August 20, 2007

Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) WCAP-16608-P, REVISION 0, "WESTINGHOUSE CONTAINMENT ANALYSIS METHODOLOGY" (TAC NO. MD2953)

Dear Mr. Gresham:

By letter dated August 15, 2006 (Agencywide Documents Access and Management System Accession No. ML062360486), Westinghouse submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR WCAP-16608-P, Revision 0, "Westinghouse Containment Analysis Methodology." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On August 3, Michael Riggs, Principal Fuel Licensing Engineer, and I agreed that the NRC staff will receive your response to the enclosed Request for Additional Information (RAI) questions by August 31, 2007.

Enclosure 1 provides a non-proprietary version of the RAI questions. Enclosure 2 provides a proprietary version of the RAI questions. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1119.

Sincerely,

/RA/

Jon H. Thompson, Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures: 1. Non-Proprietary RAI Questions
2. Proprietary RAI Questions

cc w/o encl 2:
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Letter to James A. Gresham from Jon H. Thompson dated: August 20, 2007

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) WCAP-16608-P, REVISION 0, "WESTINGHOUSE CONTAINMENT ANALYSIS METHODOLOGY

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-16608-P, REVISION 0, "WESTINGHOUSE

CONTAINMENT ANALYSIS METHODOLOGY"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

All section, table, paragraph, figure, and reference numbers refer to Topical Report (TR) WCAP-16608-P, Revision 0, unless specified otherwise.

1. Section 4.1 of TR WCAP-16608-P, Revision 0, states that although the containment models and methods described in this report were developed using GOTHIC 7.2a, Westinghouse intends to use future versions of GOTHIC for plant specific containment analyses as they become available. This, as stated, is not acceptable.

   (a) A new version of GOTHIC may contain models or yield results that would not be acceptable to the U.S. Nuclear Regulatory Commission (NRC) staff, at least without more justification than might be provided in the Numerical Applications, Inc. (NAI) documentation. On the other hand, the NRC staff would prefer not to perform a complete review of the application of a new version of GOTHIC. Therefore, please propose a procedure that will allow Westinghouse flexibility, but that still meets the NRC staff's responsibility to review significant changes from one version of GOTHIC to another.

   (b) For the calculation of containment backpressure for the Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 loss-of-coolant accident (LOCA) analyses, the version of GOTHIC proposed in this TR (7.2a) must be considered part of the LOCA evaluation model. Therefore, please describe the Westinghouse proposal for including GOTHIC 7.2a in the Westinghouse boiling water reactor (BWR) emergency core cooling system (ECCS) evaluation model.

   (c) Describe how the Westinghouse use of GOTHIC 7.2a will comply with 10 CFR Part 50, Appendix B, and 10 CFR Part 21.

   (d) Please describe the Westinghouse procedure for responding to Electric Power Research Institute notifications of discovered errors in GOTHIC 7.2a so that licensing calculations done by Westinghouse with GOTHIC 7.2a remain valid.

2. Introduction of TR WCAP-16608-P, Revision 0, lists 9 applications of the methodology for both BWRs and pressurized water reactors (PWRs). However, only BWR Mark I containment applications are described in detail in the report in Appendices A and B. Therefore, the NRC staff safety evaluation (SE) will only discuss application to BWR Mark I containments.

3. What steps are taken to verify that all relevant GOTHIC 7.2a empirical correlations will be used within their range of applicability?

4. What characteristic length is used in the heat transfer correlations applied to BWR Mark I containment heat transfer?

5. Subcompartment analysis Standard Review Plan (SRP) Section 6.2.1.2 is not listed as one of the applications of this methodology. Please verify that the methods of TR WCAP-16608-P, Revision 0 (i.e., GOTHIC) will not be applied to BWR Mark I subcompartment analysis.

6. For the applications considered in TR WCAP-16608-P, Revision 0, is surface-to-surface radiation heat transfer considered? If so, please describe and provide the source of the emissivities and shape factors used. For which, if any, of the BWR Mark I containment safety analyses is thermal radiation heat transfer significant?

7. Please verify that the GOTHIC height scaling factor is not included in the calculation of heat transfer to structures.

8. Discuss how suppression pool level and/or volume changes during postulated events are calculated. If level and/or volume change is not included in the calculation, why is this acceptable, especially for the available net positive suction head (NPSH) calculation?

9. Do any of the calculations of TR WCAP-16608-P, Revision 0, use the GOTHIC jet and drop breakup model? If yes, which calculations. If yes, please reference qualification studies or other references to demonstrate that the model is conservative.

10. SRP 6.2.1.1.A.II.5.b states that to satisfy the requirements of General Design Criterion-38, the containment pressure should be reduced to less than 50 percent of the peak calculated pressure for the design-basis LOCA within 24 hours after the postulated accident. Demonstrating that the containment pressure is reduced by 50 percent in 24 hours permits the containment leakage rate to be reduced by 50 percent after 24 hours. This guidance was intended for PWRs. However, the NRC staff has approved the reduction in leakage with a reduction in pressure for BWRs also. Does Westinghouse intend to perform this type of calculation for BWR Mark I containments? If so, please provide information on how the calculation would be done.

11. Section 2. Table 2-1 Item 1. What assurance is there that the **[**
] if analyzed with GOTHIC?

12. Section 4.2.4 Page 4-7 Item 4. Licensing calculations maximizing BWR suppression pool temperature typically assume the torus outside surface is adiabatic, i.e., no heat transfer from the torus outer surface to the reactor building atmosphere. The NRC staff requests that, for added conservatism, you reconsider including this heat transfer.

13. TR WCAP-16608-P, Revision 0, Table A.3-1. This table gives values of **[ ]** for Model 1, Model 2, and the Generic Model. There is a large discrepancy between the values for Model 1 and Model 2 and the Generic Model. The large values for Model 1 and Model 2, according to the description on Page A-40, were chosen to match the benchmark data. The value **[ ]** for the Generic Model appears to be more consistent with values used in other BWR modeling.

    Figures A.3-3 and A.3-8 for Models 1 and 2, respectively, give peak drywell pressures of approximately 46 pounds per square inch gauge (psig) and 42 psig, respectively, while the peak drywell pressure for the Generic Model is approximately 44 pounds per square inch absolute or 29.3 psig - considerably less than either Model 1 or Model 2.

    It therefore appears that when attempting to reproduce other BWR calculations, a **[ ]** is necessary. When using a **[ ]**, GOTHIC calculates a pressure much less than the other cases.

    One explanation for this could be the modeling of droplets in the drywell and the downcomer flow. If a large fraction of the droplets are removed in the drywell, this would account for the need to **[ ]** in Models 1 and 2 and also account for the low drywell pressure calculated by the Generic Model. This might also explain the lack of agreement with the wetwell pressure in Figure A.3-4.

    It has been the NRC position that 100 percent entrainment should be assumed **[ ]**. This is consistent with comparisons with Humboldt Bay and Bodega Bay data (Reference 2).

    Please verify that GOTHIC is not removing droplets from the drywell atmosphere prior to flowing through the vents.

    If removal of droplets is not the cause of the discrepancy between the **[ ]** Models 1 and 2 and the Generic Model, please explain why **[ ]** are necessary. Please also assess the impact of droplet behavior on vent flow for the small and intermediate break accidents.

14. American Nuclear Society (ANS) 56.4-1983 guidelines state that flashing should be assumed at the transient atmosphere steam partial pressure. GOTHIC calculates flashing based on the atmosphere total pressure. Since the partial pressure results in a higher steam fraction, shouldn't the recommendations of the ANS standard be used for

peak pressure and the GOTHIC approach used for conservative suppression pool temperature?

15. Please provide the information in the table below to reflect the Westinghouse approach to BWR containment analyses. The first row is completed as an example.

BWR Accident Parameter	Event Interval Analyzed (secs)	Short-Term Mass and Energy Release Computer Code	Long-Term Mass and Energy Release Computer Code	Decay Heat Model and Uncertainty	Assumed Single Active Failure	Passive heat sinks included	Drywell and Wetwell Spray Credited
Double-ended recirculation suction line break (RSLB) Peak Pressure	[ ]	[ ]	-	[ ]	[ ]	?	[ ]
Hydrodynamic loads criteria Double-ended RSLB, Intermediate Break Accident (IBA)/ and Small Break Accident (SBA) LOCAs							
RSLB Minimum ECCS Back Pressure (All breaks considered to demonstrate compliance with 10 CFR 50.46)							
Minimum NPSH available							
RSLB Long Term Pressure and Temperature							
Main Steam Line Break (MSLB) Peak Pressure							
MSLB Long-Term Pressure and Temperature							
Anticipated Transient Without Scram (ATWS)							
SBA							

16. TR WCAP-16608-P, Revision 0, Section 4.2.1.1, states that the DEFAULT option for revaporization will be used. Justify not limiting the revaporization fraction to 8 percent, according to the guidelines of NUREG-0588. If possible to determine, what revaporization fractions are typical for BWR Mark I calculations?

17. TR WCAP-16608-P, Revision 0, Section 4.2.1.2. For which calculations discussed in TR WCAP-16608-P, Revision 0, is the mist model used? For these cases, provide a sensitivity to show the calculated effect of the mist model on BWR containment peak pressure and peak temperature. What is the basis for the 200 micron drop size? How sensitive are BWR conditions to the default assumption of a 200 micron mist droplet size.

18. Section 4.2.2 states that: **[ ]**

    This contradicts the approach of a BWR licensee who assumes a **[ ]** Please perform a sensitivity calculation to verify that **[ ]** is the conservative approach for peak containment pressure. Should this be determined on a plant specific basis?

19. Appendix A Section A.4.3:

    (a) Describe how feedwater is modeled in the BWR applications described in TR WCAP-16608-P, Revision 0.

    (b) What assurance is there that the feedwater model added to GOBLIN models the feedwater system correctly and conservatively?

20. Appendix A Section A.4.3. In considering NPSH available for the ECCS pumps during BWR Mark I postulated events, it is also important to consider events other than the LOCA, including a stuck open relief valve, and non-design basis events such as Appendix R fire, anticipated transient without scram, and station blackout (i.e., ability to cool the suppression pool following restoration of AC power). These events may either be discussed in this TR review or the NRC staff will request information on the calculation of these events for each plant-specific application which includes consideration of ECCS pump NPSH. Such information as that listed below should be discussed and all licensing analyses referencing TR WCAP-16608-P, Revision 0, should then be consistent with these descriptions:
    - assumed initial power level
    - assumed initial pressures, temperatures and relative humidities in the drywell and wetwell
    - decay heat model (with or without 2σ uncertainty)
    - pump configuration and flow rates assumed
    - credit for non-safety systems
    - use of drywell and wetwell sprays
    - suppression pool level
    - initial pressures, temperatures and relative humidity values for drywell and wetwell
    - passive heat sinks

21. The limiting NPSH margin for a Mark I BWR may occur during the short-term period following the initiation of a LOCA (that is, the first 10 minutes prior to operator action) or the long-term period (after 10 minutes when the operator can reduce the residual heat removal and core spray pump flow rates and activate sprays). TR WCAP-16608-P,

Revision 0, did not separately address the analysis of available NPSH for the short-term period. Are there any differences between the containment analyses for available NPSH between the short term and long term periods in terms of assumptions and modeling?

22. Appendix A Section A.4.3. Although a loss of one train of emergency alternate current power (following an assumed loss-of-offsite power) was assumed as the worst single failure for the example, this may not always be the worst single failure. The worst single failure should be assessed for each plant to which this methodology is applied.

23. Section A.4.3 (Item 6). Explain why **[** **]** is conservative for available NPSH calculations.

24. Section A.4.3. Containment leakage should be included in available NPSH calculations which credit containment accident pressure.

25. Section A.4.3 (Item 8). It is the NRC staff's position to request that available NPSH calculations be carried out until credit for containment accident pressure is no longer needed. This time period could be greater than 50,000 seconds.

26. Section A.4.3 Figure A.4.3-1. Explain the initial decrease in drywell pressure during blowdown.

27. **[** **]** Justify their use for the diffusion layer model.

28. Verify that for minimum backpressure and available NPSH calculations, non-safety equipment which cools the containment will be included in the calculations.

29. Appendix A Figure A.4.4-1. At times greater than 2X10^3 seconds, the wetwell pressure appears to be slightly greater than the drywell pressure. Please explain.

30. Appendix A Section A.4.1. In Table 2.9-1, please provide the bias assumed in each analysis (peak drywell pressure, ECCS minimum backpressure, etc.) for wetwell humidity. Why is this conservative? How important is the wetwell humidity for each analysis?

31. Section A.4.2 states that the **[** **]** Is the same method used for the available NPSH calculations?

32. Section A.4.8. How does GOTHIC calculate core power after BISON? Is it just decay heat?

33. Please provide Reference A-10.

34. For the cases described in Table B-3, please provide the calculated containment parameters of drywell and wetwell pressure and temperature and suppression pool temperature, if available.

35. Section B.3 lists several assumptions made to minimize the mass and energy release for the minimum containment pressure calculations.

    (a) Please indicate the significance of these assumptions on containment pressure.

    (b) Why couldn't other assumptions be included such as critical flow correlation less conservative than Moody, assuming 100 percent power rather than 102 percent power, a nominal decay heat, etc.

36. Westinghouse states (Table B-1 of Appendix B) that the licensed core power plus uncertainty is used as input to the BWR mass and energy release model. Please verify that this will be a 2 percent uncertainty as specified by NRC Regulatory Guide 1.49 unless the licensee has justified a smaller uncertainty based on more accurate feedwater flow measurements.

<u>REFERENCES</u>

1. US NRC, "CONTAIN Code Qualification Report/User Guide for Auditing Design Basis BWR Calculations," SMSAB-03-02, Table 2.1, March 2003 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML030700335)

2. The General Electric Pressure Suppression Containment Analytical model, NEDO-10320 Supplement 1, Section 3.2, General Electric Company, May 1971.

3. W.D. Crouch, Tennessee Valley Authority (TVA), letter to US NRC, Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Technical Specifications (TS) Changes TS-431 and TS-418 - Extended Power Uprate (EPU) - Response to Round 6 Request for Additional information, RAI ACVB 38/36, July 21, 2006 (ADAMS Accession No. ML062090071).