March 10, 2003

Mr. Philip W. Richardson, Manager
Windsor Nuclear Licensing
Westinghouse Electric Company
Mail Stop 126009 - 1901
2000 Day Hill Road
Windsor, CT 06095-0500

SUBJECT: ACCEPTANCE FOR REFERENCING TOPICAL REPORT WCAP-15682-P, "WESTINGHOUSE BWR ECCS EVALUATION MODEL: SUPPLEMENT 2 TO CODE DESCRIPTION, QUALIFICATION AND APPLICATION" (TAC NO. MB4276)

Dear Mr. Richardson:

By letter dated February 8, 2002, and its supplement dated October 16, 2002, Westinghouse Electric Company (WEC) submitted WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," to the NRC for staff review and approval. The objective of this topical report (TR) is to introduce improved fuel clad rupture criteria in the loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) evaluation model (EM) and provide qualification bases for the improvement while maintaining the overall conservatism of the previously approved LOCA ECCS EM.

The NRC staff has completed its review of WCAP-15682-P, and concludes that the proposed Westinghouse LOCA ECCS EM change is acceptable. The enclosed safety evaluation (SE) documents the staff's evaluation of WEC's justifications for the proposed changes.

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

The staff requests that WEC publish an accepted version within 3 months of receipt of this letter. The accepted version shall incorporate (1) this letter and the enclosed SE between the title page and the abstract, (2) all requests for additional information from the staff and all associated responses, and (3) a "-A" (designating "accepted") following the report identification symbol.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. 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 2.790.

We do not intend to repeat our review of the matters described in the subject report, and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. Our acceptance applies only to matters approved in the report.

In the event that any comments or questions arise, please contact Drew Holland at (301) 415-1436.

Sincerely,

/RA/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WESTINGHOUSE TOPICAL REPORT WCAP-15682-P,

"WESTINGHOUSE BWR ECCS EVALUATION MODEL: SUPPLEMENT 2 TO CODE

DESCRIPTION, QUALIFICATION AND APPLICATION"

PROJECT NO. 700

1.0 INTRODUCTION AND BACKGROUND

By letter dated February 8, 2002 (Reference 1), Westinghouse Electric Company (WEC) submitted WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," to the NRC for staff review and approval. The objective of this topical report (TR) is to introduce improved fuel clad rupture criteria in the loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) evaluation model (EM) and provide the qualification bases for the improvement while maintaining the overall conservatism of the previously approved LOCA ECCS EM (References 2 and 3).

WCAP-15682-P describes changes to the Westinghouse ECCS EM for boiling water reactors (BWRs). This version of the EM is identified as USA4. The only difference between the USA4 and the previously approved USA2 version is the methodology used to determine when the fuel rod cladding will rupture. The USA2 EM, which predicts cladding rupture when the burst stress criterion is exceeded, is applied in a way that limits the maximum average planar linear heat generation rate (MAPLHGR) to prevent rod-to-rod contact. The USA4 EM predicts cladding rupture when either there is contact with a neighboring rod or the burst stress criterion is exceeded - whichever comes first. The MAPLHGR is limited in the application of the USA4 EM to ensure that the 10 CFR 50.46 (Reference 4) criteria are met. WCAP-15682-P provides the basis for extending cladding rupture criteria to occur when either there is contact between adjacent rods or the burst stress criterion has been exceeded. In response to the staff's request for additional information (RAI), WEC submitted their justification for the proposed changes to WCAP-15682-P in their letter dated October 16, 2002 (Reference 5). The staff's evaluation of WEC's justification for the proposed changes to the Westinghouse BWR ECCS EM follows.

2.0 REGULATORY BASIS

10 CFR 50.46

A LOCA is a postulated accident defined in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Reactors" to determine the design acceptance criteria for the plant's ECCS. There are five specific design acceptance criteria for the plant defined in 10 CFR 50.46:

- Peak cladding temperature - "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

- Maximum cladding oxidation - "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

- Maximum hydrogen generation - "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times 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."

- Coolable geometry - "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

- Long-term cooling - "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The Westinghouse BWR ECCS reload fuel licensing methodology requires demonstration of compliance with the first three acceptance criteria for each new fuel type introduced in a specific plant. Criterion 4 is assured by meeting Criteria 1 and 2. Criterion 5 is demonstrated during the initial review of the plant's ECCS design.

10 CFR Part 50, Appendix K (Reference 6)

Section I.B of 10 CFR Part 50, Appendix K, "Swelling and Rupture of the Cladding and Fuel Rod Parameters," states:

> Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculation shall be based on applicable data in such a way that the degree of swelling and rupture shall be taken into account in calculation of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.
>
> The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

The Westinghouse USA4 EM's compliance with 10 CFR Part 50, Appendix K, is summarized as follows:

> Section 6.2 of CENPD-293-P-A describes the comparison of the mechanistic swelling and rupture model to the applicable set of data. Section 4.1 of WCAP-15682-P

describes the revision to the Westinghouse BWR LOCA EM which considers that burst stress criterion is reached or rod-to-rod contact is predicted. When rod-to-rod contact occurs, rupture is conservatively assumed. When burst occurs due to rod-to-rod contact, limiting the strain to this value provides a reasonable upper bound to the cladding strain in the region defined by 1.5 inches above and below the burst elevation. This strain limit is defined in Section 5.6.3 of CENPD-293-P-A. Therefore, neither the incidence of rupture nor the degree of swelling is underestimated.

3.0 TECHNICAL EVALUATION

The original BWR LOCA EM (USA1), which was approved by the NRC in 1987, is described in Licensing TRs RPB-90-93-P-A and RPB-90-94-P-A. This methodology was revised in 1996 with the USA2 EM described in Licensing TRs CENPD-283-P-A and CENPD-293-P-A.

WCAP-15682-P describes a proposed change to the Westinghouse BWR LOCA EM that is identified as the USA4 EM[1]. The USA2 EM, which predicts cladding rupture when the burst stress criterion is exceeded, is applied in a way that limits the MAPLHGR to prevent rod-to-rod contact. However, the USA4 EM predicts cladding rupture when either there is contact with a neighboring rod or the burst stress criterion is exceeded - whichever comes first. WEC has confirmed that this is the only difference between the USA2 and USA4 versions, therefore, this is the only change being reviewed by the staff for this safety evaluation.

3.1 LOCA Evaluation Model Analysis Process

The application of the Westinghouse BWR LOCA EM to a specific plant consists of the following steps:

- The plant-specific ECCS licensing basis is determined.
- Plant-specific GOBLIN, DRAGON, and CHACHA-3D code models are developed.
- A confirmatory reactor coolant system LOCA break spectrum evaluation is performed to identify the "limiting break" from the potentially limiting breaks defined in the plant licensing bases.
- A set of conservative initial reactor core conditions are defined that bound the expected conditions for each reload cycle that the fuel design in question shall be in the reactor. Initial core conditions related to nuclear design, thermal hydraulics and mechanical properties are defined in CENPD-300-P-A.
- For the limiting break and initial conditions, the MAPLHGR operating limit as a function of exposure throughout the life of the fuel is determined for the reload fuel design to ensure that Criteria 1 and 2 from 10 CFR 50.46 are met.

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1 The USA3 EM uses the ANS 7.1 1979 decay heat standard plus two standard deviations, where the USA2 EM used the ANS 7.1 1971 decay heat standard plus 20 percent. The USA3 EM was not approved for evaluations demonstrating compliance with Appendix K.

- The total hydrogen generation for a core of the particular fuel design is evaluated and confirmed to meet the acceptance limit of Reference 4.

3.2 Major Features of the Westinghouse BWR LOCA Evaluation Model

The major features of the Westinghouse BWR LOCA EM are described in detail in References 2, 3, and 7. The analysis is performed in three parts:

- The response of the reactor system to the LOCA event is determined using the GOBLIN computer code. The analysis models the actuation of automatic features such as the main steam isolation valve closure, reactor scram and the ECCS. This analysis also determines the boundary conditions that are applied to the hot channel analysis.

- The response of the hot assembly is determined using the DRAGON computer code. The DRAGON computer code is essentially the GOBLIN computer code, but several of the features necessary for determining the system response are disabled. This analysis determines the response of the hot channel to the LOCA event (e.g., boiling transition, dryout and refill). These results and the calculated thermal hydraulic conditions in the hot assembly are used to establish the heat transfer coefficients and boundary conditions that are applied to the limiting cross section.

- The response of the limiting cross section of the hot assembly is determined using the CHACHA-3D computer code. CHACHA-3D determines the detailed temperature distribution for all components at the limiting cross section. It includes the effects of cladding oxidation and fuel rod swell and rupture.

3.3 Rod Heat-up Analysis Code Modifications

The Westinghouse BWR LOCA methodology includes a detailed heat transfer analysis of the limiting axial cross section of fuel assembly. This analysis is embodied in the CHACHA-3D computer code. CHACHA-3D is provided time-dependent thermal-hydraulic boundary conditions from the hot assembly thermal-hydraulic analysis (i.e., DRAGON). As described in CENPD-293-P-A, CHACHA-3D accounts for thermal radiation between all relevant structures at that cross section, dimensional changes of the fuel rods resulting from different pressure loading (cladding thickness and outside diameter), and the cladding ductility and fuel pellet gas release.

CHACHA-3D calculates the incidence of rupture by determining when the calculated stress exceeds the value predicted by a mechanistic burst stress model. The burst stress model accounts for the change of material properties with temperature, degree of burnup, as well as the surface oxide and the oxygen that has diffused into the zircaloy cladding. The true circumferential stress is determined from the internal and external pressures, and the transient cladding dimensions. In accordance with CENPD-293-P-A, a bias of -0.5 MPa is added to the calculated burst stress to ensure that rupture is calculated conservatively.

CHACHA-3D analyses are performed in an iterative manner by changing the nodal peaking until the applicable criteria are met. Since the qualification basis of the cladding rupture model in the USA2 EM was based only on single tube test data, WEC’s practice has been to perform the hot plane analyses in a manner that limits the MAPLHGR to a value that prevents either adjacent rods from coming into contact or cladding from exceeding the 10 CFR 50.46 acceptance criteria, whichever is more limiting. Due to the decrease in ductility with increasing burnup, the second criterion becomes limiting later in life. The first criterion is limiting early in life.

The methodology change described below uses available bundle data to justify the assumption of cladding rupture on contact.

Methodology

The USA4 EM criteria for determining fuel rod rupture are that cladding rupture occurs when either the cladding contacts a neighboring rod or the burst stress criterion is exceeded - whichever comes first. The MAPLHGR is limited to a value that ensures the 10 CFR 50.46 acceptance criteria are met.

CHACHA-3D calculates the cladding strain as the sum of the thermal strain and the plastic strain. As discussed in CENPD-293-P-A, elastic strain is not significant in a LOCA analysis. The previously approved model in CHACHA-3D uses a mechanistic model for high temperature plastic strain, which accounts for the oxidation and embrittlement that takes place at high temperatures. When compared to the rupture strain data in NUREG-0630 (Reference 8), the predicted post-rupture strains are scattered above and below measured values as shown in Figure 7-22 of CENPD-293-P-A.

The mechanistic strain model in CHACHA-3D accounts for the change in ductility of Zircaloy with burnup. The decrease in ductility with burnup has an effect on the predicted burst strain post-LOCA. For low burnup (e.g., <25,000 MWd/MtU), fuel rods are predicted to come into contact before the burst stress criterion is met. Since the rods are less ductile at higher burnups, rods are predicted to rupture before contact above 25,000 Mwd/MtU burnup.

Although not stated in CENPD-293-P-A, CHACHA-3D limits the burst strain by initiating cladding rupture when two adjacent fuel rods come in contact. This feature conservatively assumes rupture on contact. As indicated above, this feature of the model was not activated in USA2 licensing applications because rod-to-rod contact was prevented by limiting the allowed nodal peaking.

4.0 CONCLUSION

After reviewing the submittal of WCAP-15682-P with the proposed change to the Westinghouse BWR LOCA ECCS EM, the staff finds that the USA4 EM complies with 10 CFR Part 50, Appendix K, in that the swelling and rupture calculations are based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The USA4 EM conservatively predicts cladding rupture when either there is contact with a neighboring rod or the burst stress criterion is exceeded - whichever comes first. When burst occurs due to rod-to-rod contact, limiting the strain to this value provides a reasonable upper bound to the cladding strain in the region defined by 1.5 inches above and below the burst elevation. The MAPLHGR is limited in the application of the USA4 EM to ensure that the 10 CFR 50.46 criteria are met.

Therefore, on the basis of the above review and justification, the staff concludes that the proposed change to the Westinghouse LOCA ECCS EM is acceptable.

5.0 REFERENCES

1. Letter from Philip W. Richardson (Westinghouse, LTR-NRC-02-5) to NRC, WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 To Code Description, Qualification and Application," dated February 8, 2002.

2. ABB Report RPB-90-93-P-A (Proprietary) and RPB-90-91-NP-A (Nonproprietary), "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," dated October 1991.

   ABB Report RPB-90-94-P-A (Proprietary) and RPB-90-92-NP-A (Nonproprietary), "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," dated December 1991.

   Letter from A. C. Thadani (NRC) to Westinghouse, "Acceptance for Referencing of Licensing Topical Reports WCAP-11284 and WCAP-11427 Regarding the Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model," dated August 22, 1989.

3. CENPD-283-P-A (Proprietary) and CENPD-283-NP-A (Nonproprietary), "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," dated July 1996.

   CENPD-293-P-A (Proprietary) and CENPD-293-NP-A (Nonproprietary), "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," dated July 1996.

4. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

5. Letter from Donald M. Rowland (Westinghouse, LTR-NRC-02-52) to NRC, "Response to NRC RAIs Regarding WCAP-15682-P," dated October 16, 2002.

6. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

7. CENPD-300-P-A (Proprietary) and CENPD-300-NP-A (Nonproprietary), "Reference Safety Report for Boiling Water Reactor Reload Fuel," dated July 1996.

8. NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis"

Principal Contributor: Tanya Ford

Date: March 10, 2003