessel design.

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

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in Code Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1.”

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," concluded that the integrity of the closure head/vessel 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. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves, once the requirements of 10CFR50 Appendix G are changed. However, until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

2.10 MINIMUM BOLTUP TEMPERATURE

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the minimum boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT} whichever is higher.

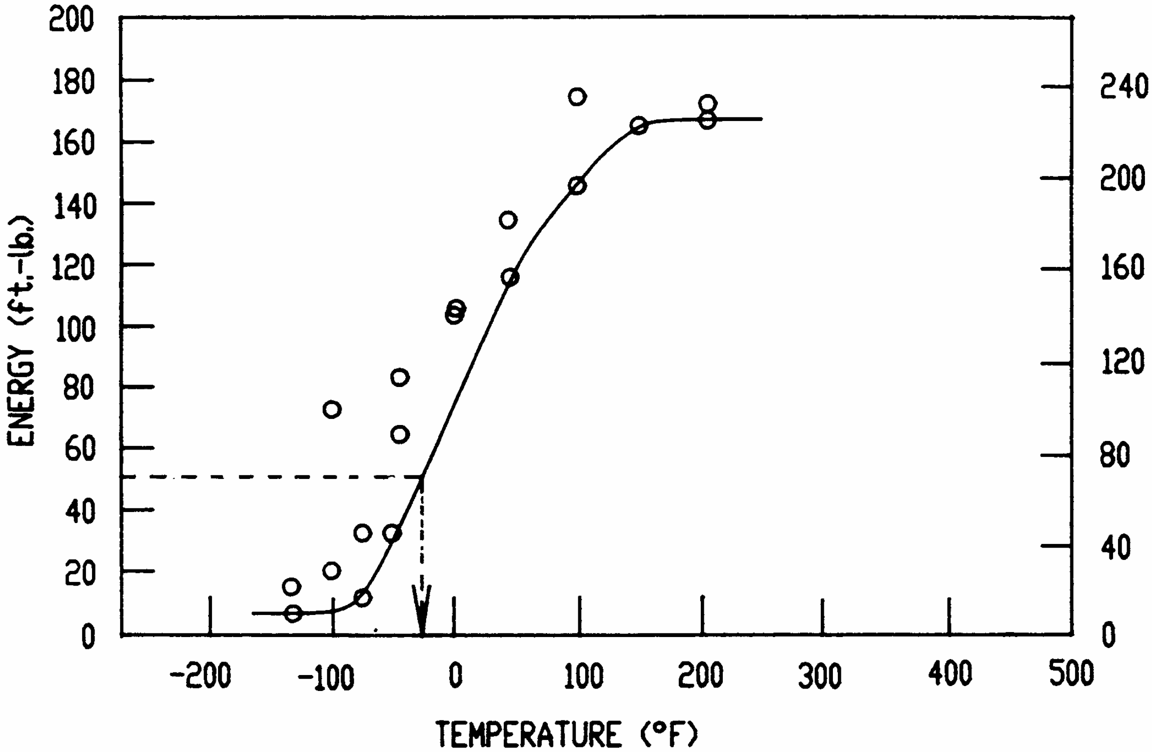


Figure 2.1 Example of a Charpy Impact Energy Curve Used to Determine IRT_{NDT} (Note: 35 mils lateral expansion is required at indicated temperature)

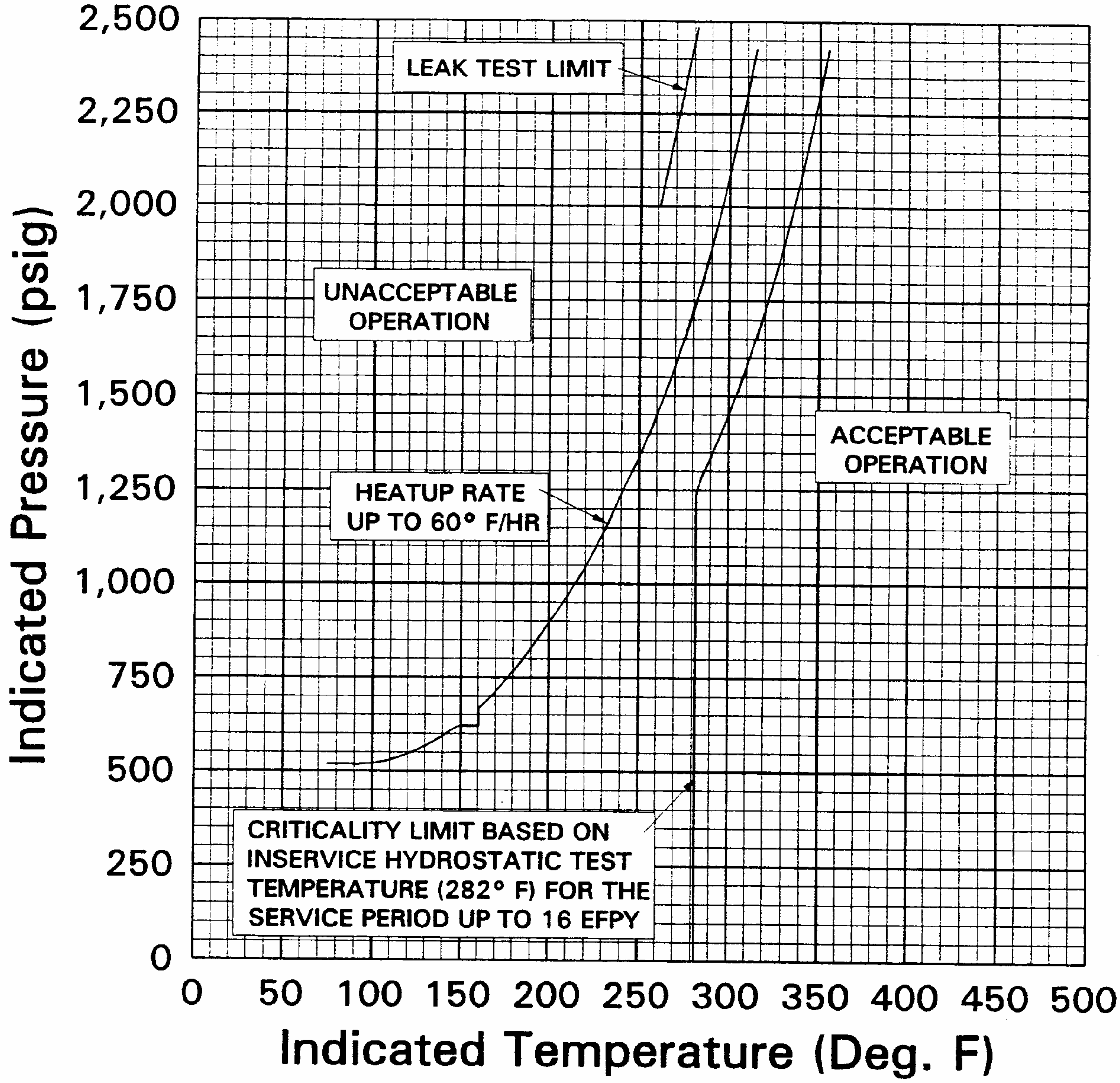


Figure 2.2 Heatup Pressure-Temperature Limit Curve For Heatup Rates up to 60°F/Hr

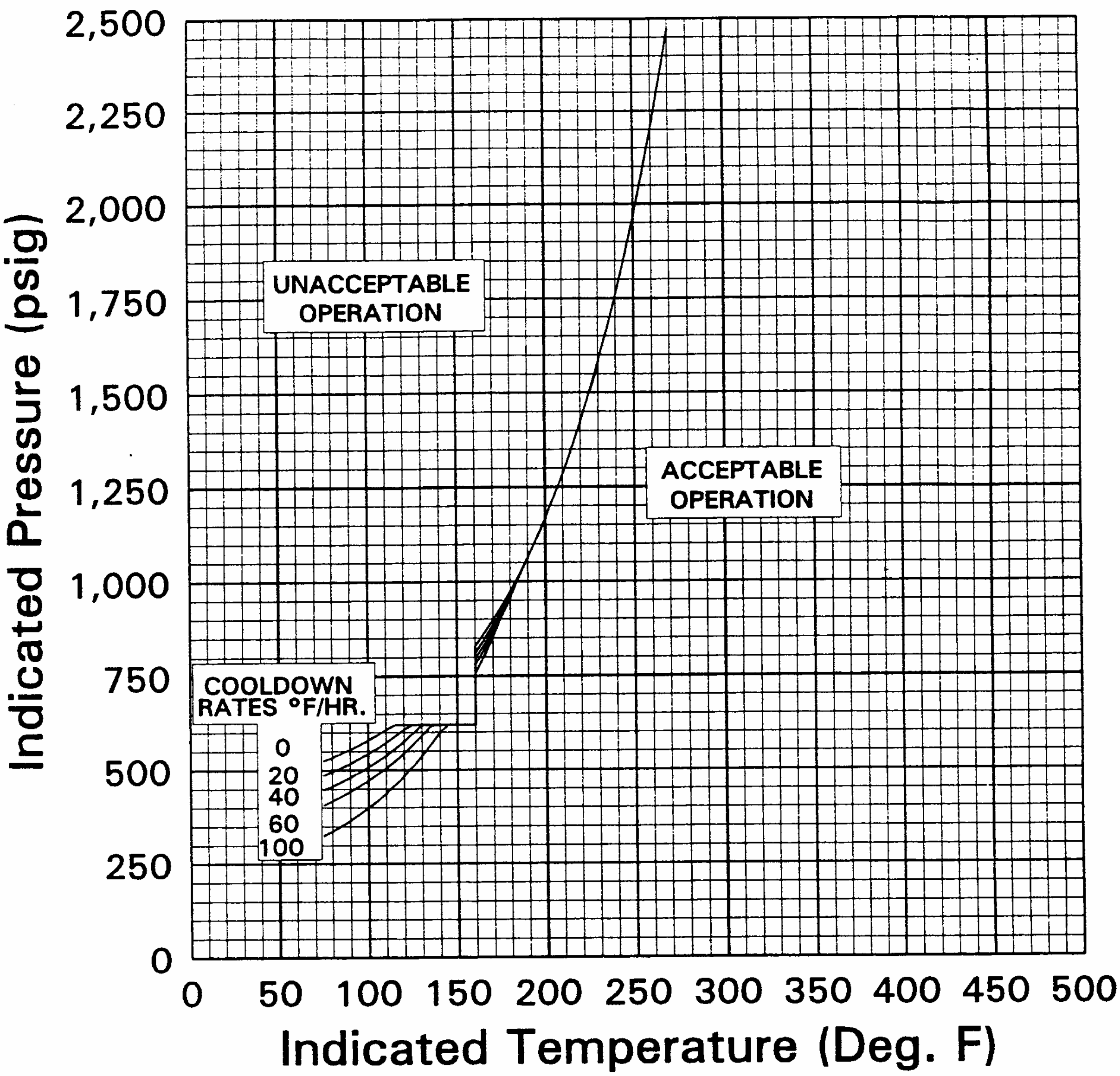


Figure 2.3 Cooldown Pressure-Temperature Limit Curves or Cooldown Rates up to 100°F/Hr

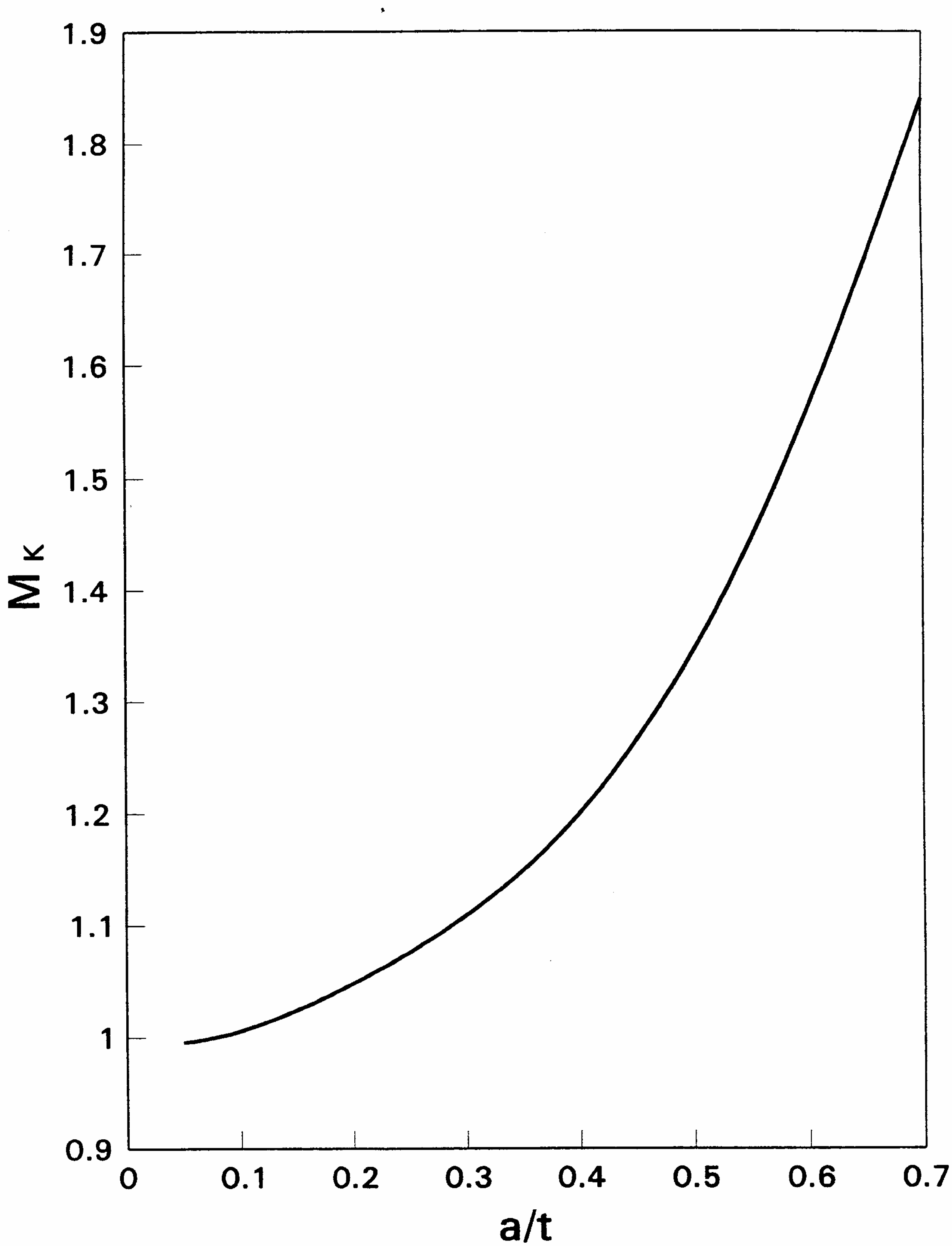


Figure 2.4 Membrane Stress Correction Factor (M_K) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6 (Welding Research Bulletin 175 Method)

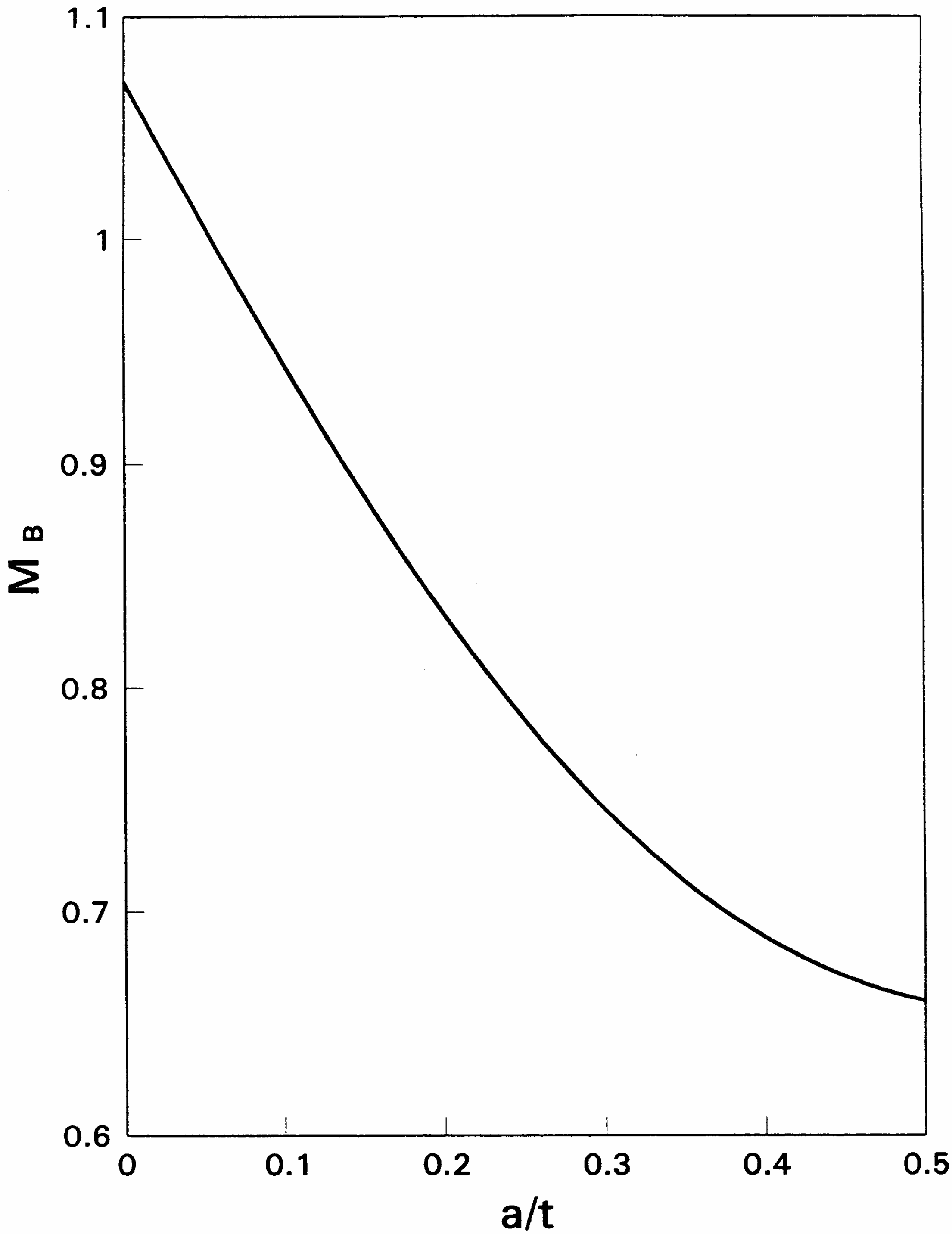


Figure 2.5 Bending Stress Correction Factor (M_B) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6 (Welding Research Bulletin 175 Method)

3.0 COLD OVERPRESSURE MITIGATING SYSTEM (COMS)

3.1 INTRODUCTION

The purpose of the COMS is to supplement the normal plant operational administrative controls and the water relief valves in the Residual Heat Removal System (RHRS) when they are unavailable to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. This has been achieved by conservatively choosing COMS setpoints which prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G[(4)] requirements. The COMS is designed to provide the capability, during relatively low temperature operation (typically less than 350°F), to automatically prevent the RCS pressure from exceeding the applicable limits. Once the system is enabled, no operator action is involved for the COMS to perform its intended pressure mitigation function. Thus, no operator action is modelled in the analyses supporting the setpoint selection, although operator action may be initiated to ultimately terminate the cause of the overpressure event.

The PORVs located near the top of the pressurizer, together with additional actuation logic from the wide-range pressure channels, are utilized to mitigate potential RCS overpressure transients defined below if the RHRS water relief valves are inadvertently isolated from the RCS. The COMS provides the supplemental relief capacity for specific transients which would not be mitigated by the RHRS relief valves. In addition, a limit on the PORV piping is accommodated due to the potential for water hammer effects to be developed in the piping associated with these valves as a result of the cyclic opening and closing characteristics during mitigation of an overpressure transient. Thus, a pressure limit more restrictive than the 10CFR50, Appendix G[(4)] allowable is imposed above a certain temperature so that the loads on the piping from a COMS event would not affect the piping integrity.

Two specific transients have been defined, with the RCS in a water-solid condition, as the design basis for COMS. Each of these scenarios assumes as an initial condition that the RHRS is isolated from the RCS, and thus the relief capability of the RHRS relief valves is not available. The first transient consists of a heat injection scenario in which a reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature and the RHRS has been inadvertently isolated. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into a water-solid RCS caused by the simultaneous isolation of the RHRS isolation of letdown and failure of the normal charging flow controls to the full flow condition. Various combinations of charging and safety injection flows may also be evaluated on a plant-specific basis; however, the mass injection transient used as a design basis should encompass the limiting pump(s) operability configuration permitted per the plant-specific Technical Specifications during the Modes when COMS is required to be in operation. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

3.2 COMS SETPOINT DETERMINATION

Westinghouse has developed the following methodology which is employed to determine PORV setpoints for mitigation of the COMS design basis cold overpressurization transients. This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel integrity.

3.2.1 Parameters Considered

The selection of proper COMS setpoints for actuating the PORVs requires the consideration of numerous system parameters including:

a. Volume of reactor coolant involved in transient

b. RCS pressure signal transmission delay

c. Volumetric capacity of the relief valves versus opening position

d. Stroke time of the relief valves (open & close)

e. Initial temperature and pressure of the RCS

f. Mass input rate into RCS

g. Temperature of injected fluid

h. Heat transfer characteristics of the steam generators

i. Initial temperature asymmetry between RCS and steam generator secondary water

j. Mass of steam generator secondary water

k. RCP startup dynamics

l. 10CFR50, Appendix G pressure/temperature characteristics of the reactor vessel

m. Pressurizer PORV piping/structural analysis limitations

n. Dynamic and static pressure difference between reactor vessel midplane and location of wide range pressure transmitter

These parameters are input to a specialized version of the LOFTRAN computer code which calculates the maximum and minimum system pressures.

3.2.2 Pressure Limits Selection

The function of the COMS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing COMS setpoints which prevent exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50[4]. The COMS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. In addition, the COMS also provides for an operational consideration to maintain the integrity of the PORV piping. A typical characteristic 10CFR50

Appendix G curve is shown by Figure 3.1 where the allowable system pressure increases with increasing temperature. This type of curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties. Superimposed on this curve is the PORV piping limit which is conservatively used, for setpoint development, as the maximum allowable pressure above the temperature at which it intersects with the 10CFR50 Appendix G curve.

When a relief valve is actuated to mitigate an increasing pressure transient, the release of a volume of coolant through the valve will cause the pressure increase to be slowed and reversed as described by Figure 3.2. The system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signalled to close. Note that the pressure continues to decrease below the reset pressure as the valve recloses. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the reactor coolant pump #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement and the PORV piping limitations) and the nominal RCP #1 seal performance criteria create a pressure range from which the setpoints for both PORVs may be selected as shown on Figures 3.3 and 3.4.

Where there is insufficient range between the upper and lower pressure limits to select PORV setpoints to provide protection against violation of both limits, setpoint selection to provide protection against the upper pressure limit violation shall take precedence.

3.2.3 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signalled to open at a specific pressure setpoint. However, as shown on Figure 3.2, 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, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed. The maximum and minimum pressures reached (P_{MAX} and P_{MIN}) in the transient are a function of the selected setpoint (P_s) as shown on Figure 3.3. The shaded area represents an optimum range from which to select the setpoint based on the particular mass input case. Several mass input cases may be run at various input flow rates to bound the allowable setpoint range.

3.2.4 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e., the mass injection transient is not sensitive to temperature). The shaded area on Figure 3.4 describes the acceptable band for a heat input transient from which to select the setpoint for a particular initial reactor coolant temperature.

3.2.5 Final Setpoint Selection

By superimposing the results of multiple mass input and heat input cases evaluated, (from a series of figures such as 3.3 and 3.4) a range of allowable PORV setpoints to satisfy both conditions can be determined. Each of the two PORVs may have a different pressure setpoint versus temperature specification such that only one valve will open at a time and mitigate the transient (i.e., staggered setpoints). The second valve operates only if the first fails to open on command. This design supports a single failure assumption as well as minimizing the potential for both PORVs to open simultaneously, a condition which may create excessive pressure undershoot and challenge the RCP #1 seal performance criteria. However, each of the sets of staggered setpoints must result in the system pressure staying below the P_{MAX} pressure limit shown on Figures 3.3 and 3.4 when either valve is utilized to mitigate the transient.

The function generator used to program the pressure versus setpoint curves for each valve has a limited number of programmable break points (typically 9). These are strategically defined in the final selection process, with consideration given to the slope of any line segment, which is limited to approximately 24 psi/°F.

The selection of the setpoints for the PORVs considers the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50[4] or the peak RCS pressure based upon piping/structural analysis loads. The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. The upper pressure limits are already based on conservative assumptions (such as a safety factor of 2 on pressure stress, use of a lower bound K_{IR} curve and an assumed 1/4T flaw depth with a length equal to 1 1/2 times the vessel wall thickness) as discussed in section 2 of this report. However, uncertainties associated with instrumentation utilized by COMS will be determined using a process described by ISA Standard S67.04-1994. These uncertainties will be accounted for in the selection of COMS PORV setpoints.

While the RHR relief valves also provide overpressure protection for certain transients, these transients are not the same as the design basis transients for COMS. The RHR relief valve design basis precedes the development of the COMS design basis, and therefore the RHR relief valves may not provide protection against the COMS design basis events. The design basis described herein should be considered as applicable only when the pressurizer PORVs are used for COMS.

3.3 APPLICATION OF ASME CODE CASE N-514

ASME Code Case N-514[17] allows low temperature overpressure protection systems (LTOPS, as the code case refers to COMS) to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G, paragraph G-2215, of Section XI of the ASME Code[5]. (Note, that the setpoint selection methodology as discussed in Section 3.2.5 specifically utilizes the steady-state curve.) The application of ASME Code Case N-514 increases the operating margin in the region of the pressure-temperature limit curves where the COMS system is enabled. Code Case N-514 requires LTOPS to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. RT_{NDT} is the highest adjusted reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2.

3.4 ENABLE TEMPERATURE FOR COMS

The enable temperature is the temperature below which the COMS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is very conservative, and is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3.

A significant improvement in the enable temperature can be obtained by application of code case N641. This code case incorporates the benefits of code cases N588, and N640. The resulting enable temperatures for the Westinghouse designs obtained using code case N641 are listed below.

Vessel Type	Axial Flaw	Circumferential Flaw
2 – loop	RT_{NDT} + 23F	Any temperature
3 – loop	RT_{NDT} + 30F	RT_{NDT} – 174F
4 – loop	RT_{NDT} + 34F	RT_{NDT} – 110F

The RCS cold leg temperature limitation for starting an RCP is the same value as the COMS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the COMS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

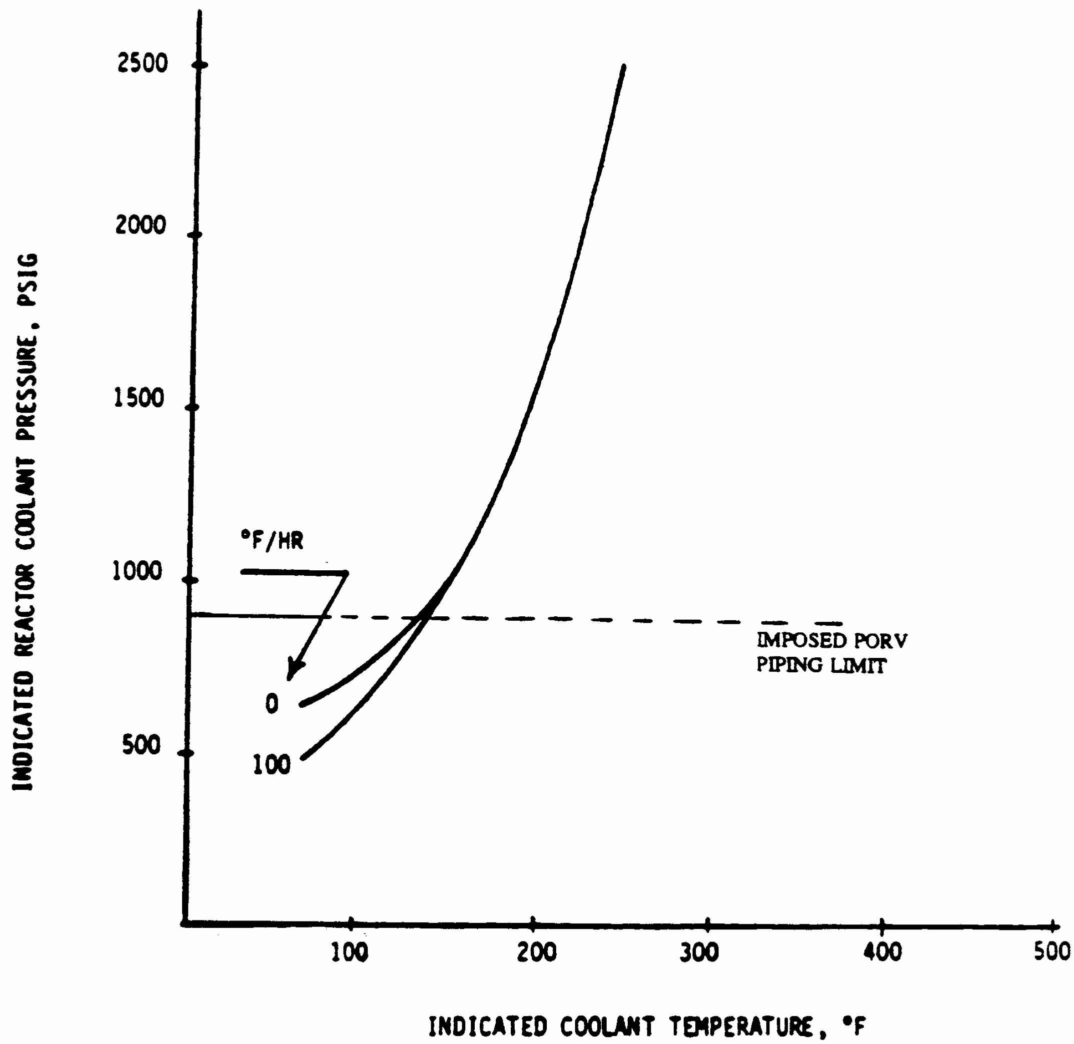


Figure 3.1 Typical Appendix G P/T Characteristics

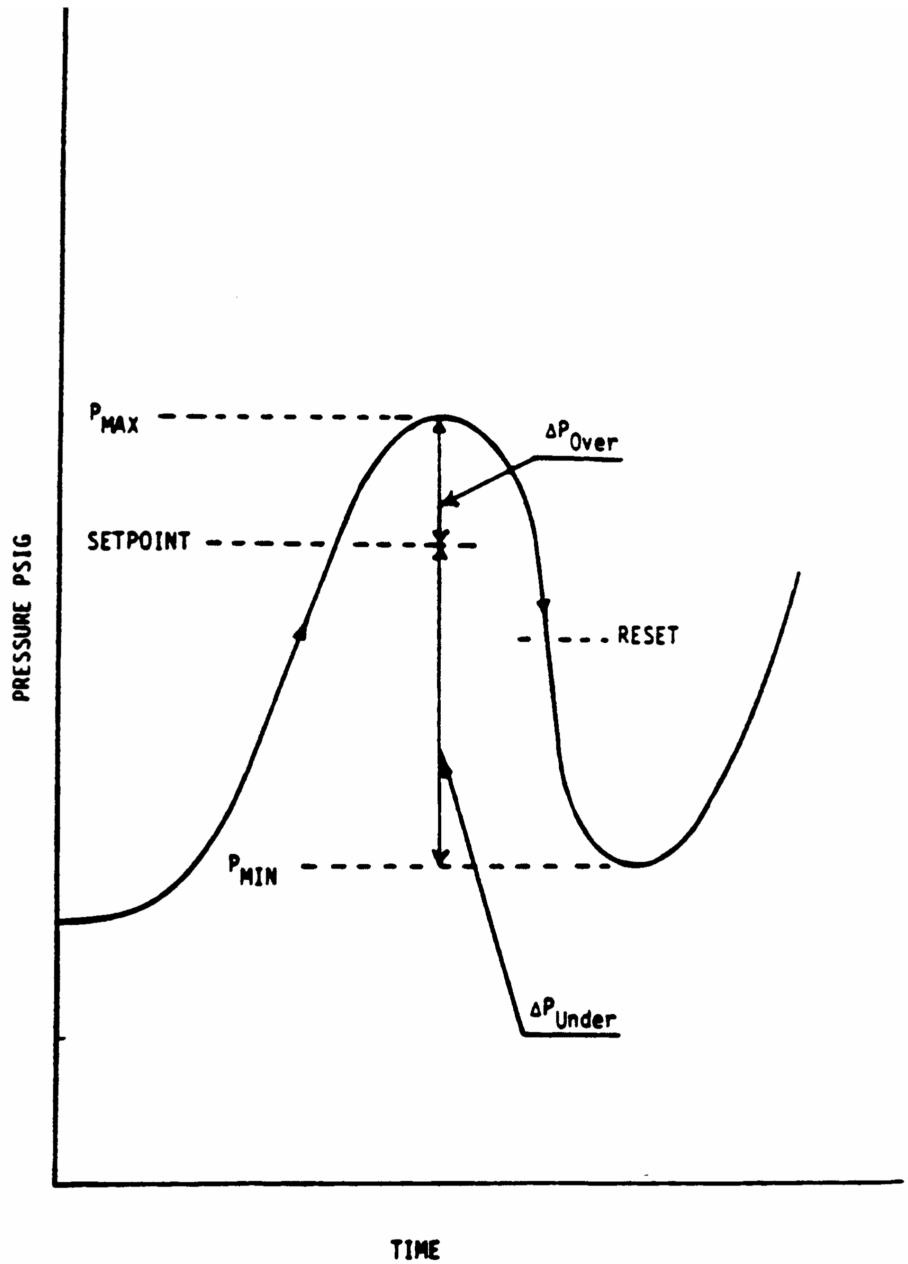
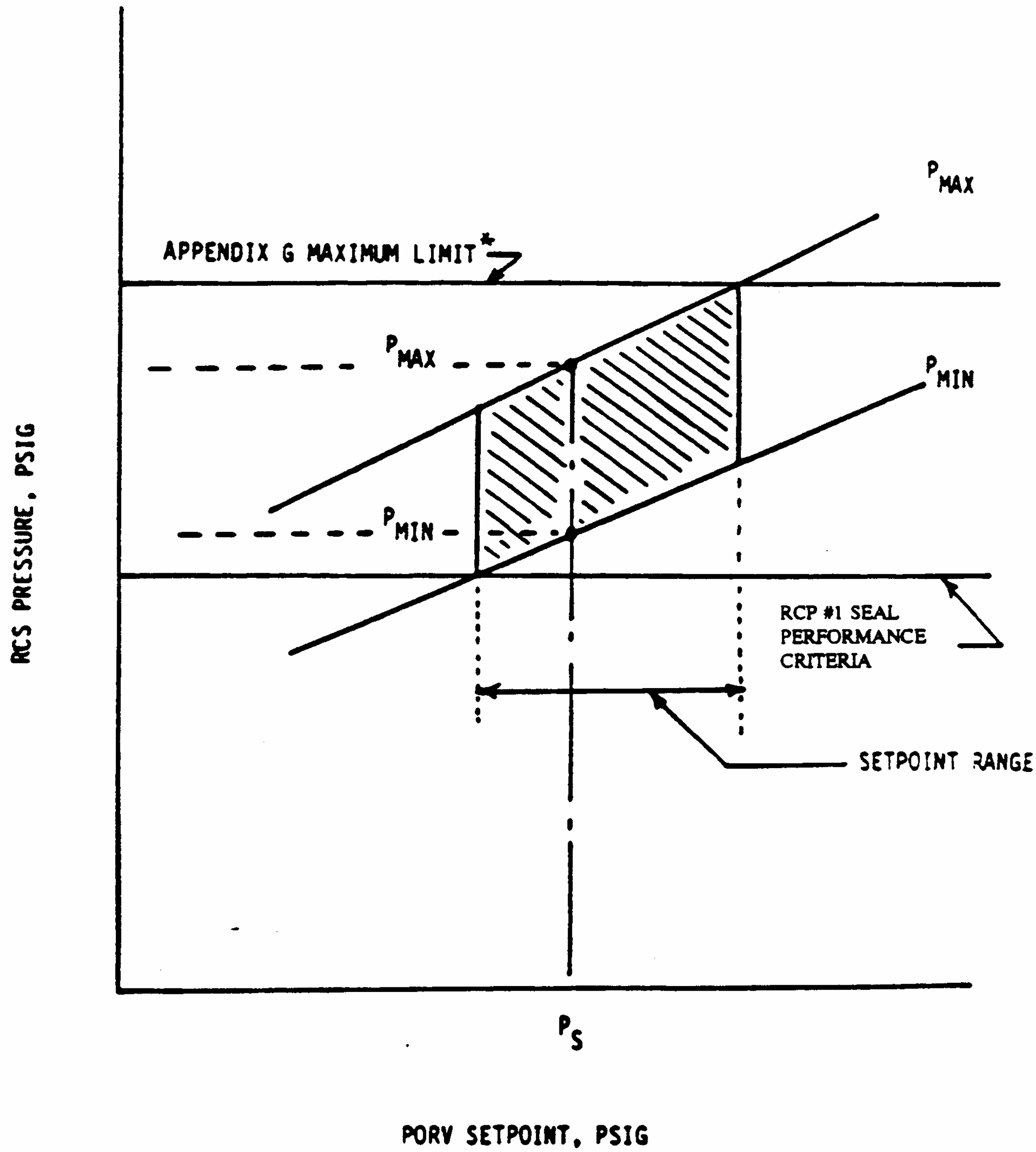
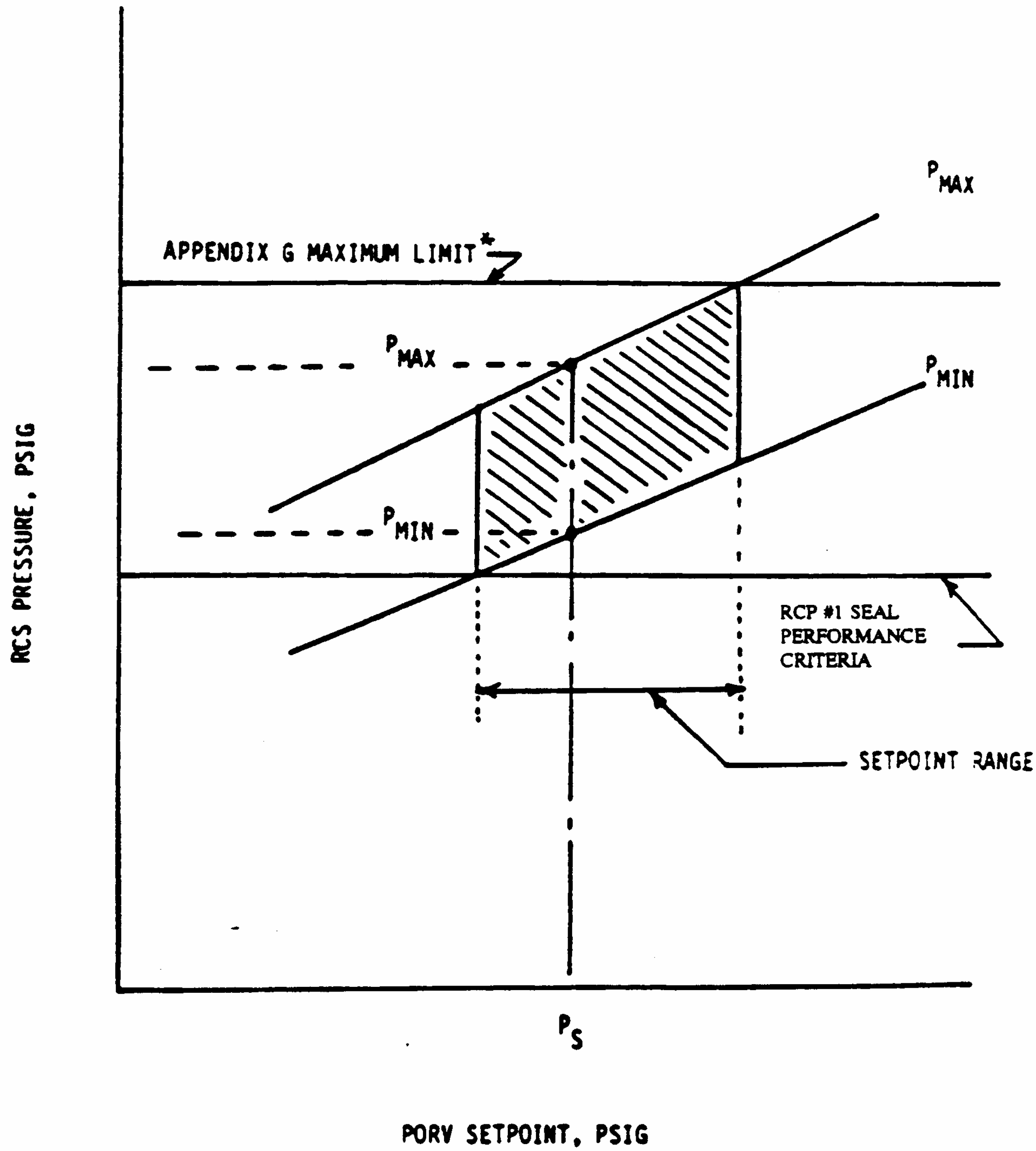


Figure 3.2 Typical Pressure Transient (1 Relief Valve Cycle)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

Figure 3.3 Setpoint Determination (Mass Input)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

Figure 3.4 Setpoint Determination (Heat Input)

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," Revision 2.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," Generic Letter 88-16, October, 1988.

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

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix G, Fracture Toughness Requirements.

5. 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.

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

7. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

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

12. I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Benchmark," NUREG/CR-6454 (ORNL/TM-13205), July 1997.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix A, Analysis of Flaws, Article A3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section X1, Division 1, "Low Temperature Overpressure Protection," Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

19. ASME Boiler and Pressure Vessel Code Case N640, Section XI, Division 1, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," February 26, 1996.

20. ASME Boiler and Pressure Vessel Code Case N588, Section XI, Division 1, "Alternative to Reference Flow Orientation of Appendix G for Circumferential Welds in Reactor Vessels," December 12, 1997.

21. ASME Boiler and Pressure Vessel Code Case N641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000.

22. I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

APPENDIX A

RELEVANT ASME NUCLEAR CODE CASES

Table A-1 Status of ASME Nuclear Code Cases Associated with the P-T Limit Curve/COMS Methodology			
Code Case	**Title**	**Approved by ASME**	**Section XI of the ASME Code**
514	Low Temperature Overpressure Protection	2/12/92	1995 Edition through the 1996 Addenda
588	Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel	12/12/97	1998 Edition through the 2000 Addenda
640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves	2/26/99	1998 Edition through the 2000 Addenda
641	Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirement	1/17/00	1998 Edition through the 2000 Addenda

APPENDIX B

CORRESPONDENCE WITH THE NRC

WOG-04-086
February 18, 2004

WCAP-14040, Rev. 3
Project Number 694

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

Attention: Chief, Information Management Branch, Division of Program Management

Subject: Westinghouse Owners Group
Transmittal of Comments on the Draft Safety Evaluation for WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (MUHP-3073, TAC No. MB5754)

On February 2, 2004, the NRC provided a draft Safety Evaluation (SE) of WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," to the Westinghouse Owners Group (WOG) for review and comment (Ref. 1). Enclosure 1 contains a mark-up of the suggested clarifications to the SE for consideration by the NRC. The WOG requests that the NRC issue a final Safety Evaluation by April 1, 2004.

If you require further information, feel free to contact Mr. Ken Vavrek, Owners Group Project Office at 412-374-4302.

Sincerely,

Frederick. P. "Ted" Schiffley, II, Chairman
Westinghouse Owners Group

Enclosure

WOG-04-086
February 18, 2004

cc: WOG Steering Committee
WOG Management Committee
WOG Licensing Subcommittee
WOG Materials Subcommittee
D. Holland, USNRC OWFN 07 E1 (1L, 1A) (via Federal Express)
S. Dinsmore, USNRC (1L, 1A) OWFN 10H4
Project Management Office
J. Gresham
J. D. Andrachek
W.H. Bamford
T.J. Laubham
J. Perock

Reference:

1. NRC Letter, S. Dembeck (NRC) to G. Bischoff (Westinghouse), "Draft Safety Evaluation of Topical Report WCAP-14040, Revision 3, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,' (TAC No. MB5754)," February 2, 2004.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMIT CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P-T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted ~~the revised~~ TR for NRC staff review and approval by letter dated October 20, 2003.

revisions to the

2.0 REGULATORY EVALUATION

Four specific topics are addressed in the context of the development of a PTLR methodology: (1) the calculation of neutron fluences for the RPV and RPV surveillance capsules; (2) the evaluation of RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these RPV material properties and the establishment of cold overpressure mitigating system (COMS) setpoints to protect the RPV from brittle failure; and (4) the development of an RPV material surveillance program to monitor changes in RPV material properties due to radiation. Regulatory requirements related to the four topics noted above are addressed in Appendices G and H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50). Appendix G to 10 CFR Part 50 provides requirements related to RPV P-T limit development and directly or indirectly addresses topics (1) through (3) above. Appendix H to 10 CFR Part 50 defines regulatory requirements related to RPV material surveillance programs and addresses topic (4) above.

For the staff's review of WCAP-14040, Revision 3, several additional guidance documents were used. NRC Standard Review Plan (SRP) Sections 5.2.2, "Overpressure Protection," 5.3.1, "Reactor Vessel Materials," and 5.3.2, "Pressure-Temperature Limits," provide specific review guidance related to RPV material property determination, P-T limit development, and COMS performance. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes analysis procedures acceptable to the NRC staff for the purpose of assessing RPV material property changes due to radiation. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," addresses NRC staff expectations for an acceptable fluence calculation methodology. American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides guidance on the establishment of RPV material surveillance programs and editions of ASTM E 185 are incorporated by reference into Appendix H to 10 CFR Part 50. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G provides specific requirements regarding the development of P-T limit curves.

Finally, specific guidance regarding topics and the level of detail to which they must be addressed as part of an acceptable PTLR methodology is given in GL 96-03.

3.0 TECHNICAL EVALUATION

The technical requirements to be addressed in an acceptable PTLR methodology are provided under the column heading "Minimum Requirements to be Included in Methodology" in the table entitled "Requirements for Methodology and PTLR" in Attachment 1 to GL 96-03. Summarized versions of the seven requirements are given below, along with the staff's technical evaluation of information in WCAP-14040, Revision 3, related to each requirement.

Requirement 1: Regarding the reactor vessel material surveillance program, the methodology should briefly describe the surveillance program. The methodology should clearly reference the requirements of Appendix H to 10 CFR Part 50.

The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements in GL 96-03 related to RPV material surveillance program ~~issues.~~

discussed the

Requirement 2: Regarding the calculation of RPV materials' adjusted reference temperatures (ART) values, the methodology should describe the method for calculating material ART values using RG 1.99, Revision 2.

provision 2 in the table of Attachment 1 to

Information regarding how material ARTs are to be determined within the WCAP-14040, Revision 3, PTLR methodology is provided in Section 2.3 and 2.4 of the TR. In Section 2.3, the determination of initial, unirradiated material properties from Charpy V-notch impact tests and/or nil-ductility drop weight tests is clearly defined. The methodology specified in Section 2.3 accurately incorporates the guidance found in ASME Code Section III, paragraph NB-2331 and additional information in SRP Section 5.3.1.

-3- No comments on pages 3, 4 and 5

In Section 2.4 of the TR, the determination of changes in material properties due to irradiation is addressed, along with the determination of margins necessary to account for uncertainties in initial properties and irradiation damage assessment. The methodology specified in Section 2.4 accurately incorporates the guidance found in RG 1.99, Revision 2.

The NRC staff, therefore, determined that the methodology described for determining material ART values in WCAP-14040, Revision 3, was consistent with the guidance provided in the ASME Code, SRP Section 5.3.1, and RG 1.99, Revision 2, and was, therefore, acceptable.

Requirement 3: Regarding the development of RPV P-T limit curves, the methodology should describe the application of fracture mechanics-based calculations in constructing P-T limit curves based on the provisions of Appendix G to Section XI of the ASME Code and SRP Section 5.3.2.

Basic and optional elements of the methodology for RPV P-T limit curve development in WCAP-14040, Revision 3, are given in Sections 2.5, 2.6, 2.7, 2.8, and Appendix A of the TR.

In Section 2.5 of the TR, the fracture toughness-based guidelines from Appendix G to Section XI of the ASME Code are specified (based on the 1995 Edition through 1996 Addenda of the ASME Code). Notably, specific reference is made to the use of: (1) the ASME Code lower bound dynamic crack initiation/crack arrest (K_{IA}) fracture toughness curve; (2) the use of a postulated flaw that has a depth of one-quarter of the wall thickness and a 6:1 aspect ratio; and (3) the use of a structural factor of 2 on primary membrane stress intensities (K_{IM}) when evaluating normal heatup and cooldown and a structural factor of 1.5 on K_{IM} when evaluating hydrostatic/leak test conditions.

Optional guidelines for P-T limit curve development are also addressed in WCAP-14040, Revision 3. The option of using the ASME Code static crack initiation fracture toughness curve (K_{IC}), as given in ASME Code Case N-640, is addressed in Sections 2.5 and 2.8. The option of using ASME Code Case N-588, which enables the postulation of a circumferentially-oriented flaw (with appropriate stress magnification factors) when evaluating a circumferential weld, is addressed in Section 2.8. WCAP-14040, Revision 3, notes, however, that licensee use of the provisions of either ASME Code Case N-640 or N-588 requires, in accordance with 10 CFR 50.60(b), an exemption if the provisions of the Code Case are not contained in the edition of the ASME Code included in a facility's licensing basis. Appendix A to WCAP-14040, Revision 3, provides additional details regarding the application of optional ASME Code Cases and includes copies of ASME Code Case N-588, N-640, and N-641 (which effectively combines the provisions of N-588 and N-640 into a single Code Case).

A detailed discussion of the calculational methodology for P-T limit curve generation is given in Section 2.6 of the TR. Specific equations are given for the determination of primary membrane stresses due to internal pressure and membrane and bending stresses due to thermal gradients. Equations related to the generation of P-T limit curves for steady-state conditions, finite heatup rates, finite cooldown rates, and hydrostatic/leak test conditions are given. The equations given in WCAP-14040, Revision 3, are equivalent to those provided in Section XI of the ASME Code and consistent with the guidance given in SRP Section 5.3.2.

Therefore, the NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P-T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P-T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

Requirement 4: Regarding the development of RPV P-T limit curves, the methodology should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied when constructing P-T limit curves.

Minimum temperature requirements regarding the material in the highly stressed region of the RPV flange are given in Appendix G to 10 CFR Part 50. Information provided in Sections 2.9 and 2.10 of the TR addresses the incorporation of minimum temperature requirements into the development of P-T limit curves. In Section 2.9, the 10 CFR Part 50, Appendix G requirements are cited. WCAP-14040, Revision 3, goes on to note that there is an effort underway to revise or eliminate these requirements based on information contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." However, WCAP-14040, Revision 3, states that until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

WCAP-14040, Revision 3, provides supplemental information in Section 2.10 regarding the establishment of RPV boltup temperature, specifically that the minimum boltup temperature should be 60 °F or equal to the highest material reference temperature in the highly stressed RPV flange region, whichever is higher (i.e., more conservative). Although no specific requirements related to boltup temperature are provided in Appendix G to 10 CFR Part 50, the information in WCAP-14040, Revision 3, is consistent with other, related requirements in Appendix G to 10 CFR Part 50 and in Appendix G to Section XI of the ASME Code.

The NRC staff concludes that the methodology specified in WCAP-14040, Revision 3, addresses RPV minimum temperature requirements in a way which is consistent with Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code and is, therefore, acceptable.

Requirement 5: Regarding the calculation of RPV materials' ARTs, the methodology should describe how the data from multiple surveillance capsules may be used in ART calculations.

Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.

The NRC staff has reviewed the information in Section 2.4 of the TR and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Requirement 6: Regarding the calculation of the neutron fluence, the methodololgy should describe how the neutron fluence is calculated.

Neutron Fluence Methodology

WCAP-14040, Revision 3, includes a revised Section 2.2. The revised section includes plant-specific transport calculations and the validity of the calculations. For the neutron transport calculations, the applicant is using the two-dimensional discrete ordinates code, DORT (Reference 1) with the BUGLE-96 cross section library (Reference 2). Approximations include a P_5 Legendre expansion for anisotropic scattering and a S_{16} order of angular quadrature. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Two dimensional flux solutions $\Phi(r, \theta, z)$ are constructed using (r,θ) and (r,z) distributions. Extreme cases, with respect to power distribution arising from part-length fuel assemblies, use the three-dimensional TORT Code (Reference 1) with the BUGLE-96 cross section library. Source distribution is obtained from a burn-up weighted average of the power distributions of individual fuel cycles. The method accounts for source energy spectral effects and neutrons/fission due to burnup by tracking the concentration of U-235, U-238, Pu-239, Pu-240, and Pu-241. Mesh spacing accounts for flux gradients and material interfaces.

The proposed methodology, as outlined above, adheres to the guidance of RG 1.190, and therefore, is acceptable.

Validation of Transport Calculations

The Westinghouse validation is structured in four parts:

- comparison to pool critical assembly (PCA) simulator results (Reference 3),
- comparison to calculations in the H. B. Robinson benchmark (Reference 4),
- comparison to a measurement database from pressurized water reactor (PWR) surveillance capsules, and

- an analytical sensitivity study addressing the uncertainty components of the transport calculations.

Comparisons of calculated results to the corresponding PCA measured quantities establish the adequacy of the basic transport calculation and the associated cross sections. Comparison to the H.B. Robinson benchmark addresses uncertainties related to the method and generally to the neutron exposure. Comparisons to the PWR database provides an indication of the presence of a bias and of the uncertainty of the calculated value with respect to the corresponding measured values. Finally, the analytical sensitivity study validates the overall uncertainties whether from the methodology or the lack of precise knowledge of the input parameters.

Comparison of the measured data to the calculations was performed on the basis of measured/calculated (M/C) ratios, and with best estimate values calculated using least squares adjusted measured values. The least squares adjustment is based on weighing individual measurements based on spectral coverage. Comparisons are done before and after spectral adjustments. This method is addressed in RG 1.190, as well as in the ASTM Standard E944-96.

The NRC staff requested that the WOG address the completeness of its database. By letter dated October 20, 2003, the WOG responded by indicating that all of the surveillance capsules analyzed with the proposed methodology (DORT and BUGLE-96) are included in the database. The NRC staff found the response acceptable.

The NRC staff concludes that the proposed benchmarking methodology adheres to the guidance in RG 1.190 and to ASTM standards, and therefore, is acceptable.

Requirement 7: Regarding the low temperature overpressure protection/cold overpressure mitigating system, the lift setting limits for the power operated relief valves should be developed using NRC-approved methodologies.

The method in this section is identical to the existing method in the approved Revision 2 of WCAP-14040. The thermal hydraulics analysis for the mass and heat input transients is using the same specialized version of LOFTRAN, which was approved in Revision 2.

The cold overpressure mitigating system is the same as in the approved version, and therefore, the NRC staff finds it acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the information provided in WCAP-14040, Revision 3, related to the requirements of GL 96-03, as cited in Section 3.0 of this SE, and finds WCAP-14040, Revision 3, to be acceptable for referencing as a PTLR methodology, subject to the following conditions:

-7- provision 2 in the table of Attachment 1 to

a. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements in GL 96-03 related to the RPV material surveillance program ~~issues.~~ discussed

b. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

c. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

5.0 REFERENCES

1. "DOORS 3.1, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996.

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

3. NUREG/CR-6454 (ORNL/TM-1320~~4~~ 5), "Pool Critical Assembly Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, July 1997.

4. NUREG/CR-6453 (ORNL/TM-13204), "H.B. Robinson Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, February 1998.

Principal Contributors: M. Mitchell
L. Lois

Date: February 2, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 2, 2004

RECEIVED
FEB 04 2004
WOG PROJECT OFFICE

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION OF TOPICAL REPORT WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES" (TAC NO. MB5754)

Dear Mr. Bischoff:

On May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" to the staff for review. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted the revised TR for staff review by letter dated October 20, 2003. Enclosed for the WOG's review and comment is a copy of the staff's draft safety evaluation (SE) for TR WCAP-14040, Revision 3.

Twenty working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes, and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes. Number the lines in the marked-up SE sequentially and provide a summary table of the proposed changes.

If you have any questions, please contact Drew Holland at (301) 415-1436.

Sincerely,

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Draft Safety Evaluation

cc w/encl: See next page

Westinghouse Owners Group Project No. 694

cc:
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Regulatory Compliance and Plant Licensing
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMIT CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P-T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety. Based on questions posed by the NRC staff necessitating clarification of statements or editorial changes, the WOG revised WCAP-14040, Revision 3, and submitted the revised TR for NRC staff review and approval by letter dated October 20, 2003.

2.0 REGULATORY EVALUATION

Four specific topics are addressed in the context of the development of a PTLR methodology: (1) the calculation of neutron fluences for the RPV and RPV surveillance capsules; (2) the evaluation of RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these RPV material properties and the establishment of cold overpressure mitigating system (COMS) setpoints to protect the RPV from brittle failure; and (4) the development of an RPV material surveillance program to monitor changes in RPV material properties due to radiation. Regulatory requirements related to the four topics noted above are addressed in Appendices G and H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50). Appendix G to 10 CFR Part 50 provides requirements related to RPV P-T limit development and directly or indirectly addresses topics (1) through (3) above. Appendix H to 10 CFR Part 50 defines regulatory requirements related to RPV material surveillance programs and addresses topic (4) above.

For the staff's review of WCAP-14040, Revision 3, several additional guidance documents were used. NRC Standard Review Plan (SRP) Sections 5.2.2, "Overpressure Protection," 5.3.1, "Reactor Vessel Materials," and 5.3.2, "Pressure-Temperature Limits," provide specific review guidance related to RPV material property determination, P-T limit development, and COMS performance. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes analysis procedures acceptable to the NRC staff for the purpose of assessing RPV material property changes due to radiation. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," addresses NRC staff expectations for an acceptable fluence calculation methodology. American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," provides guidance on the establishment of RPV material surveillance programs and editions of ASTM E 185 are incorporated by reference into Appendix H to 10 CFR Part 50. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix G provides specific requirements regarding the development of P-T limit curves.

Finally, specific guidance regarding topics and the level of detail to which they must be addressed as part of an acceptable PTLR methodology is given in GL 96-03.

3.0 TECHNICAL EVALUATION

The technical requirements to be addressed in an acceptable PTLR methodology are provided under the column heading "Minimum Requirements to be Included in Methodology" in the table entitled "Requirements for Methodology and PTLR" in Attachment 1 to GL 96-03. Summarized versions of the seven requirements are given below, along with the staff's technical evaluation of information in WCAP-14040, Revision 3, related to each requirement.

Requirement 1: Regarding the reactor vessel material surveillance program, the methodology should briefly describe the surveillance program. The methodology should clearly reference the requirements of Appendix H to 10 CFR Part 50.

The provisions of the methodology described in WCAP-14040, Revision 3, do not specify how the plant-specific RPV surveillance programs should be maintained in order to be in compliance with Appendix H to 10 CFR Part 50. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must submit additional information to address the methodology requirements in GL 96-03 related to RPV material surveillance program issues.

Requirement 2: Regarding the calculation of RPV materials' adjusted reference temperatures (ART) values, the methodology should describe the method for calculating material ART values using RG 1.99, Revision 2.

Information regarding how material ARTs are to be determined within the WCAP-14040, Revision 3, PTLR methodology is provided in Section 2.3 and 2.4 of the TR. In Section 2.3, the determination of initial, unirradiated material properties from Charpy V-notch impact tests and/or nil-ductility drop weight tests is clearly defined. The methodology specified in Section 2.3 accurately incorporates the guidance found in ASME Code Section III, paragraph NB-2331 and additional information in SRP Section 5.3.1.

- 3 -

In Section 2.4 of.the TR, the determination of changes in material properties due to irradiation is addressed, along with the determination of margins necessary to account for uncertainties in initial properties and irradiation damage assessment. The methodology specified in Section 2.4 accurately incorporates the guidance found in RG 1.99, Revision 2.

The NRC staff, therefore, determined that the methodology described for determining material ART values in WCAP-14040, Revision 3, was consistent with the guidance provided in the ASME Code, SRP Section 5.3.1, and RG 1.99, Revision 2, and was, therefore, acceptable.

Requirement 3: Regarding the development of RPV P-T limit curves, the methodology should describe the application of fracture mechanics-based calculations in constructing P-T limit curves based on the provisions of Appendix G to Section XI of the ASME Code and SRP Section 5.3.2.

Basic and optional elements of the methodology for RPV P-T limit curve development in WCAP-14040, Revision 3, are given in Sections 2.5, 2.6, 2.7, 2.8, and Appendix A of the TR.

In Section 2.5 of the TR, the fracture toughness-based guidelines from Appendix G to Section XI of the ASME Code are specified (based on the 1995 Edition through 1996 Addenda of the ASME Code). Notably, specific reference is made to the use of: (1) the ASME Code lower bound dynamic crack initiation/crack arrest (K_{IA}) fracture toughness curve; (2) the use of a postulated flaw that has a depth of one-quarter of the wall thickness and a 6:1 aspect ratio; and (3) the use of a structural factor of 2 on primary membrane stress intensities (K_{IM}) when evaluating normal heatup and cooldown and a structural factor of 1.5 on K_{IM} when evaluating hydrostatic/leak test conditions.

Optional guidelines for P-T limit curve development are also addressed in WCAP-14040, Revision 3. The option of using the ASME Code static crack initiation fracture toughness curve (K_{IC}), as given in ASME Code Case N-640, is addressed in Sections 2.5 and 2.8. The option of using ASME Code Case N-588, which enables the postulation of a circumferentially-oriented flaw (with appropriate stress magnification factors) when evaluating a circumferential weld, is addressed in Section 2.8. WCAP-14040, Revision 3, notes, however, that licensee use of the provisions of either ASME Code Case N-640 or N-588 requires, in accordance with 10 CFR 50.60(b), an exemption if the provisions of the Code Case are not contained in the edition of the ASME Code included in a facility's licensing basis. Appendix A to WCAP-14040, Revision 3, provides additional details regarding the application of optional ASME Code Cases and includes copies of ASME Code Case N-588, N-640, and N-641 (which effectively combines the provisions of N-588 and N-640 into a single Code Case).

A detailed discussion of the calculational methodology for P-T limit curve generation is given in Section 2.6 of the TR. Specific equations are given for the determination of primary membrane stresses due to internal pressure and membrane and bending stresses due to thermal gradients. Equations related to the generation of P-T limit curves for steady-state conditions, finite heatup rates, finite cooldown rates, and hydrostatic/leak test conditions are given. The equations given in WCAP-14040, Revision 3, are equivalent to those provided in Section XI of the ASME Code and consistent with the guidance given in SRP Section 5.3.2.

- 4 -

Therefore, the NRC staff has concluded that the basic methodology specified in WCAP-14040, Revision 3, for establishing P-T limit curves meets the regulatory requirements of Appendix G to 10 CFR Part 50 and the guidance provided in SRP Section 5.3.2. However, the NRC staff has concluded that the discussion provided in WCAP-14040, Revision 3, regarding the use of optional guidelines for the development of P-T limit curves, including the use of ASME Code Cases N-588, N-640, and N-641 is not acceptable. The NRC staff has concluded, based on guidance provided by the NRC's Office of the General Counsel, that licensees do not need to obtain exemptions to use the provisions of ASME Code Case N-588, N-640, or N-641. The basis for this decision is as follows. Appendix G to 10 CFR Part 50 references the use of ASME Code Section XI, Appendix G and defines the acceptable Editions and Addenda of the Code by reference to those endorsed in 10 CFR 50.55a. The 2003 Edition of 10 CFR Part 50, 10 CFR 50.55a, endorses editions and addenda of ASME Section XI up through the 1998 Edition and 2000 Addenda. The provisions of N-588, N-640, and N-641 have been directly incorporated into the Code in the 2000 Addenda version of ASME Section XI, Appendix G. Therefore, licensees may freely make use of the provisions in Code Cases N-588, N-640, and N-641 by using the methodology in the 2000 Addenda version of ASME Section XI without the need for an exemption. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

Requirement 4: Regarding the development of RPV P-T limit curves, the methodology should describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied when constructing P-T limit curves.

Minimum temperature requirements regarding the material in the highly stressed region of the RPV flange are given in Appendix G to 10 CFR Part 50. Information provided in Sections 2.9 and 2.10 of the TR addresses the incorporation of minimum temperature requirements into the development of P-T limit curves. In Section 2.9, the 10 CFR Part 50, Appendix G requirements are cited. WCAP-14040, Revision 3, goes on to note that there is an effort underway to revise or eliminate these requirements based on information contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." However, WCAP-14040, Revision 3, states that until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

WCAP-14040, Revision 3, provides supplemental information in Section 2.10 regarding the establishment of RPV boltup temperature, specifically that the minimum boltup temperature should be 60 °F or equal to the highest material reference temperature in the highly stressed RPV flange region, whichever is higher (i.e., more conservative). Although no specific requirements related to boltup temperature are provided in Appendix G to 10 CFR Part 50, the information in WCAP-14040, Revision 3, is consistent with other, related requirements in Appendix G to 10 CFR Part 50 and in Appendix G to Section XI of the ASME Code.

The NRC staff concludes that the methodology specified in WCAP-14040, Revision 3, addresses RPV minimum temperature requirements in a way which is consistent with Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code and is, therefore, acceptable.

Requirement 5: Regarding the calculation of RPV materials' ARTs, the methodology should describe how the data from multiple surveillance capsules may be used in ART calculations.

Requirement 2 of Section 2.4 of WCAP-14040, Revision 3, addresses the determination of changes in material properties due to irradiation. This information includes a description of how surveillance capsule test results may be used to calculate RPV material properties in a manner which is consistent with Section C.2.1 of RG 1.99, Revision 2, and other NRC staff guidance.

The NRC staff has reviewed the information in Section 2.4 of the TR and determined that it is consistent with NRC staff guidance, including RG 1.99, Revision 2, and is, therefore, acceptable.

Requirement 6: Regarding the calculation of the neutron fluence, the methodololgy should describe how the neutron fluence is calculated.

Neutron Fluence Methodology

WCAP-14040, Revision 3, includes a revised Section 2.2. The revised section includes plant-specific transport calculations and the validity of the calculations. For the neutron transport calculations, the applicant is using the two-dimensional discrete ordinates code, DORT (Reference 1) with the BUGLE-96 cross section library (Reference 2). Approximations include a P_5 Legendre expansion for anisotropic scattering and a S_{16} order of angular quadrature. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Two dimensional flux solutions $\Phi(r, \theta, z)$ are constructed using (r,θ) and (r,z) distributions. Extreme cases, with respect to power distribution arising from part-length fuel assemblies, use the three-dimensional TORT Code (Reference 1) with the BUGLE-96 cross section library. Source distribution is obtained from a burn-up weighted average of the power distributions of individual fuel cycles. The method accounts for source energy spectral effects and neutrons/fission due to burnup by tracking the concentration of U-235, U-238, Pu-239, Pu-240, and Pu-241. Mesh spacing accounts for flux gradients and material interfaces.

The proposed methodology, as outlined above, adheres to the guidance of RG 1.190, and therefore, is acceptable.

Validation of Transport Calculations

The Westinghouse validation is structured in four parts:

- comparison to pool critical assembly (PCA) simulator results (Reference 3),
- comparison to calculations in the H. B. Robinson benchmark (Reference 4),
- comparison to a measurement database from pressurized water reactor (PWR) surveillance capsules, and

- an analytical sensitivity study addressing the uncertainty components of the transport calculations.

Comparisons of calculated results to the corresponding PCA measured quantities establish the adequacy of the basic transport calculation and the associated cross sections. Comparison to the H.B. Robinson benchmark addresses uncertainties related to the method and generally to the neutron exposure. Comparisons to the PWR database provides an indication of the presence of a bias and of the uncertainty of the calculated value with respect to the corresponding measured values. Finally, the analytical sensitivity study validates the overall uncertainties whether from the methodology or the lack of precise knowledge of the input parameters.

Comparison of the measured data to the calculations was performed on the basis of measured/calculated (M/C) ratios, and with best estimate values calculated using least squares adjusted measured values. The least squares adjustment is based on weighing individual measurements based on spectral coverage. Comparisons are done before and after spectral adjustments. This method is addressed in RG 1.190, as well as in the ASTM Standard E944-96.

The NRC staff requested that the WOG address the completeness of its database. By letter dated October 20, 2003, the WOG responded by indicating that all of the surveillance capsules analyzed with the proposed methodology (DORT and BUGLE-96) are included in the database. The NRC staff found the response acceptable.

The NRC staff concludes that the proposed benchmarking methodology adheres to the guidance in RG 1.190 and to ASTM standards, and therefore, is acceptable.

Requirement 7: Regarding the low temperature overpressure protection/cold overpressure mitigating system, the lift setting limits for the power operated relief valves should be developed using NRC-approved methodologies.

The method in this section is identical to the existing method in the approved Revision 2 of WCAP-14040. The thermal hydraulics analysis for the mass and heat input transients is using the same specialized version of LOFTRAN, which was approved in Revision 2.

The cold overpressure mitigating system is the same as in the approved version, and therefore, the NRC staff finds it acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the information provided in WCAP-14040, Revision 3, related to the requirements of GL 96-03, as cited in Section 3.0 of this SE, and finds WCAP-14040, Revision 3, to be acceptable for referencing as a PTLR methodology, subject to the following conditions:

a. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements in GL 96-03 related to RPV material surveillance program issues.

b. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

c. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

5.0 REFERENCES

1. "DOORS 3.1, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Shielding Information Center (RSIC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996.

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

3. NUREG/CR-6454 (ORNL/TM-13204), "Pool Critical Assembly Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, July 1997.

4. NUREG/CR-6453 (ORNL/TM-13204), "H.B. Robinson Pressure Vessel Benchmark," by I. Remek and F.B.K. Kam, Oak Ridge National Laboratory, February 1998.

Principal Contributors: M. Mitchell
L. Lois

Date: February 2, 2004

October 20, 2003
WOG-03-550

WCAP-14040 Rev. 3
Project Number 694

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

Attention: Chief, Information Management Branch,
Division of Program Management

Subject: **<u>Westinghouse Owners Group Response to Request for Additional Information on WCAP-14040 Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (TAC No. MB5754)</u>**

References:

1. WOG Letter, R. Bryan to Document Control Desk, "Transmittal of WCAP-14040, Rev. 3, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," OG-02-018, May 23, 2002.

2. NRC Letter, D. Holland to G. Bischoff, "Request for Additional Information - WCAP-14040, Revision 3, Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," TAC NO. 5754, June 18, 2003.

In May 2002, the Westinghouse Owners Group submitted WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for approval (Ref. 1). In June 2003, the NRC issued Requests for Additional Information (RAIs) concerning WCAP-14040, Rev. 3 (Ref. 2).

Attachment 1 to this letter contains the responses to the RAIs. Attachment 2 contains revisions to the affected pages of WCAP-14040, Rev. 3 that incorporate the responses to the RAIs. Attachment 3 contains revisions to Section 2.2 "Neutron Fluence Methodology" of WCAP-14040, Rev. 3. Although not made in response to any RAI, the changes to Section 2.2 of WCAP-14040, Rev. 3 were made to:

- Discuss how the current neutron fluence methodology follows the guidance contained in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001.

- Revise the text and benchmarking results to reflect the use of the BUGLE-96 ENDF/B-VI based cross-section library. The BUGLE-96 library provides an improved calculation relative to the previously used BUGLE-93 data set for some comparisons, particularly in the vessel wall and at ex-vessel dosimetry locations.

- Revise the discussion of the current version of the DORT code currently used.

October 20, 2003
WOG-03-550

- Revise the text to reflect that consistent with the guidance of Regulatory Guide 1.190, the final results for the pressure vessel fluence projections are based on the plant specific transport calculations, and that the dosimetry data is only used to validate the calculated results.

The approved version of WCAP-14040 that will be issued following receipt of the NRC Safety Evaluation will incorporate the changes contained in Attachments 2 and 3.

If you require further information, feel free to contact Mr. Ken Vavrek, Westinghouse Owners Group Project Office at 412-374-4302.

Sincerely,

Frederick P. "Ted" Schiffley, II
Chairman, Westinghouse Owners Group

Attachments

cc: WOG Management Committee
WOG Materials Subcommittee
WOG Licensing Subcommittee
WOG Project Management Office
S.L. Anderson
J. D. Andrachek
W.H. Bamford
T.J. Laubham
J. Perock
H. A. Sepp
D. Holland, USNRC OWFN 07 E1 (1L, 1E) (via Federal Express)

Attachment 1

Responses to NRC Request For Additional Information on WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"

1. Section 2.3, page 2-5, Branch Technical Position MTEB 5-2 does not give fracture toughness "requirements." Revise WCAP-14040, Revision 3, to refer to the information in MTEB 5-2 as "guidelines" rather than "requirements."

   Response to RAI 1:

   The first sentence in the last paragraph of Section 2.3 on page 2-5 will be revised to "fracture toughness guidelines" rather than "fracture toughness requirements."

2. Section 2.4, page 2-6, when referring to the "Ai" term in Equation 2.4-3, revise your definition which refers to it as the "measured value of ΔRTNDT" - instead call it the "measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, fi."

   Response to RAI 2:

   The fifth paragraph in Section 2.4 on page 2-6 will be revised to "the measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, f_i."

3. Section 2.4, page 2-7, revise the sentence which reads, "If the measured value exceeds the predicted value (ΔRTNDT + 2σΔ), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology," to state "If the measured value exceeds the predicted value (ΔRTNDT + 2σΔ), a supplement to the PTLR methodology must be provided for NRC staff review and approval to demonstrate how the results affect the approved methodology."

   Response to RAI 3:

   The last sentence in the second paragraph on page 2-7 will be revised to state that "a supplement to the PTLR must be submitted for NRC review and approval..."

4. Section 2.5, page 2-7, it is stated that Kia is the reference fracture toughness curve in Appendix G to Section XI of the ASME Code. Clarify this to note that this refers to Editions of the Code through the 1995 Edition/1996 Addenda. The most recent Edition and Addenda of the Code (1998 Edition through 2000 Addenda) incorporated by reference into 10 CFR 50.55a, however, uses Kic as the reference fracture toughness curve.

   Response to RAI 4:

   The reference to Appendix G, to Section XI of the ASME Code will be clarified that it is referring to the 1995 Edition through the 1996 Addenda in the first sentence of the second paragraph of Section 2.5 on page 2-7.

5. Section 2.5, page 2-8, the "note" regarding the use of a 1.223 vs. 1.233 coefficient in the Kia equation is meaningless and confusing unless one also explains that there was a typographical error in the 1989 Edition of Section XI, Appendix G (i.e., where the 1.233 was used). Revise WCAP-14040, Revision 3, to either eliminate this note or revise the note to offer additional explanation regarding the historical basis for the 1.223 vs. 1.233 issue.

   Response to RAI 5:

   The Note in the first paragraph on page 2-8 discussing the historical basis of 1.223 versus 1.233 will be deleted.

6. Section 2.5, page 2-8, when discussing ASME Code Case N-640, it is not correct to say that an exemption is required to implement N-640 because the NRC has not "endorsed" the Code Case. "Endorsement" implies that it has been included in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability -- ASME Section XI, Division 1." Code Case N-640 would have to be included in the edition of the ASME Code which the licensee has adopted in their facility's licensing basis in order to comply with 10 CFR 50.55a before an exemption is no longer required.

   Response to RAI 6:

   The fifth paragraph on page 2-8 will be revised to delete the text "has not yet been endorsed by the NRC, and therefore use of this Code Case will" and to add the statement "if it is not contained in the edition of the ASME Code included in the unit licensing basis."

7. The statement in Section 2.5, page 2-10, regarding need for an exemption relative to modifying existing 10 CFR Part 50, Appendix G flange requirements should, for consistency be repeated in Section 2.8.

   Response to RAI 7:

   A statement that the flange requirement must be included in the P-T limits unless an exemption request is submitted and approved by the NRC will be added to the fourth paragraph in Section 2.8 on page 2-20.

8. Section 2.6.1, page 2-12, it is stated "[t]hese stress components are used for determining the thermal stress intensity factors, Kit, as described in the following subsection." The following subsection is 2.6.2, "Steady-State Analyses," and it does not address the calculation of Kit. Revise WCAP-14040, Revision 3, to address this apparent inconsistency.

   Response to RAI 8:

   The last sentence in the last paragraph of Subsection 2.6.1 on page 2-12 will be revised to "in subsections 2.6.3 and 2.6.4."

9. Section 2.6.2, page 2-14, and Section 2.6.5, page 2-15, Mm factors of 1.84, 0.918, and 3.18 are given for various reactor pressure vessel wall thickness ranges to be used when steady-state analyses are performed. It is unclear as to where these Mm factors come from (unable to locate them in any edition of ASME Section XI, Appendix G). Further, they are not consistent with what should be the same Mm factors cited on page 2-15. Revise WCAP-14040, Revision 3, to address this apparent inconsistency in the cited Mm factors.

   Response to RAI 9:

   The M_m factors discussed in Subsection 2.6.2 on page 2-14, and in Subsection 2.6.5 on page 2-15 will be deleted.

10. Section 2.7, page 2-19, it should be noted that an exemption is required when a licensee wishes to make use of ASME Code Case N-588. Revise WCAP-14040, Revision 3, accordingly.

   Response to RAI 10:

   A sentence will be added to the first paragraph in Section 2.7 on page 2-19 that states "An exemption request must be submitted and approved by the NRC if Code Case N-588 is not contained in the edition of the ASME Code included in the unit licensing basis."

Attachment 2

Revised WCAP-14040, Revision 3 Pages Incorporating NRC RAIs

The results of this latter comparison expressed in terms of the ratio of adjusted flux to calculated flux (A/C) are summarized as follows for the 82 capsule data base:

PARAMETER	A/C	STD DEV
$\phi(E > 1.0\ MeV)$	1.00	7.3%

As with the comparisons based on the linear average of reaction rate M/C ratios, the comparisons of the least squares adjusted results with the plant specific transport calculations demonstrate that the calculated results are essentially unbiased with an uncertainty well within the 20% criterion established in Regulatory Guide 1.190.

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[8]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[9] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness guidelines in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

2.4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The adjusted reference temperature (ART) for each material in the beltline region is calculated in accordance with Regulatory Guide 1.99, Revision 2[3]. The most limiting ART values (i.e., highest value at 1/4t and 3/4t locations) are used in determining the pressure-temperature limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (2.4\text{-}1)$$

IRT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8] and calculated per Section 2.3. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for that class of material can be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the shift in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF\ f^{(0.28 - 0.10 \log f)} \qquad (2.4\text{-}2)$$

CF (°F) is the chemistry factor and is a function of copper and nickel content. CF is given in Table 1 of Reference 3 for weld metal and in Table 2 in Reference 3 for base metal (Position 1.1 of Regulatory Guide 1.99, Revision 2). In Tables 1 and 2 of Reference 3 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in Regulatory Guide 1.99, Revision 2, Paragraph B.4) become available they may be used to calculate the chemistry factor per Position 2.1 of Regulatory Guide 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n} [A_i\ f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n} [f_i^{(0.28-0.10\log f_i)}]^2} \qquad (2.4\text{-}3)$$

Where "n" is the number of surveillance data points, "A_i" is the measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, "f_i." Where "f_i" is the fluence for each surveillance data point.

If Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a higher value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a lower value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4t), the following formula is used to attenuate the fast neutron fluence (E> 1 MeV) at the specified depth.

$$f = f_{surface}\, e^{(-0.24x)} \qquad (2.4\text{-}4)$$

where $f_{surface}$ 10^{19} n/cm^2, E > 1 MeV) is the value, calculated per Section 2.2, of the neutron fluence at the base metal surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (2.4-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by equation (2.4-2) plus two standard deviations ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be submitted for NRC review and approval to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by Appendix G to 10 CFR Part 50[4]. Margin is calculated by the following equation:

$$\text{Margin} = 2\, [(\sigma_I^2 + \sigma_\Delta^2)]^{0.5} \qquad (2.4\text{-}5)$$

σ_I, is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_I, is estimated from the precision of the test method ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per Regulatory Guide 1.99, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

2.5 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME Code requirements[5] for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates specify that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, the fracture toughness for the metal temperature at that time. Two values of fracture toughness may be used, K_{Ia} or K_{Ic}.

K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G, to Section XI of the ASME Code (1995 Edition through the 1996 Addenda). (Note that in Appendix G, to Section III of the ASME Code, the reference fracture toughness is denoted as K_{IR}, whereas in Appendix G of Section XI, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.) The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \exp[0.0145\,(T - RT_{NDT} + 160)] \qquad (2.5\text{-}1)$$

where,

K_{Ia} = lower bound of dynamic and crack arrest toughness as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RT_{NDT} is the adjusted reference temperature (ART) of Section 2.4.

K_{Ic} is also obtained from Section XI of the ASME Code, for example in Appendix A, and is a lower bound of static fracture toughness. Since heatup and cooldown is a slow process, static properties are appropriate. The K_{Ic} curve is given by the following expression:

$$K_{Ic} = 33.20 + 20.734 \exp [0.0200 (T - RT_{NDT})] \qquad (2.5\text{-}2)$$

The use of the K_{Ic} curve (Section XI, Appendix A) as a basis for developing P-T limit curves is currently contained in ASME Code Case N640. Use of the K_{Ic} fracture toughness will yield less limiting P-T curves, which is clearly a benefit.

However, the use of Code Case 640 presently includes a restriction on the setpoints for the Cold Overpressure Mitigation System (COMS). This maximum pressure for the COMS system is 100% of the pressure allowed by the P-T limit curves. This essentially disallows the use of Code Case N514 in these circumstances, meaning that the COMS system must protect to the actual P-T limit curve, rather than 110 percent, as allowed by Code Case N514.

The use of Code Case N640 requires an exemption under 10CFR50.60 paragraph (b), pertaining to proposed alternatives to the requirements of Appendices G and H, if it is not contained in the edition of the ASME Code included in the unit licensing basis.

The governing equation for generating pressure-temperature limit curves is defined in Appendix G of the ASME Code[5] as follows:

$$C\, K_{IM} + K_{It} < \text{Reference Fracture Toughness} \qquad (2.5\text{-}3)$$

where,

K_{IM} = stress intensity factor caused by membrane (pressure) stress,

K_{It} = stress intensity factor caused by the thermal gradients through the vessel wall,

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

C = 1.5 for hydrostatic and leak test conditions when the reactor core is not critical

Reference Fracture Toughness = K_{Ia} or K_{Ic}, as discussed above

(Note: K_{It} is set to zero for hydrostatic and leak test calculations since these tests are performed at isothermal conditions).

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α are evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r)r\,dr}{r_o^2 - r_i^2} \qquad (2.6.1\text{-}5)$$

E and α are calculated as a function of this equivalent temperature and the Eα product in equation (2.6.1-4) is treated as a constant in the computation of hoop thermal stress.

The linear bending (σ_b) and constant membrane (σ_m) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in Appendix A, to Section XI of the ASME Code[15]. These stress components are used for determining the thermal stress intensity factors, K_{It}, as described in subsections 2.6.3 and 2.6.4.

2.6.2 Steady-State Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at the 1/4t location where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius.

The allowable pressure P(T_c) is a function of coolant temperature, and the pressure temperature curve is calculated for the steady state case at the assumed 1/4t inside surface flaw. First, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t}}{2.0} \qquad (2.6.2\text{-}1)$$

where,

K_{Ia} (T-RT_{NDT}) = allowable reference stress intensity factor as a function of T-RT_{NDT} at 1/4t. (See Sections 2.7 and 2.8 for the new approach using Code Cases N640 and N588.)

Next, the maximum allowable pressure stress is determined using an iterative process and the following three equations:

$$Q = \varphi^2 - 0.212\left(\frac{\sigma_P}{\sigma_y}\right)^2 \qquad (2.6.2\text{-}2)$$

$$\sigma_P = \frac{K_{IM(max)}}{1.1\, M_K \sqrt{\frac{\pi a}{Q}}} \tag{2.6.2-3}$$

$$K_{IP} = 1.1 M_K \sigma_P \sqrt{\frac{\pi a}{Q}} \tag{2.6.2-4}$$

where,

Q	=	flaw shape factor modified for plastic zone size(16),
ϕ	=	is the elliptical integral of the 2nd kind (ϕ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw)(16),
0.212	=	plastic zone size correction factor(16),
σ_P	=	pressure stress,
σ_y	=	yield stress,
1.1	=	correction factor for surface breaking flaws,
M_K	=	correction factor for constant membrane stress (16), M_K as function of relativeflaw depth (a/t) is shown in Figure 2.4,
a	=	crack depth of 1/4t,
K_{IP}	=	pressure stress intensity factor.

The maximum allowable pressure stress is determined by incrementing σ_p from an initial value of 0.0 psi until a pressure stress is found that computes a K_{IP} value within 1.0001 of the $K_{IM(max)}$ value. After the maximum allowable σ_p is found, the maximum allowable internal pressure is determined by

$$P(T_c) = \sigma_P \left[\frac{r_o^2 - r_i^2}{r_o^2 + r_i^2}\right] \tag{2.6.2-5}$$

where,

$P(T_c)$	=	calculated allowable pressure as a function of coolant temperature.

2.6.3 Finite Cooldown Rate Analyses

For each cooldown rate the pressure-temperature curve is calculated at the inside 1/4t location. First, the thermal stress intensity factor is calculated for a coolant temperature at a given time using the following equation from the Welding Research Council[16]:

$$K_{It} = [\sigma_m 1.1 M_K + \sigma_b M_B]\sqrt{\frac{\pi a}{Q}} \qquad (2.6.3\text{-}1)$$

where,

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_K = correction factor for membrane stress[16] (see Figure 2.4),

M_B = correction factor for bending stress[16], M_B as a function of relative flaw depth (a/t) is shown in Figure 2.5.

The flaw shape factor Q in equation (2.6.2-6) is calculated from[16]

$$Q = \phi^2 - 0.212\left(\frac{\sigma_m + \sigma_b}{\sigma_y}\right)^2 \qquad (2.6.3\text{-}2)$$

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \qquad (2.6.3\text{-}3)$$

From $K_{IM(max)}$, the maximum allowable pressure is determined using the iterative process described above and equations (2.6.2-2) through (2.6.2-5).

The steady-state pressure-temperature curve of Section 2.6.2 is compared to the cooldown curves for the 1/4t inside surface flaw at each cooldown rate. At any time, the allowable pressure is the lesser of the two values, and the resulting curve is called the composite cooldown limit curve.

Finally, the 10 CFR Part 50[4] requirement for the closure flange region is incorporated into the cooldown composite curve as described in Section 2.5.

2.6.4 Finite Heatup Rate Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) or (2.5-2) at both the 1/4t and 3/4t locations where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius. At the 3/4t location, a 1/4t flaw is assumed to originate on the outside of the vessel.

For each heatup rate a pressure-temperature curve is calculated at the 1/4t and 3/4t locations. First, the thermal stress intensity factor is calculated at the 1/4t and 3/4t locations for a coolant temperature at a given time using Option 1 or 2 from Section 2.6.3.

Once K_{It} is computed, the maximum allowable membrane (pressure) stress intensity factors at the 1/4t and 3/4t locations are determined using the following equations:

$$\text{At 1/4t,} \quad K_{IM(max)1/4t} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{It}(T_c)_{1/4t}}{2.0} \tag{2.6.4-1}$$

$$\text{At 3/4t,} \quad K_{IM(max)3/4t} = \frac{K_I * (T - RT_{NDT})_{3/4t} - K_{It}(T_c)_{3/4t}}{2.0} \tag{2.6.4-2}$$

From $K_{IM(max)1/4t}$ and $K_{IM(max)3/4t}$, the maximum allowable pressure at both the 1/4t and 3/4t locations is determined using the iterative process described in Section 2.6.2 and equations (2.6.2-2) through (2.6.2-5).

As was done with the cooldown case, the steady state pressure-temperature curve of Section 2.6.2 is compared with the 1/4t and 3/4t location heatup curves for each heatup rate, with the lowest of the three being used to generate the composite heatup limit curve. The composite curve is then adjusted for the 10 CFR Part 50[4] rule for closure flange requirements, as discussed in Section 2.5.

2.6.5 Hydrostatic and Leak Test Curve Analyses

The minimum inservice hydrostatic leak test curve is determined by calculating the minimum allowable temperature at two pressure values (pressure values of 2000 psig and 2485 psig, approximately 110% of operating pressure, are generally used). The curve is generated by drawing a line between the two pressure-temperature data points. The governing equation for generating the hydrostatic leak test pressure-temperature limit curve is defined in Appendix G, Section X1, of the ASME Code[5] as follows:

$$1.5\, K_{IM} < K_{Ia} \tag{2.6.5-1}$$

where, K_{IM} is the stress intensity factor caused by the membrane (pressure) stress and K_{Ia} is the reference stress intensity factor as defined in equation (2.5-1). Note that the thermal stress intensity factor is neglected (i.e., $K_{It} = 0$) since the hydrostatic leak test is performed at isothermal conditions.

The pressure stress is determined by,

$$\sigma_P = \left[\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2}\right] P \qquad (2.6.5\text{-}2)$$

where,

P = the input pressure (generally 2000 and 2485 psig)

Next, the pressure stress intensity factor is calculated for a 1/4t flaw by,

$$K_{IM} = \left[1.1 M_K \sqrt{\frac{\pi a}{Q}}\right] \sigma_P \qquad (2.6.5\text{-}3)$$

The K_{IM} result is multiplied by the 1.5 factor of equation (2.5-2) and divided by 1000,

$$K_{HYD} = \frac{1.5 K_{IM}}{1000} \qquad (2.6.5\text{-}4)$$

Finally, the minimum allowable temperature is determined by setting K_{HYD} to K_{Ia} in equation (2.5-1) and solving for temperature T:

$$T = \frac{\ell n\left[\frac{(K_{HYD} - 26.78)}{1.223}\right]}{0.0145} + RT_{NDT} - 160.0 \qquad (2.6.5\text{-}5)$$

The 1983 Amendment to 10CFR50[4] has a rule which addresses the test temperature for hydrostatic pressure tests. This rule states that, when there is no fuel in the reactor vessel during hydrostatic pressure tests or leak tests, the minimum allowable test temperature must be 60°F above the adjusted reference temperature of the beltline region material that is controlling. If fuel is present in the reactor vessel during hydrostatic pressure tests or leak tests, the requirements of this section and Section 2.5 must be met.

2.7 1996 ADDENDA TO ASME SECTION XI, APPENDIX G METHODOLOGY

ASME Section XI, Appendix G was updated in 1996 to incorporate the most recent elastic solutions for K_I due to pressure and radial thermal gradients. The new solutions are based on finite element analyses for inside surface flaws performed at Oak Ridge National Laboratories and sponsored by the NRC, and work published for outside surface flaws. These solutions provide results that are very similar to those obtained by using solutions previously developed by Raju and Newman.

This revision provides consistent computational methods for pressure and thermal K_I, for thermal gradients through the vessel wall at any time during the transient. Consistent with the original version of

Appendix G, no contribution for crack face pressure is included in the K_I due to pressure, and cladding effects are neglected.

Using these elastic solutions in the low temperature region will provide some relief to restrictions associated with reactor operation at relatively low temperatures. Although the relief is relatively small in terms of the absolute allowable pressure, the benefits are substantial, because even a small increase in the allowable pressure can be a significant percentage increase in the operating window at relatively low temperatures. Implementing this revision results in a safety benefit (reduced likelihood of lifting COMS relief valves), with no reduction in vessel integrity.

The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension:

$$K_{Im} = M_m \times (pR_i / t) \qquad (2.7\text{-}1)$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,

M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,

M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 3.09 for $\sqrt{t} > 3.464$

where,

p = internal pressure,

Ri = vessel inner radius, and

t = vessel wall thickness.

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where $M_b = 0.667\ M_m$

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{It} = 0.953 \times 10^{-3}\ CR\ t^{2.5} \qquad (2.7\text{-}2)$$

where:

CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{It} = 0.753 \times 10^{-3}\ HU\ t^{2.5} \qquad (2.7\text{-}3)$$

where:

HU = the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from ASME Section XI, Appendix G, Figure G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Section XI, Appendix G, Figure G-2214-2 for the maximum thermal K_I .

1. The maximum thermal K_I relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2) of Appendix G to ASME Section XI.

2. Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (2.7\text{-}4)$$

or similarly, K_{It} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \qquad (2.7\text{-}5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the equation:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.7\text{-}6)$$

where x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Once K_{Ia} (As calculated via Equation 2.5-1) is known, the pressure can be solved using Equation 2.5-3 with the newly calculated K_{It} and new equation for K_{IM}.

$$C * [M_m \times (pR_i / t)] + K_{It} < K_{Ia}$$

where:

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

C = 1.5 for hydrostatic and leak test conditions when the reactor core is not critical

This results in a pressure equation as follows:

$$p = \frac{[K_{It} - K_{Ia}]}{C * M_m * (R_i / t)} \qquad (2.7\text{-}7)$$

Note that K_{It} is equal to zero for steady state and hydrostatic leak test conditions. In addition, K_{Ia} and K_{It} must be calculated individually for inside and outside flaw locations (i.e., the ¼T and ¾T wall locations) and the minimum pressure must be used from these two locations. [Note: K_{Ia} for ¼ T steady state is not the same as K_{Ia} for ¼T thermal conditions since the wall temperature is equal to the water temperature in steady state, but is not the case under thermal conditions.]

2.7 CODE CASES N-640 FOR K_{Ic} and N-588 FOR CIRCUMFERENTIAL WELD FLAWS

2.8.1 ASME Code Case N-640

In February of 1999, the ASME Code approved Code Case N-640 which allows the use of the reference fracture toughness curve K_{Ic}, as found in Appendix A of Section XI, in lieu of Figure G-2110-1 in Appendix G for the development of pressure-temperature limit curves. (This is also described in Section 2.5 herein). Thus, when developing pressure-temperature limit curves, it is acceptable to calculate the reference stress intensity via Equation 2.5-2, in lieu of Equation 2.5-1. In addition, the K_{Ic} can be substituted for K_{Ia} in Equations 2.5-3, 2.6.2-1, 2.6.3-3, 2.6.4-1, 2.6.4-2, 2.6.5-1 and 2.7-7. An exemption request must be submitted and approved by the NRC if ASME Code Case N-640 is not contained in the edition of the ASME Code included in the unit licensing basis.

2.8.2 ASME Code Case N-588

In 1997, ASME Section XI, Appendix G was revised to add a methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588. An exemption request must be submitted and approved by the NRC if Code Case N-588 is not contained in the edition of the ASME Code included in the unit licensing basis.

The original ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds. In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Section XI, Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of G-2120 is

$$K_{IM} = M_m \times (PR_i/t)$$

Where, M_m for an inside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$

$$M_m = 0.443 \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$

$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Similarly, M_m for an outside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$

$$M_m = 0.443 \sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$

$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circumferential weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology.

2.9 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule 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 pre-service hydrostatic test pressure (3106 psig), which is 621 psig for a typical Westinghouse reactor vessel design.

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

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{Ic} in the development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," concluded that the integrity of the closure head/vessel 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. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves, once the requirements of 10CFR50 Appendix G are changed. However, until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

2.10 MINIMUM BOLTUP TEMPERATURE

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the minimum boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT} whichever is higher.

reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2.

3.4 ENABLE TEMPERATURE FOR COMS

The enable temperature is the temperature below which the COMS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is very conservative, and is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3.

A significant improvement in the enable temperature can be obtained by application of code case N641. This code case incorporates the benefits of code cases N588, and N640. The resulting enable temperatures for the Westinghouse designs obtained using code case N641 are listed below.

The use of Code Case N641 has not yet been approved by the NRC, and therefore the use of this Code Case will require approval of an exemption request, as discussed in under 10CFR50.60 paragraph (b), pertaining to proposed alternatives to the requirements of Appendices G and H.

Vessel Type	Axial Flaw	Circumferential Flaw
2 – loop	RT_{NDT} + 23F	Any temperature
3 – loop	RT_{NDT} + 30F	RT_{NDT} – 174F
4 – loop	RT_{NDT} + 34F	RT_{NDT} – 110F

The RCS cold leg temperature limitation for starting an RCP is the same value as the COMS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the COMS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

Table A-1 Status of ASME Nuclear Code Cases Associated with the P-T Limit Curve/COMS Methodology

Code Case	Title	Approved by ASME	Section XI of the ASME Code	Exemption Request Granted
514	Low Temperature Overpressure Protection	2/12/92	1995 Edition through the 1996 Addenda	Yes
588	Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel	12/12/97	1998 Edition through the 2000 Addenda	Yes
640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves	2/26/99	1998 Edition through the 2000 Addenda	Yes
641	Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirement	1/17/00	1998 Edition through the 2000 Addenda	Yes

Attachment 3

Revisions to Section 2.2 "Neutron Fluence Methodology" of WCAP-14040, Revision 3

2.2 NEUTRON FLUENCE METHODOLOGY

The methodology used to provide neutron exposure evaluations for the reactor pressure vessel is based on the requirements provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."[6]. The vessel exposure projections are based on the results of plant specific neutron transport calculations that are validated by benchmarking of the analytical approach, comparison with industry wide power reactor data bases, and finally, by comparison to plant specific surveillance capsule and reactor cavity dosimetry data. In the validation process, the measurement data are used solely to confirm the accuracy of the transport calculations. The measurements are not used in any way to modify the results of the transport calculations.

2.2.1 Plant Specific Transport Calculations

In the application of the methodology to the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, plant specific forward transport calculations are carried out on a fuel cycle specific basis using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = [\phi(r,\theta)] * [\phi(r,z)]/[\phi(r)]$$

where:

$\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution,

$\phi(r,\theta)$ is the transport solution in r,θ geometry,

$\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and

$\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

All of the transport calculations are carried out using the DORT discrete ordinates code Version 3.1[7] and the BUGLE-96 cross-section library[11]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering is treated with a P_5 legendre expansion and the angular discretization is modeled with an S_{16} order of angular quadrature. Energy and space dependent core power distributions as well as system operating temperatures are treated on a fuel cycle specific basis. The synthesis procedure combining the $\phi(r,\theta)$, $\phi(r,z)$, and $\phi(r)$ transport solutions into the three-dimensional flux/fluence maps within the reactor geometry is accomplished by post-processing the output files generated by the [r,θ], [r,z], and [r] DORT calculations.

In some extreme cases where part length poisons or shielded fuel assemblies have been inserted into the reactor core to reduce the fluence locally in the vicinity of key vessel materials, the calculational approach may be modified to use either a multi-channel synthesis approach or a fully three-dimensional technique. For the full three-dimensional analysis, the TORT[7] three-dimensional discrete ordinates transport code is used in conjunction with either the BUGLE-96 ENDF/B-VI based library to provide a complete solution without recourse to the use of flux synthesis techniques.

In developing an analytical model of the reactor geometry, nominal design dimensions are normally employed for the various structural components. In some cases as-built dimensions are available; and, in those instances, the more accurate as-built data are used for model development. However, for the most part, as built dimensions of the components in the beltline region of the reactor are not available, thus, dictating the use of design dimensions. Likewise, water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor are normally taken to be representative of full power operating conditions. The reactor core itself is treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The spatial mesh description used in the transport models depends on the overall size of the reactor and on the complexity required to model the core periphery, the in-vessel surveillance capsules, and the details of the reactor cavity. Mesh sizes are chosen to assure that proper convergence of the inner iterations is achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the r,θ calculations is set at a value of 0.001.

The mesh selection process results in a smaller spatial mesh in regions exhibiting steep gradients, in material zones of high cross-section (Σ_t), and at material interfaces. In the modeling of in-vessel surveillance capsules, a minimum set of 3 radial by 3 azimuthal mesh are employed within the test specimen array to assure that sufficient information is produced for use in the assessment of fluence gradients within the materials test specimens, as well as in the determination of gradient corrections for neutron sensors. Additional radial and azimuthal mesh are employed to model the capsule structure surrounding the materials test specimen array. In modeling the stainless steel baffle region at the periphery of the core, a relatively fine spatial mesh is required to adequately describe this rectilinear component in r,θ geometry. In performing this x,y to r,θ transition, care is taken to preserve both the thickness and volume of the steel region in order to accurately address the shielding effectiveness of the component.

The spatial variation of the neutron source is generally obtained from a burnup weighted average of the respective power distributions from individual fuel cycles. These spatial distributions include pinwise gradients for all fuel assemblies located at the periphery of the core and typically include a uniform or flat distribution for fuel assemblies interior to the core. The spatial component of the neutron source is transposed from x,y to [r,θ], [r,z], and [r] geometry by overlaying the mesh schematic to be used in the transport calculation on the pin by pin array and then computing the appropriate relative source applicable to each spatial interval within the reactor core.

These x,y to [r,θ], [r,z], and [r] transpositions are accomplished by first defining a fine mesh working array. The sizes of the fine mesh are usually chosen so that there is at least a 10x10 array of fine mesh over the area of each fuel pin at the core periphery. The coordinates of the center of each fine mesh interval and its associated relative source strength are assigned to the fine mesh based on the pin that is coincident with the center of the fine mesh. In the limit as the sizes of the fine mesh approach zero, this technique becomes an exact transformation.

Each space mesh in the transport geometry is checked to determine if it lies totally within the area of a particular fine working mesh. If it does, the relative source of that fine mesh is assigned to the transport space mesh. If, on the other hand, the transport space mesh covers a part of one or more fine mesh, then the relative source assigned to the transport mesh is determined by an area weighting process as follows:

$$P_m = \frac{\sum_i A_i P_i}{\sum_i A_i}$$

where:

P_m = the relative source assigned to transport mesh m.

A_i = the area of fine working mesh i within transport mesh m.

P_i = the relative source within fine working mesh i.

The energy distribution of the source is determined on a fuel assembly specific basis by selecting a fuel assembly burnup representative of conditions averaged over each fuel cycle and an initial enrichment characteristic for each assembly. From this average burnup and initial enrichment, a fission split by isotope including ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu is derived; and, from that fission split, composite values of energy release per fission, neutron yield per fission, and fission spectrum are determined for each fuel assembly. These composite values are then combined with the spatial distribution to produce the overall absolute neutron source for use in the transport calculations.

2.2.2 Validation of the Transport Calculations

The validation of the methodology described in Section 2.2.1 is based on the guidance provided in Regulatory Guide 1.190. In particular, the validation consists of the following stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL)[12].

2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment[22].

3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant specific transport calculations used in the exposure assessments.

4. Comparisons of calculations with a measurements data base obtained from a large number of surveillance capsules withdrawn from a variety of pressurized water reactors.

At each subsequent application of the methodology, comparisons are made with plant specific dosimetry results to demonstrate that the plant specific transport calculations are consistent with the uncertainties derived from the methods qualification.

The first stage of the methods validation addresses the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This stage, however, does not test the accuracy of commercial core neutron source calculations nor does it address uncertainties in operational or geometric variables that impact power reactor calculations. The second stage of the validation addresses uncertainties that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations. The third stage of the validation identifies the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant specific parameters. The overall calculational uncertainty is established from the results of these three stages of the validation process.

The following summarizes the uncertainties determined from the results of the first three stages of the validation process:

PCA Benchmark Comparisons	3%
H. B. Robinson Benchmark Comparisons	3%
Analytical Sensitivity Studies	11%
Internals Dimensions	3%
Vessel Inner Radius	5%
Water Temperature	4%
Peripheral Assembly Source Strength	5%
Axial Power Distribution	5%
Peripheral Assembly Burnup	2%
Spatial Distribution of the Source	4%
Other Factors	5%

The category designated "Other Factors" is intended to attribute an additional uncertainty to other geometrical or operational variables that individually have an insignificant impact on the overall uncertainty, but collectively should be accounted for in the assessment.

The uncertainty components tabulated above represent percent uncertainty at the 1σ level. In the tabulation, the net uncertainty of 11% from the analytical sensitivity studies has been broken down into its individual components. When the four uncertainty values listed above (3%, 3%, 11%, and 5%) are combined in quadrature, the resultant overall 1σ calculational uncertainty is estimated to be 13%.

To date the methodology described in Section 2.2.1 coupled with the BUGLE-96 cross-section library has been used in the evaluation of dosimetry sets from 82 surveillance capsules from 23 pressurized water reactors. These capsule withdrawals included 2-5 capsules from individual reactors. The comparisons of the plant specific calculations with the results of the capsule dosimetry are used to further validate the calculational methodology within the context of a 1σ calculational uncertainty of 13%.

This 82 capsule data base includes all surveillance capsule dosimetry sets analyzed by Westinghouse using the Bugle-96 cross-section library and the synthesis approach described in Section 2.2.1. No surveillance capsule dosimetry sets were excluded from the M/C data base. As additional capsules are analyzed using the synthesis approach with the BUGLE-96 cross-section library the M/C comparisons will be added to the database.

The comparisons between the plant specific calculations and the data base measurements are provided on two levels. In the first instance, measurement to calculation (M/C) ratios for each fast neutron sensor reaction rate from the surveillance capsule irradiations are listed. This tabulation provides a direct comparison, on an absolute basis, of measurement and calculation. The results of this comparison for the surveillance capsule data base are as follows:

REACTION	M/C	STD DEV
$^{63}Cu(n\alpha)^{60}Co$	1.09	7.9%
$^{54}Fe(n,p)^{54}Mn$	0.99	8.4%
$^{58}Ni(n,p)^{58}Co$	0.99	8.9%
$^{238}U(n,f)^{137}Cs$	1.01	11.8%
$^{237}Np(n,f)^{137}Cs$	1.06	11.3%
Linear Average	1.03	9.8%

These comparisons show that the calculations and measurements for the surveillance capsule data base fall well within the 13% calculational uncertainty for all of the fast neutron reactions.

The second comparison of calculations with the data base is based on the least squares adjustment of the individual surveillance capsule data sets. The least squares adjustment procedure provides a weighting of the individual sensor measurements based on spectral coverage and allows a comparison of the neutron flux (E > 1.0 MeV) before and after adjustment. The neutron flux/fluence (E > 1.0 MeV) is the primary parameter of interest in the overall pressure vessel exposure evaluations.

The least squares evaluations of the 82 surveillance capsule dosimetry sets followed the guidance provided in Section 1.4.2 of Regulatory Guide 1.190 and in ASTM Standard E944-96, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

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

For the data base comparisons, the calculated neutron spectra were obtained from the results of plant specific neutron transport calculations applicable to each of the 82 surveillance capsules. The sensor reaction rates and dosimetry cross-sections were the same as those used in the direct M/C comparisons noted above.

The results of this latter comparison expressed in terms of the ratio of adjusted flux to calculated flux (A/C) are summarized as follows for the 82 capsule data base:

PARAMETER	A/C	STD DEV
$\phi(E > 1.0 \text{ MeV})$	1.00	7.3%

As with the comparisons based on the linear average of reaction rate M/C ratios, the comparisons of the least squares adjusted results with the plant specific transport calculations demonstrate that the calculated results are essentially unbiased with an uncertainty well within the 20% criterion established in Regulatory Guide 1.190.

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[8]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[9] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness guidelines in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," Revision 2.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," Generic Letter 88-16, October, 1988.

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

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix G, Fracture Toughness Requirements.

5. 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.

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

7. RSICC Computer Code Collection CCC-650, "DOORS 3.1, One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

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

12. I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Benchmark," NUREG/CR-6454 (ORNL/TM-13205), July 1997.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix A, Analysis of Flaws, Article A3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section X1, Division 1, "Low Temperature Overpressure Protection," Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

19. ASME Boiler and Pressure Vessel Code Case N640, Section XI, Division 1, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," February 26, 1996.

20. ASME Boiler and Pressure Vessel Code Case N588, Section XI, Division 1, "Alternative to Reference Flow Orientation of Appendix G for Circumferential Welds in Reactor Vessels," December 12, 1997.

21. ASME Boiler and Pressure Vessel Code Case N641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000.

22. I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 18, 2003

Mr. Gordon Bischoff, Project Manager
Westinghouse Owners Group
Westinghouse Electric Company
Mail Stop ECE 5-16
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN CURVES" (TAC NO. MB5754)

Dear Mr. Bischoff:

By letter dated May 23, 2002, the Westinghouse Owners Group submitted for staff review Topical Report WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves." The staff has completed its preliminary review of WCAP-14040, Revision 3, and has identified a number of items for which additional information is needed to continue its review. This was discussed in a telephone conversation with Mr. Ken Vavrek of your staff on June 5, 2003, and it was agreed that a response would be provided within 30 days of receipt of this letter.

If you have any questions, please call me at (301) 415-1436.

Sincerely,

Drew Holland, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Request for Additional Information

cc w/encl: See next page

RECEIVED
JUN 24 2003
H. A. SEPP, JR.

Westinghouse Owners Group Project No. 694

cc:
Mr. H. A. Sepp, Manager
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Westinghouse Electric Company
P.O. Box 355
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REQUEST FOR ADDITIONAL INFORMATION

WCAP-14040, REVISION 3, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN CURVES"

WESTINGHOUSE OWNERS GROUP

PROJECT NO. 694

Please address the following NRC staff issues pertaining to the review of this topical report.

1. Section 2.3, page 2-5, Branch Technical Position MTEB 5-2 does not give fracture toughness "requirements." Revise WCAP-14040, Revision 3, to refer to the information in MTEB 5-2 as "guidelines" rather than "requirements."

2. Section 2.4, page 2-6, when referring to the "A_i" term in Equation 2.4-3, revise your definition which refers to it as the "measured value of ΔRT_{NDT}" – instead call it the "measured shift in the Charpy V-notch 30 ft-lb energy level between the unirradiated condition and the irradiated condition, f_i."

3. Section 2.4, page 2-7, revise the sentence which reads, "[i]f the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_{\Delta}$), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology," to state "[i]f the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_{\Delta}$), a supplement to the PTLR methodology must be provided for NRC staff review and approval to demonstrate how the results affect the approved methodology."

4. Section 2.5, page 2-7, it is stated that K_{Ia} is the reference fracture toughness curve in Appendix G to Section XI of the ASME Code. Clarify this to note that this refers to Editions of the Code through the 1995 Edition/1996 Addenda. The most recent Edition and Addenda of the Code (1998 Edition through 2000 Addenda) incorporated by reference into 10 CFR 50.55a, however, uses K_{Ic} as the reference fracture toughness curve.

5. Section 2.5, page 2-8, the "note" regarding the use of a 1.223 vs. 1.233 coefficient in the K_{Ia} equation is meaningless and confusing unless one also explains that there was a typographical error in the 1989 Edition of Section XI, Appendix G (i.e., where the 1.233 was used). Revise WCAP-14040, Revision 3, to either eliminate this note or revise the note to offer additional explanation regarding the historical basis for the 1.223 vs. 1.233 issue.

6. Section 2.5, page 2-8, when discussing ASME Code Case N-640, it is not correct to say that an exemption is required to implement N-640 because the NRC has not "endorsed" the Code Case. "Endorsement" implies that it has been included in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability -- ASME Section XI, Division 1." Code Case N-640 would have to be included in the edition of the ASME Code which the licensee has adopted in their facility's licensing basis in order to comply with 10 CFR 50.55a before an exemption is no longer required.

- 2 -

7. The statement in Section 2.5, page 2-10, regarding need for an exemption relative to modifying existing 10 CFR Part 50, Appendix G flange requirements should, for consistency be repeated in Section 2.8.

8. Section 2.6.1, page 2-12, it is stated "[t]hese stress components are used for determining the thermal stress intensity factors, K_{It}, as described in the following subsection." The following subsection is 2.6.2, "Steady-State Analyses," and it does not address the calculation of K_{It}. Revise WCAP-14040, Revision 3, to address this apparent inconsistency.

9. Section 2.6.2, page 2-14, and Section 2.6.5, page 2-15, M_m factors of 1.84, 0.918, and 3.18 are given for various reactor pressure vessel wall thickness ranges to be used when steady-state analyses are performed. It is unclear as to where these M_m factors come from (unable to locate them in any edition of ASME Section XI, Appendix G). Further, they are not consistent with what should be the same M_m factors cited on page 2-15. Revise WCAP-14040, Revision 3, to address this apparent inconsistency in the cited M_m factors.

10. Section 2.7, page 2-19, it should be noted that an exemption is required when a licensee wishes to make use of ASME Code Case N-588. Revise WCAP-14040, Revision 3, accordingly.

OG-02-018
May 23, 2002

WCAP-14040, Rev. 3
Project Number 694

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

Attention: Chief, Information Management Branch,
Division of Inspection and Support Programs

Subject: Westinghouse Owners Group
Transmittal of WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (MUHP-3073)

Reference: 1) Westinghouse Owners Group Letter, R. Bryan to Document Control Desk, "Transmittal of WCAP-15315, Rev. 1, 'Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants'," OG-02-019, May 23, 2002.

This letter transmits five copies of the WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC review and approval. WCAP-14040-A, Rev. 2, was approved by the NRC on October 16, 1995, and contains a methodology for developing Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves and Cold Overpressure Mitigation System (COMS) setpoints and enable temperature that can be referenced by licensees in the Administrative Controls Section of the Technical Specifications when relocating P-T limit curves, COMS setpoints and COMS enable temperature to a Pressure and Temperature Limits Report (PTLR).

Several ASME Nuclear Code Cases (N-588, N-640, and N-641) associated with the development of P-T limit curves and the COMS enable temperature have been approved by the ASME subsequent to the approval of WCAP-14040-NP-A, Rev. 2 in October 1995. Exemption requests have been approved by the NRC to allow the use of these ASME Nuclear Code Cases in the development of P-T limit curves.

WCAP-14040, Rev. 3 has been revised to incorporate these approved ASME Nuclear Code Cases into the methodology used to develop the P-T limit curves and COMS enable temperature that is contained in WCAP-NP-A, Rev. 2.

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WCAP-14040, Rev. 3 also contains an option to develop the P-T limit curves without the flange requirement, currently required by 10CFR50 Appendix G. The option to develop P-T limit curves without the flange requirement would require NRC approval of an exemption request, or rulemaking to eliminate the requirement. A Petition for Rulemaking to eliminate the flange requirement of 10CFR50 Appendix G from the P-T limit curves was submitted by Westinghouse Electric Co. in November 1999.

The technical justification for eliminating the flange requirement is contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," Rev. 0. WCAP-15315, Rev. 0 was submitted to the NRC with the Petition for Rulemaking to eliminate the flange requirement of 10CFR50 Appendix G by Westinghouse Electric Co., in November 1999. WCAP-15315, Rev. 1 contains the additional information for eliminating the flange requirement as requested by the NRC during a meeting between Westinghouse and the NRC on August 28, 2001. WCAP-15315, Rev. 1 is also being submitted for NRC review as justification for eliminating the flange requirement of 10CFR50 Appendix G (Reference 1).

The WOG is submitting WCAP-14040, Rev. 3 under the NRC licensing topical report program for review and acceptance for referencing in licensing actions. The objective is that once approved, each WOG member can reference a single methodology in the Administrative Controls Section of the Technical Specifications when relocating or revising P-T limit curves and COMS setpoints and enable temperature in a PTLR.

The WOG requests that the NRC complete the review of WCAP-14040, Rev. 3, by September 30, 2002. Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, "Processing Request for Reviews of Topical Reports," the WOG requests that the NRC provide an estimate of the review hours, and target dates for any Request(s) for Additional Information and for completion of the Safety Evaluation for WCAP-14040, Rev. 3.

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If you require further information, please contact Mr. Ken Vavrek in the Westinghouse Owners Group Project Office at 412-374-4302.

Very truly yours,

Robert H Bryan

Robert H. Bryan, Chairman
Westinghouse Owners Group

enclosures

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