FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-16865-P/WCAP-16865-NP, REVISION 1, "WESTINGHOUSE BWR ECCS EVALUATION MODEL UPDATES: SUPPLEMENT 4 TO CODE DESCRIPTION, QUALIFICATION AND APPLICATION"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 1, 2009, Westinghouse submitted Topical Report (TR) WCAP-16865-P/WCAP-16865-NP, Revision 1, "Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application" (Reference 1) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The purpose of this report was to describe the updates made to the BWR Loss-of-Coolant Accident (LOCA) ECCS Evaluation Model. The Evaluation Model (Reference 2) was updated to include a change to the manner in which the end of lower plenum flashing is defined. The previously approved determination of the end of lower plenum flashing was not based on the physical process of flashing in the lower plenum, but on other related phenomena. Thus the convective cooling from lower plenum flashing that was passed as a boundary condition to the downstream heat-up code was ended prematurely. The new determination of the end of lower plenum flashing is based on the completion of flashing in the lower plenum and as such, credit is taken for the convective cooling due to flashing in the lower plenum as long as the lower plenum continues to flash or the core spray flow reaches rated conditions.

In response to the NRC staff's Request for Additional Information (RAI), dated June 30, 2010 (Reference 3), Westinghouse clarified the TR by LTR-NRC-10-47, dated August 11, 2010 (Reference 4).

2.0 REGULATORY EVALUATION

Licensees must evaluate the consequences of various transients and accidents that could occur at their nuclear power plant. A transient, or anticipated operational occurrence (AOO), is an event which is expected to occur one or more times during the life of the nuclear power plant.

Examples of transients include tripping of the turbine generator, isolation of the main condenser, and loss of offsite power. An accident is an event which is not expected to occur during the life of the nuclear power plant, but must still be examined because of its potential to release significant amounts of radiation to the public. Examples of accidents include a major pipe rupture on the primary loop, a major pipe rupture on the secondary loop, and an ejection of a control rod assembly.

Licensees use a variety of methods to evaluate the transients and accidents that could occur at their nuclear power plant. The NRC staff reviews the methods to ensure that they provide realistic or conservative results and that they adhere to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR). To assure the quality and uniformity of NRC staff reviews, the NRC created the Standard Review Plan (SRP) to guide the NRC staff in performing the reviews. Chapter 15.0 of the SRP focuses on transient and accident analysis (Reference 5) and Section 15.0.2 specifically focuses on the review of transient and accident analysis methods (Reference 6). Similar guidance for transient and accident analysis methods is also set forth for the industry in Regulatory Guide 1.203 (Reference 7).

Regulations which are applicable to transient and accident analysis methods are found in 10 CFR 50.34, 10 CFR 50.46, and 10 CFR Part 50, Appendix K. Additionally, because the results of the transient and accident analysis methods are important to the safety of nuclear power plants, the methods must be kept under a quality assurance program which meets the criteria set forth in 10 CFR Part 50, Appendix B.

WCAP-16865-P/WCAP-16865-NP, Revision 1, requests a change to the manner in which the end of lower plenum flashing is defined. The NRC staff's review was based on an evaluation of this change, compliance with the applicable regulations, and guidance from SRP Section 15.0.2. The most applicable regulation dealing with this change is Appendix K.D.6 to 10 CFR Part 50:

> Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling. Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7x7 fuel assembly array, the following convective coefficients are acceptable:
>
> a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.
>
> b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 Btu-hr^{-1}-ft^{-2} $^{o}F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.
>
> c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu-hr^{-1}-ft^{-2} $^{o}F^{-1}$ shall be applied to all fuel rods.

3.0 TECHNICAL EVALUATION

3.1 Background Information

The design basis accident of a BWR is a double ended guillotine break of the largest pipe in the plant, which is a recirculation pipe. Following this accident, the reactor coolant system (RCS) will lose most of its coolant inventory before any ECCS cooling flow is available. Once ECCS flow is available, the reactor will slowly refill terminating the temperature rise in the core. Computer codes, such as GOBLIN, are used to calculate the heat being transferred from the fuel to the coolant during such an accident. GOBLIN calculates the system transient response to a LOCA and other figures of merit that are used as boundary conditions in the heat-up calculation where the results are compared to regulatory limits to ensure the safety of the public if such an unlikely event were to occur.

3.1.1 BWR LOCA

A postulated LOCA is initiated by a break in the RCS. Initially liquid water will flow out of the break. While the RCS is losing its liquid inventory and if only liquid water flows out of the break, the pressure will increase as the steam being generated in the core due to boiling has no escape path. For a large-break LOCA, the liquid level will decrease and uncover the break location allowing steam to escape. For small-break LOCAs, the automatic depressurization system (ADS) may be used to allow the steam to escape. Once the steam starts to exit the core, the core pressure starts to decrease.

As the core pressure decreases, both the saturation pressure and saturation temperature also decrease. The decrease in the saturation temperature will increase steam generation by increasing the rate of boiling and cause some flashing to occur. Flashing occurs when the temperature of the liquid is higher than the saturation temperature of that liquid at the current pressure. Some of the liquid will flash to steam to reduce the temperature of the liquid.

The boiling and flashing in the core will continue as the pressure decreases. Eventually, all of the water in the core will become steam leaving no liquid remaining to cool the core. However, the large volume below the core, the lower plenum, contains a large amount of liquid water at very high temperatures. As the RCS depressurizes, this water will be above its saturation temperature and therefore a portion of the water in the lower plenum will flash to steam. Some of the steam generated will flow up the jet pump and out the break, and some of the steam will flow into the core providing convective cooling to the core.

Once the pressure in the core has decreased such that the ECCS pumps can pump against the core pressure, the ECCS cooling water will flow into the core. When core spray pumps' injection valves open, subcooled water will spray into the upper plenum of the core thereby condensing steam. The condensing of the steam further decreases the pressure which allows more ECCS water into the core. The ECCS flow eventually fills the core from the bottom up and cools the fuel rods as it rises.

3.1.2 Lower Plenum Flashing

Before any ECCS flow starts to inject into the core, but after the liquid water in the core has become steam, the only means for convective cooling of the core is the steam generated in the lower plenum due to flashing. As the RCS pressure decreases, the saturation pressure and consequently the saturation temperature of the water in the lower plenum decreases. The water temperature will be above the saturation temperature, and therefore some of the water flashes to steam. The steam generated can be quite substantial and thus the cooling provided by the steam can dramatically reduce the fuel temperatures. Therefore it is important to ensure that this phenomenon is conservatively modeled.

The steam generated by lower plenum flashing is calculated directly from the water temperature and the current saturation temperature. Once the water in the lower plenum is above the saturation temperature for the current pressure, enough steam is flashed to reduce the water temperature to the saturation temperature. Because the flashing model is tied to the saturation temperature, it is tied to the RCS pressure, and, therefore, tied to the depressurization.

The split between the steam flowing into the core and the steam flowing up the jet pump and out the break is based on the pressure drops between the two paths and is modeled by the code. It is important to ensure that these paths are modeled correctly, including the split between the steam flow into the hot channel and the steam flow into the average channels.

The heat transfer which occurs in the core due to the steam cooling is single phase heat transfer and is well understood. A combination of the Dittus-Boelter, Seider-Tate, and Jakob correlations are used, depending on the type of flow (turbulent forced convection, laminar forced convection, laminar natural convection).

Finally, if the water in the lower plenum stops flashing, there is no longer a steam generation source and the flashing in the lower plenum ends. With no steam cooling, the core experiences an almost adiabatic heat up until ECCS flow reaches the core. This heat up can cause dramatically higher PCT, therefore it is important to conservatively model the end of lower plenum flashing.

3.2 Purpose of WCAP-16865-P/WCAP-16865-NP, Revision 1, “Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application”

In this TR, Westinghouse proposes to change how the end of lower plenum flashing is determined. This change credits lower plenum flashing for a longer time in the current evaluation model. The additional crediting of lower plenum flashing would provide increased steam cooling to the fuel and reduce the PCT.

3.2.1 Westinghouse’s Current Determination of the End of Lower Plenum Flashing

Westinghouse’s currently approved methodology (Reference 2) determines the end of lower plenum flashing by assuming flashing ends at:

> …the time whenever the heat transfer coefficient based on the GOBLIN/DRAGON heat transfer package described in Sec 3.5, shows a marked degradation in heat transfer as a consequence of high voiding and reduced flow rates at the axial plane of interest.

This determination is conservative, in that the credit for lower plenum flashing ends prematurely and the calculated PCT is higher than would be expected. To gain back some margin, Westinghouse submitted this TR to more accurately determine the end of lower plenum flashing.

3.2.2 Westinghouse's Proposed Determination of the End of Lower Plenum Flashing

Westinghouse's proposed methodology (Reference 1) would use code calculated parameters in the lower plenum to determine if the remaining water in the lower plenum is flashing. Lower plenum flashing would end when: (1) depressurization in the lower plenum stops or the water in the lower plenum becomes subcooled, or (2) there is no longer any water in the lower plenum to flash.

3.3 Validation of the Proposed Methodology

Westinghouse performed various analyses to ensure that the change in the methodology in determining the end of lower plenum flashing would still result in a conservative calculation. The GOBLIN code was compared against the Two-Loop Test Apparatus (TLTA) data to further validate the steam cooling models. The GOBLIN code was also compared against The Rig of Safety Assessment (ROSA-III) data to further validate the code's ability to accurately and conservatively calculate the thermal-hydraulic conditions during a LOCA. Additionally two sensitivity studies were performed. The first confirmed that a reduction in core spray remained a conservative assumption. The second compared different break flow models and confirmed that the Moody break flow model remains conservative, and that the flashing model is not overly sensitive to the depressurization rate.

3.3.1 TLTA Test Comparison

The TLTA bundle uncovery test simulated one full-length BWR fuel assembly as well as all the major regions of a BWR/6 system including the lower plenum, core region, upper plenum, steam separator region, annular downcomer regions, steam dome, and two recirculation loops. GOBLIN was compared to a TLTA test which was setup to obtain data for evaluating the thermal-hydraulic conditions in a partially uncovered bundle. This was accomplished by filling the test section to an elevation above the top of the fuel assembly, and allowing the heat from the fuel rod simulator to boil off the water. Thermocouples and pressure transducers in the test section recorded the temperatures and pressures which were compared to a GOBLIN calculation.

The comparison between GOBLIN and the TLTA data indicated that GOBLIN conservatively captures the decreasing two phase level and subsequent fuel heat up above the two phase level. Initially, the heat up has a two phase component, but it then transitions to single phase steam. GOBLIN captures the rates and the peak temperatures seen in the test.

Based on the comparison to the TLTA test, the NRC staff has determined that GOBLIN's steam cooling model remains acceptable for use. The new method for determining the end of lower plenum flashing will place more emphasis on the steam cooling modeling and Westinghouse has provided analysis which demonstrates that the steam cooling model can accurately and conservatively predict steam cooling.

3.3.2 ROSA-III Test Comparison

The ROSA-III test simulates a BWR with all critical core components including two recirculation loops, four jet pumps, a lower plenum, four half height fuel assemblies, core bypass, an upper plenum, and steam separators. Additionally, the facility contains the major subsystems of the pressure vessel, main steam line, feedwater line, coolant recirculation system, and ECCS. The use of four fuel assemblies enabled data collection on the flow split between the hot assembly and surrounding average assemblies. GOBLIN was compared to a ROSA-III test which was setup to obtain data for evaluating the thermal-hydraulic conditions following a postulated large-break LOCA. Thermocouples and pressure transducers in the test section recorded the temperatures and pressures which were compared to a GOBLIN calculation.

The comparison between GOBLIN and the ROSA-III data indicated that GOBLIN conservatively captured all major portions of the event, including the rapid blowdown and subsequent heat up. GOBLIN was also able to capture the parallel channel effect by modeling increased temperatures in the hot assembly which were conservative relative to the test data.

Based on the ROSA-III test, the NRC staff has determined that GOBLIN is able to conservatively determine the steam generated in the lower plenum, the heat transfer from the fuel rods to the steam, and the subsequent heatup of the fuel. The new method for determining the end of lower plenum flashing will increase the amount of time the steam is generated in the lower plenum and likewise lead to lower PCTs due to the additional steam generated. Westinghouse has provided an analysis which demonstrates that the additional steam cooling is modeled conservatively and the analysis will still result in a conservative PCT calculation.

3.3.3 Reduction in Core Spray Sensitivity Study

Any delay in core spray reaching rated flow conditions is assumed to be a conservatism. The time between core spray initiation and core spray rated flow is assumed to be longer in the analysis than in an actual plant. This is to account for uncertainty and add conservatism. However, the proposed methodology for determining the end of lower plenum flashing may result in a delay in core spray no longer being conservative. That is, due to the additional heat transfer occurring as a result of lower plenum flashing, assuming a delay in core spray may be non-conservative. Before core spray reaches rated flow, the heat transfer rate in the hot assembly is actually higher than it is when core spray reaches rated flow. Therefore, it would seem that a delay in core spray allows for higher heat transfer rates for a longer period of time and causes an overall decrease in PCT. To determine if this is the case, Westinghouse performed an analysis where core spray flow rate was slightly reduced such that the core spray was delayed by two seconds.

The comparison between delayed core spray and non-delayed core spray demonstrated that the increased heat transfer obtained from delaying core spray was inconsequential, and that the delay in core spray did have a dramatic impact on the recovery time of the fuel. Thus, the delay in core spray caused a delayed recovery time, which resulted in higher fuel temperatures when compared to the non-delayed case.

Based on this sensitivity study, the NRC staff has determined that a delay in core spray reaching rated flow conditions remains conservative, and, therefore, the assumption remains valid.

3.3.4 Break Flow Sensitivity Study

The break flow model mandated by Appendix K to 10 CFR Part 50 is the Moody model. This model is known to conservatively predict break flow, in that it will predict a higher break flow than is actually expected. The break flow model used to compare GOBLIN to ROSA-III data was the Homogeneous Equilibrium Model (HEM). The HEM model was used because it will predict more accurate break flow than the Moody model, which allows for a better comparison to test data.

Because the Moody model overpredicts the break flow, it would also overpredict the depressurization. A faster depressurization equates to more steam flashing and increases core cooling by the additional steam flashing. This issue was raised by the NRC staff and Westinghouse responded by performing a sensitivity analysis which compared the HEM and Moody break flow models in GOBLIN.

The comparison demonstrated that the faster depressurization of the Moody model resulted in less flashing and less flow into the hot assembly. The faster depressurization early in the transient results in less water to flash later in the transient when steam cooling provides much of the heat transfer. Therefore, by using the Moody model, much of the core inventory is lost before it has changed to steam and cools the core. This results in a conservative PCT calculation.

Additionally, Westinghouse compared the ROSA-III test data to GOBLIN runs with the Moody model, the HEM model, and the HEM model with 110 percent break flow to demonstrate that there was no extreme sensitivity to the depressurization model. All three break flow models resulted in a conservative PCT calculation. The HEM model with 110 percent of break flow was slightly more conservative than the HEM model. The Moody model was more conservative than the HEM model with 110 percent break flow.

Based on this sensitivity study, the NRC staff has determined that the Moody model remains appropriate for performing LOCA analysis with the new method for determining the end of lower plenum flashing. The faster depressurization rate of the Moody model continues to provide conservative PCT calculations and the PCT calculation is not overly sensitive to the depressurization rate.

3.3.5 Validation Summary

The NRC staff has determined that the validations presented by Westinghouse demonstrate that the proposed method to determine the end of lower plenum flashing is acceptable because it will continue to result in a conservative safety analysis. The new method is more physically based than the currently approved method, and results in a more realistic analysis.

4.0 CONCLUSION

Based on the forgoing considerations, the NRC staff concludes that the proposed method to determine the end of lower plenum flashing is acceptable for use in GOBLIN provided the conditions and limitations described in Section 5.0 are met.

5.0 LIMITATIONS AND CONDITIONS

As proposed by Westinghouse **[**

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If the NRC's criteria or regulations change so that its conclusions about the acceptability of the thermal-hydraulic methods or statistical analyses are invalidated, the licensee referencing the report (Reference 1) will be expected to revise and resubmit its respective documentation, or submit justification for the continued effective applicability of these methodologies without revision of the respective documentation.

6.0 REFERENCES

1. Shum, F.D., Blaisdell, J.A., Perez, J.A., Kroll, K.J., “Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application”, WCAP-16865-P, Revision 1, November 2009. (ADAMS Package Accession No. ML093410016)

2. Ebeling-Koning, D.B, et al., “Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification”, WCAP-11284-P-A, Addendum 1, “RPB-90-93-P-A,” October 30, 1989.

3. Letter from U.S. Nuclear Regulatory Commission to Gresham, J. A. (Westinghouse), “REQUEST FOR ADDITIONAL INFORMATION RE: WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT WCAP-16865-P, REVISION 1, “WESTINGHOUSE SUPPLEMENT 4 TO CODE DESCRIPTION, QUALIFICATION AND APPLICATION” (TAC NO. ME2901)” dated June 30, 2010. (ADAMS Package Accession No. ML101730125)

4. Letter from Gresham, J. A. (Westinghouse) to U.S. Nuclear Regulatory Commission, “Response to the NRC’s Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16865-P, Rev. 1, “Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application,” dated August 11, 2010. (ADAMS Package Accession No. ML102310070)

5. Chapter 15.0, “Introduction – Transient and Accident Analyses” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Revision 0, March 2007. (ADAMS Accession No. ML070710376)

6. Section 15.0.2, “Review of Transient and Accident Analysis Methods” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 0, USNRC, December 2005. (ADAMS Accession No. ML053550265)

7. Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” December 2005. (ADAMS Accession No. ML053500170)

Attached: Appendix A – Request for Additional Information Summary

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Appendix A – Request for Additional Information (RAI) Summary

Often, the NRC staff finds it necessary to make RAIs to gain a more complete understanding of the material under review. Included below are staff comments on the reasons for requesting the information as well as the resolutions of each response. The responses to the RAIs referenced below can be found in Reference 4.

RAI 1

Comment: The staff was unclear on how the change in the determination of the end of lower plenum flashing would impact the GOBLIN calculation. Specifically, the staff was unclear as to how the proposed determination would impact the heat transfer coefficients used in the core and how the proposed definition would impact the core spray model.

Resolution: Westinghouse responded by detailing the convective heat transfer models used in GOBLIN. The proposed change would only impact the hot channel heat transfer calculations, as the average channel temperature calculations were calculated on a more 'best estimate' basis to better quantify the boundary conditions of the hot channel. The staff determined that the further explanation by Westinghouse provided the necessary clarification. The NRC staff has concluded that this RAI has been resolved.

RAI 2

Comment: According to Reference 2, the spray cooling heat transfer coefficients prescribed by Appendix K to 10 CFR Part 50 were conservative compared to test data, but not extremely conservative. Because spray cooling starts to inject sometime before it reaches rated flow, the staff expected a gradual transition from the heat transfer value calculated by GOBLIN to the Appendix K to 10 CFR Part 50 prescribed value. However, the analysis demonstrated a dramatic reduction in the heat transfer to the Appendix K to 10 CFR Part 50 values. The staff did not understand this dramatic reduction as the Appendix K to 10 CFR Part 50 values were not extremely conservative.

Resolution: Westinghouse responded by detailing how the Appendix K to 10 CFR Part 50 heat transfer values were obtained. While the values prescribed by Appendix K to 10 CFR Part 50 were close to values obtained from testing, the testing did not include any additional steam cooling due to flashing in the lower plenum and instead were derived from single bundle separate effects tests. The staff determined that the further explanation by Westinghouse provided the necessary clarification to explain the dramatic reduction in heat transfer. The NRC staff has concluded that this RAI has been resolved.

RAI 3

Comment: The staff was concerned that a delay in core spray reaching rated flow would result in a less and not a more limiting condition. If that were the case, many of the assumptions in the analysis (such as time delay between core spray initiation and core spray reaching rated flow) would have to be revisited because an overall assumption made is that a delay in core spray is conservative.

Resolution: Westinghouse responded by performing a sensitivity study. The sensitivity study demonstrated that a delay in core spray reaching rated flow would result in a more conservative peak cladding temperatures (PCT) calculation. Any delay in core spray early on in the transient will delay reflood and the subsequent recovery. The delay in the recovery results in a higher PCT. The staff determined that the further explanation by Westinghouse provided the necessary clarification to demonstrate that a delay in core spray reaching rated flow would result in a more conservative calculation. The NRC staff has concluded that this RAI has been resolved.

RAI 4

Comment: The staff was unclear on the proposed methodology to determine the end of lower plenum flashing and if lower plenum flashing would cease if the water in the lower plenum became subcooled.

Resolution: Westinghouse confirmed that the lower plenum flashing would cease if the water in the lower plenum became subcooled. The staff determined that the further explanation by Westinghouse provided the necessary clarification to the definition of the end of lower plenum flashing. The NRC staff has concluded that this RAI has been resolved.

RAI 5

Comment: The staff was unclear as to the meaning of Side Entry Orifice (SEO) in reference to a BWR fuel.

Resolution: Westinghouse provided further description of the SEO, including reference to its location in a figure. The staff determined that the further explanation by Westinghouse provided the necessary clarification to define the SEO. The NRC staff has concluded that this RAI has been resolved.

RAI 6

Comment: The staff requested Westinghouse add the ROSA data to the lower plenum water temperature from the plot. Previously, the plot only contained the GOBLIN predictions.

Resolution: Westinghouse provided the figure with the ROSA data. The figure shows that GOBLIN predicts a slightly faster depressurization than the ROSA test. The faster depressurization by GOBLIN is one of the causes of the more conservative PCT, as the inventory is lost early in the transient which reduces the convective heat transfer later in the transient during the time of PCT. The staff determined that the figure provided the requested clarification. The NRC staff has concluded that this RAI has been resolved.

RAI 7

Comment: The break flow model used in GOBLIN to compare to the ROSA-III test data was the Homogeneous Equilibrium Model (HEM) model. This model was used because it provides a more realistic break flow and consequently more realistic depressurization. However, Appendix K prescribes the Moody model for conservatism. The Moody model will have higher break flow and consequently a higher depressurization rate than the HEM model. Because the Moody model predicted a higher depressurization rate, the NRC staff was concerned that the increased flashing caused by the increased depressurization rate would lead to the Moody model predicting a lower non-conservative PCT as compared to the HEM model.

Resolution: Westinghouse responded by performing a sensitivity study. The sensitivity study demonstrated that the Moody model predicted conservatively higher fuel temperatures than the HEM model. While the higher depressurization rate did increase the heat transfer early on in the transient, the higher depressurization in the Moody calculation also caused faster initial inventory loss. Therefore, during the time corresponding to fuel heatup, the Moody calculation had less inventory and therefore lower flashing and lower convective heat transfer coefficient resulting in higher calculated fuel temperatures than the HEM model. The staff determined that the further explanation by Westinghouse provided the necessary clarification to demonstrate that the Moody model was still conservative compared to the HEM model. The NRC staff has concluded that this RAI has been resolved.

RAI 8

Comment: The proposed change to the definition of the end of lower plenum flashing places more emphasis on the lower plenum flashing model. The staff was concerned that when GOBLIN was initially reviewed, its sensitivity to the steam cooling was minimized due to definition of the end of lower plenum flashing causing a very limited window where steam cooling was applicable. The proposed change to the definition would result in steam cooling models having a larger role in the heat transfer during blow down and the staff was unsure of the sensitivity of this model.

Resolution: Westinghouse responded by performing a sensitivity study. The sensitivity study compared the Moody model with ROSA-III test data and the HEM model at 100 percent and 110 percent break flow. The calculation models all behaved similarly and each predicted a conservative PCT calculation compared to the ROSA test data. The staff determined that the sensitivity study provided by Westinghouse demonstrated that overall PCT calculation is not overly sensitive to the flashing model and steam cooling models. The close agreement between the Moody, HEM, and 110 percent HEM demonstrate this. The NRC staff has concluded that this RAI has been resolved.

RAI 9

Comment: The NRC staff was concerned that the proposed change to the definition of the end of lower plenum flashing may cause excessively large PCT changes in certain events where those changes were not warranted.

Resolution: Westinghouse responded by performing a sensitivity study. The changes in PCT due to the change in the definition of the end of lower plenum flashing were consistent with break size. The largest impact occurred on a non-limiting break size, where the core was uncovered and the automatic depressurization system (ADS) was activated. Previously, this case assumed an adiabatic heat up due to the flashing model being turned off during a significant portion of the transient. With the proposed change to the definition of the end of lower plenum flashing, the fuel is cooled by the steam generated from the depressurization caused by the ADS. This change caused a significant, but appropriate reduction in PCT. The staff determined that the reduction in PCT for the various break sizes was appropriate. The NRC staff has concluded that this RAI has been resolved.