June 7, 2007

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

SUBJECT: DRAFT SAFETY EVALUATION FOR WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) CENPD-132 SUPPLEMENT 4-P-A, ADDENDUM 1-P, "CALCULATIVE METHODS FOR THE CE NUCLEAR POWER LARGE BREAK LOCA EVALUATION MODEL - IMPROVEMENT TO 1999 LARGE BREAK LOCA EM STEAM COOLING MODEL FOR LESS THAN 1 IN/SEC CORE REFLOOD" (TAC NO. MD2161)

Dear Mr. Gresham:

By letter dated May 11, 2006, Westinghouse submitted CENPD-132, Supplement 4-P-A, Addendum 1-P, Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model - Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. Enclosed for Westinghouse review and comment is a copy of the NRC staff's draft safety evaluation (SE) for the TR.

Pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR), we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the draft SE will be made publicly available, and an additional 10 working days are provided to you to comment on any factual errors or clarity concerns contained in the draft SE. The final SE will be issued after making any necessary changes and will be made publicly available. The NRC staff's disposition of your comments on the draft SE will be discussed in the final SE.

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

If you have any questions, please contact Jon H. Thompson at 301-415-1119.

Sincerely,

/RA/

Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Draft SE

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

J. Gresham

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ADAMS ACCESSION NO.: ML071350187*No major changes to SE input. NRR-043

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) CENPD-132, SUPPLEMENT 4-P-A, ADDENDUM 1-P

"CALCULATIVE METHODS FOR THE CE [COMBUSTION ENGINEERING]

NUCLEAR POWER LARGE BREAK LOCA [LOSS-OF-COOLANT-ACCIDENT]

EVALUATION MODEL [EM] - IMPROVEMENT TO 1999 LARGE BREAK LOCA [LBLOCA]

EM STEAM COOLING MODEL FOR LESS THAN 1 IN/SEC CORE REFLOOD"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

1.0 INTRODUCTION AND BACKGROUND

The Westinghouse Electric Company (Westinghouse) submitted a TR revision, by letter LTR-NRC-06-25, dated May 11, 2006, (Reference 1) titled Submittal of CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," and CENPD-132 Supplement 4-NP-A, Addendum 1-NP "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood" (Proprietary/Non-Proprietary).

This revision documented both a proposed and an optional change to the Westinghouse 1999 LBLOCA EM for CE-designed plants. The change would remove some of the conservatism in the approved steam cooling heat transfer component model found in Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix K, by including spacer grid heat transfer effects for reflood rates less than 1 in/sec. This change was proposed because of the calculated adverse consequences to the emergency core cooling system (ECCS) performance using the 1999 EM for the CE 16x16 Next Generation Fuel (NGF) fuel design. The ECCS calculations were adversely impacted by the increase in the core hydraulic pressure loss, the increase in the core cross-sectional flow area, and the decrease in the fuel rod cladding outside diameter. In particular, the core reflood calculations during a LBLOCA were adversely impacted by the CE 16x16 NGF design changes and the core reflood rates that were used to calculate the reflood heat transfer coefficients for the hot rod decreased. The CE 16x16 NGF design changes were estimated by Westinghouse to have an insignificant impact on the ECCS performance peak cladding temperature. However, the impact on the ECCS performance maximum cladding local oxidation percentage for the hot rod rupture node was estimated by Westinghouse to be large enough to warrant specific consideration.

Westinghouse provided additional information, in response to the NRC staff's request for additional information (RAI), to supplement and clarify portions of the submittal by letter LTR-NRC-07-2, dated January 10, 2007, (Reference 2) letter LTR-NRC-07-7, dated February 1, 2007, (Reference 3) and by letter LTR-NRC-07-18 dated March 28, 2007 (Reference 4).

2.0 REGULATORY EVALUATION

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The functional requirements for the ECCS performance, under all LOCA conditions postulated in the design, must satisfy 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." The ECCS calculated cooling performance is based on an acceptable EM for which there is sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor coolant system during a LOCA or, in this case, an ECCS EM developed in conformance with Appendix K to 10 CFR Part 50.

The three specific ECCS performance criteria important to this safety evaluation are:

1. The calculated maximum fuel element cladding temperature does not exceed 2200 °F.

2. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

The impact of the CE 16x16 NGF design changes on the ECCS performance maximum cladding local oxidation percentage for the hot rod rupture node was estimated to be large enough to warrant specific consideration for item 2, above.

The following additional requirements were considered for an Appendix K based model:

1. 10 CFR Part 50, Appendix K.D.5.b states "during refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer."

2. 10 CFR Part 50, Appendix K.C.4.e states "after CHF [Critical Heat Flux] is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA."

3.0 TECHNICAL EVALUATION

3.1 1999 EM Steam Cooling Model for Core Reflood Rate Less Than 1 in/sec.

- The 1999 EM steam cooling model is a component of the Appendix K ECCS calculational method, and is applied to the hot rod rupture node elevation and above when the core reflood rate is less than 1 in/sec. The 1999 EM NRC-accepted steam cooling model is documented in:

    - CENPD-132P, Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975, Section S III.D.6.b, "Heat Transfer Coefficients for Reflood Rates Less Than 1.0 in./sec,"

    - CENPD-132P, Supplement 2, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," July 1975, Section S III.D.6.a, "Application of FLECHT Data to 16x16 Fuel Bundles,"

    - Combustion Engineering letter LD-81-095, December 1981, Enclosure 1-P-A, "C-E ECCS Evaluation Model, Flow Blockage Analysis," and

    - CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001, Section 2.7, "STRIKIN-II Hot Rod Steam Cooling Heat Transfer."

A number of computer programs are used to evaluate the hot fuel rod response to a LOCA. The COMPERC-II (Reference 5) computer program is used for reflood thermal-hydraulic calculations and the output provides boundary conditions for use in the PARCH (Reference 6) computer program. PARCH calculates the steam cooling heat transfer coefficients through the rupture node blockage and above, including the effect of steam superheating. The PARCH hot rod-to-coolant energy balance for calculating the steam temperature includes heat from cladding oxidation and decay heat. The HCROSS (Reference 7) computer program calculates single phase steam flow diversion from the hot rod rupture node blocked subchannel to the unblocked adjacent subchannels, including flow recovery above the blockage. The STRIKIN-II (Reference 8) computer program calculates the rod-to-rod radiation heat transfer for the hot rod enclosure, which is also used by PARCH to calculate hot rod cladding temperatures needed for the steam cooling analysis. The heat transfer coefficient is limited to the FLECHT (Reference 9) correlation upper bound as specified in the original NRC staff approval of the CE Large Break LOCA evaluation methodology.

A detailed description of the 1999 EM method and computer programs used for a licensing analysis was provided by Westinghouse in Reference 3 as response to the NRC staff's RAI. Justification for the implementation of the optional steam cooling model within the context of the 1999 EM was also included in the response to the NRC staff's RAI.

3.2 CE 16x16 NGF Design

The CE 16x16 NGF design is described in TR WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report," February 2006, (Reference 10) and is the subject of a separate review (TAC No. MD0560). The CE 16x16 NGF design changes include mid-grids and intermediate flow mixing (IFM) grids in the fuel assemblies.

Spacer grids are structural members of the fuel assembly which support the fuel rods at a prescribed rod-to-rod pitch. With the exception of CE 16x16 NGF IFM grids in transition cores, all fuel assemblies have spacer grids at the same elevations across the core. Because the grids are at the same elevations, no flow bypass or flow redistribution occurs. The grid reduces the fuel assembly flow area and the flow is first contracted and accelerated, and then the flow is expanded downstream of each grid layer in the core. As the flow is accelerated within the grid length and then expands downstream, it reestablishes the thermal boundary layer on the fuel rod. This results in an increase in the local heat transfer within, and downstream of, the grid. When the flow is a two-phase dispersed droplet flow, characteristic of Pressurized Water Reactor (PWR) blowdown or reflood, the grid promotes additional heat transfer through droplet breakup, interfacial heat transfer, and dispersed flow convective heat transfer. Because the spacer grid blocks a portion of the fuel assembly flow area, the velocity of the vapor passing through the grid is higher than the velocities in nearby fuel bundles. As a result, the vapor-film relative velocity at the grid is larger, so that a wetted grid below the rupture node elevation has a higher interfacial heat transfer coefficient compared to nearby droplets.

Spacer grids have an important effect on several key phenomena during the reflood period, including droplet breakup, interfacial heat transfer, and dispersed-flow convective heat transfer. For the 1999 EM, these aspects of the reflood heat transfer were considered through the use of the empirically based FLECHT correlation required by 10 CFR Part 50, Appendix K. The FLECHT correlation did not explicitly consider spacer grids, and was based on test measurements taken at mid-span locations, which were away from the direct effects of spacer grids. The FLECHT correlation included the effects of spacer grids, even though the egg-crate grids used in those tests were not like the spacer grids in the CE 16x16 NGF assembly design.

The 1999 EM does not have an accepted spacer grid heat transfer model available for use in licensing calculations. The 1999 EM ECCS calculations for the CE 16x16 NGF assembly design were adversely impacted by the increase in the core hydraulic pressure loss, the increase in the core cross-sectional flow area, and the decrease in the fuel rod cladding outside diameter. The core reflood rates used to calculate the reflood heat transfer coefficients for the hot rod decreased as a result of the grid design.

3.3 Optional 1999 EM Steam Cooling Enhancement for Reflood Rates Less Than 1 in/sec

The 1999 EM assumes that the grids are not powered and have a large surface area-to-volume ratio and will quench before the fuel rods. When the grids quench, an additional liquid surface area is created, which helps core cooling conditions by adding additional steam to the vapor flow stream from the evaporation of the liquid film from the grid surface as a result of thermal radiation from the hotter fuel rod (after grid quench occurs). An optional approach for improving the steam cooling heat transfer model was proposed by Westinghouse to utilize the beneficial aspects, as described in Section 3.2 above, of the CE 16x16 NGF spacer grids (both mid-grid and IFM grids) that were not included in the current model. The model was patterned on an NRC-approved approach used by Westinghouse for Westinghouse plants as documented in TR WCAP- 10484-P-A, "Spacer Grid Heat Transfer Effects During Reflood," dated March 1991, (Reference 11) and in TR WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA [BELOCA] Analysis," dated March 1998 (Reference 12). The limitations and restrictions on the use of this Westinghouse plant model were also applied to the CE plant optional steam cooling model.

The 1999 EM optional steam cooling model for core reflood rates less than 1 in/sec includes the following features and methodology constraints:

The optional steam cooling model considers only the spacer grids above the core two-phase level (both mid-grid and IFM grids).

Below the rupture node and above the core two-phase level, the steam flow rate is non-mechanistically augmented based on the Westinghouse BELOCA spacer grid heat transfer enhancement model.

PARCH steam cooling heat transfer coefficients on the rupture node and above are non-mechanistically augmented based on the Westinghouse BELOCA spacer grid heat transfer enhancement model.

The FLECHT upper-bound heat transfer coefficient, as required by the current NRC licensing constraint, is also applied to the spacer grid model improvement. That is, the result of the grid model enhancement cannot result in the use of a heat transfer coefficient greater than FLECHT.

Unlike the Westinghouse plant model, the CE plant model is not mechanistic and is performed as a separate calculation within the PARCH computer program. The important characteristics of the optional model, along with the physical characteristics of the grids, are the grid rewet temperature and the heat transfer coefficient, used to determine the amount of steam evaporated from the grid surface. The grid rewet temperature establishes the time at which the steam is added to the vapor flow stream and enhances the rod cooling. The heat transfer coefficient defines the amount of steam generated from the grid surface as a result of thermal radiation from the hotter fuel rod (after grid quench occurs).

The physical characteristics of the optional spacer grid heat transfer enhancement model include the maximum flow area reduction or spacer grid blockage fraction, the fuel lattice hydraulic diameter, the height of the spacer grid (used to estimate wetted surface area), and the elevation of the top edge of each spacer grid (relative to the bottom of the core). These parameters are unique to a specific grid type.

3.3.1 Spacer Grid Rewet Temperature

The spacer grid rewet temperature determines when the grid surface will quench and develop a liquid film that can be evaporated by radiation heat transfer from the hotter, unquenched fuel rod. The optional steam cooling model is based on a fixed, predetermined rewet temperature.

Because the rewet temperature is not a physical property, but a simplification of a complicated process, many factors which could be taken into account are not considered in this non-mechanistic approach, and rewet is only characterized as a function of void fraction and surface temperature. However, this approach is consistent with the previously accepted Westinghouse BELOCA model.

Westinghouse had originally proposed the use of a rewet temperature higher than the value accepted by the NRC staff for the Westinghouse BELOCA model. The temperature was based on two reflood tests and the value selected was between the observed rewet temperatures. In response to the NRC staff's concern with the proposed value, Westinghouse committed to using the previously accepted value for licencing analyses with the optional steam cooling model (Reference 4, Clarification Question 2a). The previously accepted rewet temperature is lower than the values observed from the reflood tests examined by Westinghouse and is, therefore, conservative. The NRC staff finds the rewet temperature acceptable for an EM based on 10 CFR Part 50, Appendix K.C.4.e.

The void fraction is not calculated in the approved 1999 EM. The void fraction in the two-phase region is specified as input for a licensing analysis (Ref. 1, response to RAI 4 and 5). This value is consistent with the void fraction range that supports the selected rewet temperature and is also consistent with the rewet temperature for the grid spacer in the Westinghouse BELOCA model.

The optional steam cooling model does not calculate the grid surface liquid film characteristics. The approved 1999 EM requires a fixed entrainment rate of liquid exiting the core mixture level per unit of steam exiting the core mixture level (Reference 2, response to RAI 4 and 5). This rate supports the non-mechanistic steam cooling model which assumes that there is always a liquid film on the grid when the grid temperature is below the rewet temperature. This liquid is then evaporated from the grid surface based on the temperature difference between the quenched grid spacer and the unquenched fuel rod. The model does check to assure that the total evaporated mass from all sources, including the grid, does not exceed the entrained liquid mass exiting the core mixture.

While the specified void fraction required for licensing evaluations provides reasonable assurance that the selected grid rewet temperature is appropriate for the CE 16x16 NGF design, its acceptability for other designs should be addressed by a licensee as part of its determination that the optional steam cooling model is appropriate for the specific application. This justification should address the sensitivity of the expected void fraction to the grid rewet temperature, because a lower void fraction could result in a lower grid rewet temperature, by about 200 °F. The justification should consider the sensitivity of rewet temperature to other properties not considered in this optional model and the sensitivity of the rewet temperature to the total additional steam cooling this optional model provides.

3.3.2 Spacer Grid Heat Transfer Coefficient

The amount of steam generated from the quenched grid surface depends on the temperature difference between the unquenched fuel rod and the grid surface, through the use of a thermal radiation heat transfer model. The vapor generated by evaporation of the liquid from the grid surface is modeled by a convective heat transfer coefficient.

The convective heat transfer coefficient from the spacer grid to the liquid film on the grid is represented by the Condie-Bengston IV correlation (Reference 13). Westinghouse chose this correlation to be consistent with the approved 1999 EM film boiling model in the CEFLASH-4A and STRIKIN-II computer programs.

This choice for the heat transfer correlation was addressed by Westinghouse in Reference 2 in response to the NRC staff's RAI. The heat transfer coefficient from the grid spacer to the liquid film, as described in the spacer grid heat transfer methods and mechanisms, documented in TR WCAP-10484-P-A, should be representative of the expected heat transfer coefficient following a return to nucleate boiling on the spacer grid. This would be one to two orders of magnitude greater than the values calculated with Condie-Bengston IV, including the heat transfer multiplier based on the Westinghouse BELOCA model, TR WCAP-12945-P-A. The use of Condie-Bengston IV is therefore conservative. The NRC staff finds the use of Condie-Bengston IV acceptable for an EM designed for compliance with 10 CFR Part 50, Appendix K.C.4.e.

The conservatism in the Condie-Bengston IV correlation for the heat transfer offsets some of the Appendix K features of the 1999 EM. For example, the fuel rod temperature is based on the decay heat of +20 percent (10 CFR Part 50, Appendix K, I.A.4) which would result in a higher fuel rod surface temperature. The radiative energy from the fuel rods, which is used to evaporate the water from the grid, is not considered in the energy balance on the rod itself because of the non-mechanistic implementation of the steam cooling model. The fuel rod is not cooled by the expected heat transfer to the grid spacer and the steam generation mechanism is not reduced, as might be expected, until the grid spacer is submerged, as a result of the ECCS reflooding the core, and steam generation on the grid is turned off. Overall, the fuel rod surface temperature is higher than expected and the amount of steam generated is overestimated.

The use of some of the 1999 EM Appendix K model features and the non-mechanistic implementation of the steam cooling model, while potentially overestimating the steam generation, is offset by the use of the Condie-Bengston IV correlation. However, the model's acceptability for other designs should be addressed by a licensee as part of its determination that the optional steam cooling model is appropriate for the specific application. This justification should consider the amount of additional steam being generated and the effect of the additional cooling for both the peak cladding temperature and the local oxidation, as appropriate. The non-mechanistic implementation of the steam cooling model may lead to more steam generation than would be reasonable and may lead to non-conservative values of steam cooling under certain conditions.

3.3.3 Grid Spacer Geometry and Flow Blockage

Westinghouse has examined the range of flow blockages for which the model has been used in Westinghouse-designed plants. Based on that review, the optional steam cooling model will only be used for cases covered by flow blockage limits which have shown that the model behaves as expected, as explained in Reference 4, Clarification Question 1.

Similarly, the optional steam cooling model will only be used for cases covered by the Reynolds number limits which have shown that the model behaves as expected, as explained in

Reference 4, Clarification Question 1. As used in this context, the Reynolds number relates to the coolant channel hydraulic diameter.

The surface area of the grid was approximated as a square lattice with the same pitch. This approximation was found to be appropriate and acceptable for the CE 16x16 NGF design. This approximation should be justified for other grid designs as the assumption may be inappropriate for other designs.

3.3.4 Sample Analyses

Westinghouse has performed analyses for the CE 16x16 NGF design with and without the optional steam cooling model in order to demonstrate the anticipated impact of the proposed change on the ECCS performance criteria. The effects of flow blockage were also evaluated. These studies showed no significant impact on the peak cladding temperature. The reduction in the local oxidation was shown to be small, consistent with the expected 0.5 percent to 1 percent reduction expected by Westinghouse.

4.0 LIMITATIONS AND CONDITIONS

1. Condition for the Spacer Grid Rewet Temperature

While the specified void fraction required for licensing evaluations provides reasonable assurance that the selected grid rewet temperature is appropriate for the CE 16x16 NGF design, its acceptability for other designs should be addressed by a licensee as part of its determination that the optional steam cooling model is appropriate for the specific application. This justification should address the sensitivity of the expected void fraction to the grid rewet temperature, because a lower void fraction could result in a lower grid rewet temperature, by about 200 °F. The justification should consider the sensitivity of rewet temperature to other properties not considered in this optional model and the sensitivity of the rewet temperature to the total additional steam cooling this optional model provides.

2. Condition for the Spacer Grid Heat Transfer Coefficient

The use of some of the 1999 EM Appendix K model features and the non-mechanistic implementation of the steam cooling model, while potentially overestimating the steam generation, is offset by the use of the Condie-Bengston IV correlation. However, the model's acceptability for other designs should be addressed by a licensee as part of its determination that the optional steam cooling model is appropriate for the specific application. This justification should consider the amount of additional steam being generated and the effect of the additional cooling for both the peak cladding temperature and the local oxidation, as appropriate. The non-mechanistic implementation of the steam cooling model may lead to more steam generation than would be reasonable and may lead to non-conservative values of steam cooling under certain conditions.

3. Limitation on the Optional Steam Cooling Heat Transfer Model

The result of the grid model enhancement cannot result in the use of a heat transfer coefficient greater than FLECHT. The FLECHT upper-bound heat transfer coefficient, as required by the current NRC licensing constraint, is also applied to the spacer grid optional steam cooling model improvement.

4. Use of the Optional Steam Cooling Model

If a licensee wants to use the optional steam cooling model, then a license amendment request should be submitted including the analyses performed to determine its applicability to the specific fuel design being evaluated, as discussed in Section 3.3.1, 3.3.2, and 3.3.3 above. In addition, the licensee should provide the results of the evaluation with and without the optional steam cooling model, in a format similar to the graphical results provided in the reference calculations presented in the supplemental TR. The peak cladding temperature, local oxidation, and steam cooling flow rates should be included in the submittal. These comparisons will enable the NRC staff to confirm the acceptability of the use of the optional steam cooling model.

5. Use of Flow Blockage and Reynolds Number Limits (Section 3.3.3)

For use of this topical report at a specific plant, the flow blockage and Reynolds number limits, as discussed in Section 3.3.3 above, should be confirmed by plant-specific analyses.

5.0 CONCLUSION

For CE-designed PWRs, the NRC staff finds acceptable Westinghouse's proposal to implement an optional steam cooling model for reflood rates less than 1 in/sec, which is applied to the hot rod rupture node elevation and above, into the currently approved 1999 EM. The proposed rewet temperature and heat transfer correlation used in the model were shown to be conservative and are acceptable.

Based on reference calculations performed by Westinghouse during this review, and subject to limitations and conditions 3, 4, and 5, identified in Section 4, the NRC staff finds the use of the optional steam cooling model appropriate for the CE 16x16 NGF design and acceptable for licensing evaluations.

Based on reference calculations performed by Westinghouse during this review, and subject to all the limitations and conditions identified in Section 4, the NRC staff finds the use of the optional steam cooling model appropriate for other CE fuel designs and acceptable for licensing evaluations for CE-designed PWRs.

6.0 REFERENCES

1. Westinghouse Electric Company, "Submittal of CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," and CENPD-132 Supplement 4-NP-A, Addendum 1-NP "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood," (Proprietary/Non-Proprietary)," LTR-NRC-06-25, May 11, 2006.

2. Westinghouse Electric Company, "Response to NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation Topical Report (TR) CENPD-132-P, Supplement 4-P-A, Addendum 1-P; "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model - Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood" (Proprietary/Non-proprietary)," (ADAMS Accession No. ML070190165), LTR-NRC-07-2, January 10, 2007.

3. Westinghouse Electric Company, "Clarification Response to NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation Topical Report (TR) CENPD-132-P, Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model - Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood" (ADAMS Accession No. ML070470379), February 1, 2007.

4. Westinghouse Electric Company, "Clarification Response to NRC's Request for Additional Information by the Office of Nuclear Reactor Regulation Topical Report (TR) CENPD-132-P, Supplement 4-P-A, Addendum 1-P; "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model - Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less Than 1 in/sec Core Reflood" (Proprietary/Non-proprietary)," (ADAMS Accession No. ML071020487), LTR-NRC-07-18, March 28, 2007.

5. CENPD-134-P, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," August 1974; CENPD-134-P, Supplement 1, February 1975; and CENPD-134-P, Supplement 2-A, June 1985.

6. CENPD-138-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974; CENPD-138-P, Supplement 1, February 1975; and CENPD-138-P, Supplement 2-P, January 1977.

7. Combustion Engineering, Inc., "Enclosure 1-P-A to letter LD-81-095, LD-81-095-ENCLIP-A Rev: 000, EDMS CDCC Number: 144893, "C-E ECCS Evaluation Model - Flow Blockage Analysis ECCS Analysis," December 1981.

8. CENPD-135-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974; CENPD-135-P, Supplement 2, February 1975; CENPD-135P, Supplement 4-P, August 1976; and CENPD-135-P, Supplement 5, April 1977.

9. CENPD-213-P, "Application of FLECHT Reflood Heat Transfer Coefficients to C-E's 16x16 Fuel Bundles," January 1976.

10. WCAP-16500-P, “CE 16x16 Next Generation Fuel Core Reference Report,” February 2006, (ADAMS Accession No. ML060670514).

11. WCAP-10484-P-A, “Spacer Grid Heat Transfer Effects During Reflood,” March 1991.

12. WCAP-112945-P-A, Volume1 (Revision 2) and Volumes 2 through 5 (Revision 1), “Code Qualification Document for Best Estimate LOCA Analysis,” March 1998.

13. "Post-CHF Heat Transfer - Analysis, Comparison, and Correlation," Condie, K.G. , Bengston S.J., Richlen S.J., EG&G, Idaho, Inc., Idaho National Engendering Laboratory.

Principle Contributors: E. Throm, J. Kaizer
Date: June 7, 2007