Mr. Gordon Bischoff, Manager   February 2, 2004
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,

/RA/
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

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Westinghouse Owners Group Project No. 694

cc:

Mr. John S. Galembush, Acting Manager
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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.

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:

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
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Date: February 2, 2004