Westinghouse Non-Proprietary Class 3

WCAP-15682-NP-A

April 2003

Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application

Westinghouse

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP – 15682-NP-A

Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application

April 2003

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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.

P. Richardson -2-

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,

Herbert N. Berkow

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

Westinghouse Electric Company

Project No. 700

cc:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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 planer 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:

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- 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.

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.

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- 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.

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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

ABSTRACT

This Licensing Topical Report describes changes to the Westinghouse Emergency Core Cooling System Evaluation Model for BWRs. This version of the Evaluation Model is identified as USA4. The only difference between this version of the Evaluation Model and the previously approved Evaluation Model (USA2) is the methodology used to determine when the fuel rod cladding will rupture. This document provides the basis for improving the cladding rupture criteria such that rupture occurs when either there is contact between adjacent rods or the burst stress criterion has been exceeded.

The USA2 evaluation model, 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 evaluation model 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 evaluation model to ensure that the 10CFR50.46 criteria are met.

TABLE OF CONTENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 4, 2002

Mr. D. M. Rowland
Manager, Fuel Licensing and Special Projects
Westinghouse Electric Company LLC
5801 Bluff Road
Drawer R
Columbia, SC 29250

SUBJECT: WESTINGHOUSE ELECTRIC COMPANY - REQUEST FOR ADDITIONAL INFORMATION (RAI) ON TOPICAL REPORT WCAP-15682-P, "WESTINGHOUSE BWR ECCS EVALUATION MODEL: SUPPLEMENT 2 TO CODE DESCRIPTION, QUALIFICATION AND APPLICATION" (TAC NO. MB4276)

Dear Mr. Rowland:

By letter dated February 8, 2002, Westinghouse Electric Company submitted for staff review Topical Report WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

The staff has completed its preliminary review of WCAP-15682-P and has identified a number of items for which additional information is needed to continue its review. The enclosed RAI was discussed with your staff on September 9, 2002. A mutually agreeable target date of October 16, 2002, was established for responding to the RAI. Please provide the requested information so that the staff's review can be completed in a timely manner. Partial submittals would be welcomed to minimize delays.

If you have any questions, please call me at (301) 415-1436.

Sincerely,

Drew Holland, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Request for Additional Information

cc w/encl: See next page

Westinghouse Electric Company Project No. 700

cc:
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Mr. H. A. Sepp, Manager
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REQUEST FOR ADDITIONAL INFORMATION

WESTINGHOUSE ELECTRIC COMPANY TOPICAL REPORT WCAP-15682-P,

"WESTINGHOUSE BWR ECCS EVALUATION MODEL:

SUPPLEMENT 2 TO CODE DESCRIPTION, QUALIFICATION AND APPLICATION"

PROJECT NO. 700

The staff is reviewing WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification, and Application," submitted February 8, 2002. To complete the review the following information is requested.

1. WCAP-15682-P states "that the only difference between this version of the Westinghouse ECCS Evaluation Model (EM), and the previously approved USA2 version, is the methodology used to determine when the fuel rod cladding will rupture." Please confirm that no other changes have been made to the previously approved USA2 version.

2. The previously approved USA2 EM was based only on single tube test data, whereas, the proposed USA4 EM uses bundle data to justify the assumption of cladding rupture on contact. Why is contact with adjacent rods a concern now and it was not a concern for the previously approved USA2 version? What has changed to make rod-to-rod touching a concern?

3. Have the limiting conditions for a loss-of-coolant accident changed with the use of the proposed emergency core cooling system EM?

Westinghouse

Westinghouse Electric Company
5801 Bluff Road
Drawer R
Columbia, SC 28250
USA

16 October, 2002
LTR-NRC-02-52

Project No. 700

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

SUBJECT: **RESPONSE TO NRC RAIs REGARDING WCAP-15682-P**
[Enclosure 1-P Contains Westinghouse Proprietary Class 2 Material]

References:
1. Letter, D. Holland (USNRC) to D. M. Rowland (Westinghouse), "Westinghouse Electric Company - Request for Additional Information (RAI) on Topical Report WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application" (TAC NO. MB4276)", October 4, 2002
2. Letter, P. W. Richardson (Westinghouse) to USNRC Document Control Desk, "WCAP-15682-P, Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application", LTR-NRC-02-5, February 8, 2002

On October 4, 2002, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI - Reference 1) regarding its review of WCAP-15682-P. WCAP-15682-P (Reference 2) introduces improved fuel clad rupture criteria in the Loss-of-Coolant Accident (LOCA) Emergency Core Cooling System (ECCS) Evaluation Model (EM) and provides qualification bases of that improvement while maintaining the overall conservatism of the already approved LOCA ECCS EM.

Westinghouse has determined that the RAI response information contained in Enclosure 1-P is proprietary in nature. Consequently, it is requested that this information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790 and that copies of the information be appropriately safeguarded. The reasons for the classification of this information as proprietary are delineated in the affidavit provided in Enclosure 2. Enclosure 3 provides a non-proprietary version of the responses to the RAI.

If you have any questions regarding this matter, please do not hesitate to call Chuck Molnar of my staff at (860) 731-6286 or Bill Harris of our technical staff at (860) 731-1848.

Very truly yours,

Donald M. Rowland 10/16/02

Donald M. Rowland
Manager, Fuel Licensing & Special Projects
Westinghouse Electric Co. LLC

Enclosure(s): As stated

xc: w/Enclosures
R. Caruso (NRC)
T. Ford (NRC)
G. Shukla (NRC)

A BNFL Group company

Responses to NRC Questions

Westinghouse received the following questions from NRC related to their review of WCAP-15682-P (Reference 1). Westinghouse responses to the questions are provided below.

NRC RAI No. 1

WCAP-15682-P states "that the only difference between this version of the Westinghouse ECCS Evaluation Model (EM) and the previously approved USA2 version is the methodology used to determine when the fuel rod cladding will rupture." Please confirm that no other changes have been made to the previously approved USA2 version.

Westinghouse Response

No methodology changes have been made between the USA2 and USA4 versions of the EM.

The USA3 EM, which was submitted to NRC in Appendix D of Reference 2, is identical to the USA2 EM except for its use of the ANS79 decay heat model. The NRC Safety Evaluation Report on CENPD-300-P-A indicated that use of the ANS79 decay heat model is not suitable for an Appendix K ECCS EM. Westinghouse does not use the USA3 EM in licensing applications.

NRC RAI No. 2

The previously approved USA2 EM was based only on single tube test data, whereas, the proposed USA4 EM uses bundle data to justify the assumption of cladding rupture on contact. Why is contact with adjacent rods a concern now and it was not a concern for the previously approved USA2 version? What has changed to make rod-to-rod touching a concern?

Westinghouse Response

Qualification of the USA2 EM involved comparison of the incidence of rupture and the degree of clad swelling to test data obtained from single tube tests (e.g., Reference 3). Since multi-tube test data were not used in the qualification of the USA2 EM, the occurrence of rod to rod contact was outside the range of qualification, which was submitted, reviewed and approved by the NRC staff in Reference 4. In applications of the USA2 EM, the limiting MAPLHGR may be determined as that value that precludes rod-to-rod contact. This occurs early in burnup when the cladding is more ductile than it is later in life. [ ] As a result, plant operation is unnecessarily restricted by the limitation in USA2 EM qualification basis. The USA4 EM removes this limitation by expanding the qualification basis using available tube bundle test data.

The multiple tube tests discussed in Section 4.1.3 of WCAP-15682-P provide the basis for limiting burst strain to [ ] As discussed in WCAP-15682-P, [ ].

NRC RAI No. 3

Have the limiting conditions for a LOCA changed with the use of the proposed ECCS EM?

Westinghouse Response

The limiting conditions for a LOCA (e.g., the limiting break size and location, the limiting single failure, etc.) are determined from the thermal-hydraulic analyses that determine the responses of the system and the hot assembly. These analyses use the GOBLIN/DRAGON computer code. The proposed methodology change affects the analysis of the limiting hot plane, which uses the CHACHA computer code. The proposed methodology change will affect the MAPLHGR limits that are provided on a cycle-specific basis, but the limiting conditions determined from the response of the system and hot assembly to the LOCA will not change.

References:

1. D. Holland, "Westinghouse Electric Company – Request for Additional Information (RAI) on Topical Report WCAP-15682-P, 'Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application,' (TAC No. MB4276)," October 4, 2002

2. "Reference Safety Report for Boiling Water Reactor Reload Fuel: Response to Request for Additional Information," CENPD-300-P-RAI, Rev. 1, August 1955

3. D. A. Powers, R. D. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630

4. "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," CENPD-293-P-A, July 1996

1 INTRODUCTION

The objective of this Licensing Topical Report Supplement is to extend the LOCA ECCS Evaluation Model qualification basis of the fuel rod rupture model in heat-up calculations while maintaining the overall conservatism of the LOCA ECCS Evaluation Model.

1.1 Background

The licensing of the Westinghouse BWR reload fuel safety analysis methodology for U.S. applications started in 1982 with the submittal of various Licensing Topical Reports by the Westinghouse Electric Corporation. These reports described codes and methodology developed by Westinghouse Atom AB, formerly known as ABB Atom (and ASEA Atom) of Sweden.

In 1988, ABB Atom continued the licensing of the BWR reload methodology, started by Westinghouse, directly with the NRC. The transfer of the licensing effort was formally facilitated by ABB resubmitting NRC approved Licensing Topical Reports under the ABB ownership. The NRC acknowledged the transfer of the Licensing Topical Reports approvals in 1992 (Reference 1).

After the acquisition of Combustion Engineering by the parent company of ABB Atom, the U.S. operations of ABB Atom were consolidated within ABB Combustion Engineering. ABB Combustion Engineering became the cognizant organization for BWR reload fuel application in the United States. Reference 2 describes the ABB BWR reload methodology that is currently used for U.S. reload and plant operational modification applications.

ABB nuclear businesses were acquired by Westinghouse Electric Company (the successor company of the Westinghouse Electric Corporation nuclear businesses) in April 2000. The cognizant organization responsible for the U.S. application and development of the BWR reload fuel safety analysis methodology within the Westinghouse Electric Company remains unchanged.

The Westinghouse BWR Loss of Coolant Accident (LOCA) emergency core cooling system (ECCS) Evaluation Model of References 3 and 4 has been accepted by the NRC and applied in numerous U.S. reload and lead fuel assembly applications since 1989.

2. SUMMARY AND CONCLUSIONS

2.1 Summary

The original BWR LOCA Evaluation Model (USA1), which was approved by the NRC in 1987, is described in the Licensing Topical Reports RPB-90-93-P-A and RPB-90-94-P-A (Reference 3). This methodology was revised in 1996 with the USA2 Evaluation Model described in Licensing Topical Reports CENPD-283-P-A and CENPD-293-P-A (Reference 4).

This Licensing Topical Report describes a change to the Westinghouse BWR LOCA Evaluation Model that is identified as the USA4 Evaluation Model.[1] The USA4 Evaluation Model contains only one change that requires NRC review and approval. The objective of this Topical Report supplement is to introduce improved fuel clad rupture criteria in the LOCA Evaluation Model and provide the qualification bases for that improvement. This change improves the predictive capability of the Evaluation Model while maintaining the conservatisms of 10CFR50, Appendix K requirements (Reference 5). The change to the evaluation model is presented in Section 4 of this Licensing Topical Report.

2.2 Conclusions

The USA4 Evaluation Model continues as an acceptable methodology for establishing BWR MAPLHGR operating limits and demonstrating ECCS performance for Appendix K reload fuel applications. The USA4 Evaluation Model is a straightforward and simple extension of the previously accepted USA1 and USA2 Evaluation Models.

The technical justification for the proposed change to the cladding rupture model is based on data presented in support of previously approved evaluation models for Westinghouse-designed PWRs.

This Licensing Topical Report demonstrates that the modification to the cladding rupture model is acceptable and that the USA4 Evaluation Model may be referenced without further review in future plant licensing applications.

[1] The USA3 Evaluation Model uses the ANS 1979 decay heat standard plus two standard deviations, where the USA2 evaluation model uses the ANS 1971 decay heat standard plus 20%. The USA3 Evaluation Model was not approved for evaluations demonstrating compliance with Appendix K.

3. BWR ECCS EVALUATION MODEL METHODOLOGY OVERVIEW

This section provides an overview of how the application of the methodology to a typical reload is performed. The overview of the BWR ECCS evaluation model is presented by summarizing:

- The ECCS design bases,
- The LOCA Evaluation Model analysis process, and
- Major features of the Westinghouse BWR LOCA Evaluation Model.

3.1 ECCS Design Bases

LOCA is a postulated accident, prescribed in the Code of Federal Regulations Title 10 Part 50.46 (Reference 5), to determine the design acceptance criteria for the plant Emergency Core Cooling System. 10CFR50.46 prescribes five specific design acceptance criteria for the plant:

1. Peak Cladding Temperature – "The calculated maximum fuel rod cladding temperature shall not exceed 2200 °F."
2. Local Oxidation – "The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness before oxidation."
3. Total 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, except the cladding surrounding the plenum volume, were to react."
4. Coolable Geometry – "Calculated changes in core geometry shall be such that the core remains amenable to cooling."
5. Long Term Cooling – "After any calculated successful 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."

As described in (Reference 2), 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.

3.2 LOCA Evaluation Model Analysis Process

The application of the Westinghouse BWR LOCA Evaluation Model to a specific plant follows the process presented in Figure 3-1. It consists of the following steps:

1. The plant-specific ECCS licensing basis is determined.
2. Plant-specific GOBLIN, DRAGON, and CHACHA-3D code models are developed.
3. A confirmatory break spectrum evaluation is performed to identify the "limiting break" from the potentially limiting breaks defined in the plant licensing bases.
4. 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.
5. 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 Section 3.1 are met.
6. The total hydrogen generation for a core of the particular fuel design is evaluated and confirmed to meet the acceptance limit.

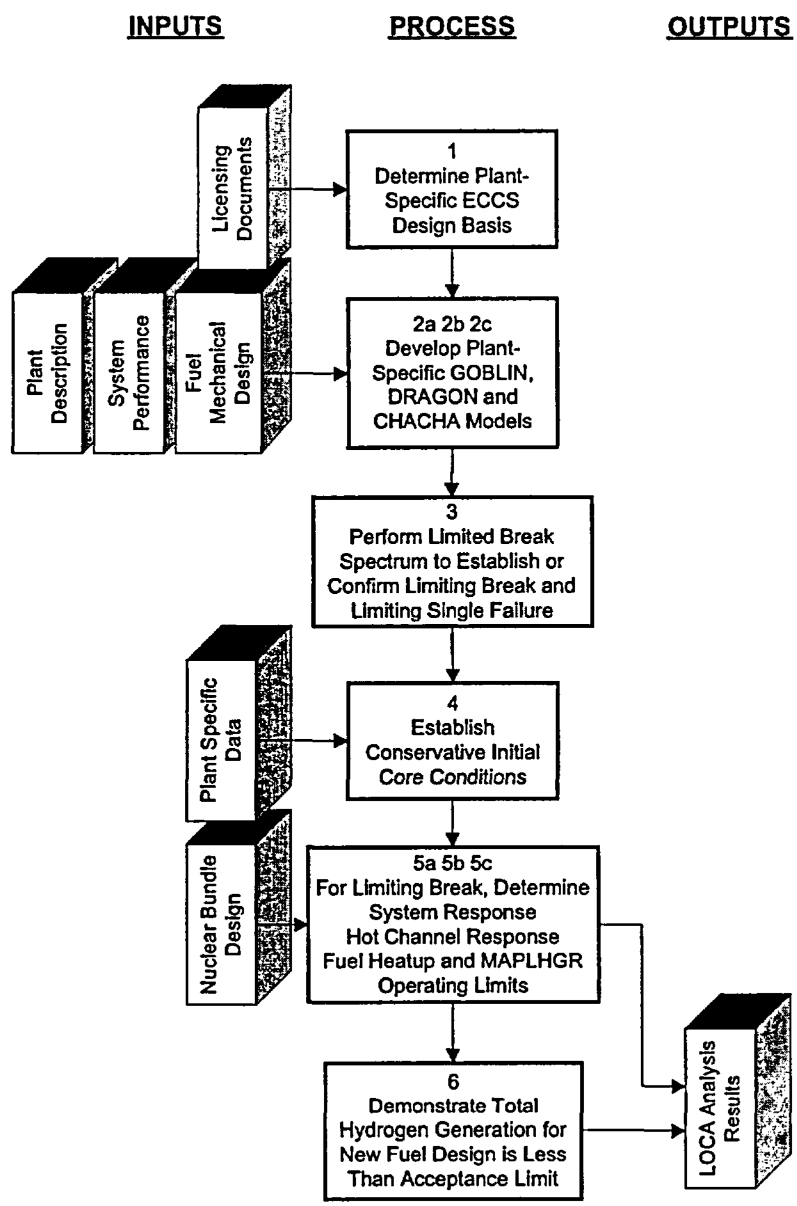

Figure 3-1 Flow Chart of BWR Reload ECCS Performance Analysis

3.3 Major Features of the Westinghouse BWR LOCA Evaluation Model

The major features of the Westinghouse BWR LOCA Evaluation Model are described in detail in References 2, 3, and 4. The analysis is performed in three parts:

1. The response of the reactor system to the LOCA event is determined using the GOBLIN computer code. This analysis models the actuation of automatic features such as MSIV closure, reactor scram and the ECCS. This analysis also determines the boundary conditions that are applied to the hot channel analysis.
2. 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.
3. 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.

Section 4 of this topical report provides the basis for a change to the way the CHACHA-3D computer code is applied.

4. EVALUATION MODEL MODIFICATIONS

The Westinghouse BWR LOCA methodology consists of a set of computer codes, which are described in Section 3.3, and plant-specific application models that utilize those codes to evaluate plant-specific ECCS performance. The following section describes a modification to the rod heat-up analysis code (CHACHA-3D).

4.1 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 (Reference 4), CHACHA-3D accounts for thermal radiation between all relevant structures at that cross section, dimensional changes of the fuel rods resulting from differential pressure loading, 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, 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 (Reference 4), 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 Evaluation Model was based only on single tube test data, Westinghouse'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 10CFR50.46 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.

4.1.1 Methodology

The USA4 Evaluation Model 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 10CFR50.46 acceptance criteria are met.

4.1.2 Discussion

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 6), the predicted post-rupture strains are scattered above and below measured values as shown in Figure 7-22 of CENPD-293-P-A (Reference 4).

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.

Although not stated in CENPD-293-P-A (Reference 4), CHACHA-3D limits the burst strain by initiating cladding rupture when two adjacent fuel rods come in 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.

4.1.3 Qualification

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] Therefore, the degree of cladding swelling is not underestimated by the proposed change to the rupture criteria.

4.1.4 Sample Application

The revised rupture strain methodology was applied to a typical plant-specific analysis using generic fuel performance and physics data. The results are shown in Figure 4-1. The base case uses the USA2 Evaluation Model where the limiting MAPLHGR was determined such that rod-to-rod contact was prevented and the peak cladding temperature remained below 2200°F. Rod rupture was predicted when the calculated stress exceeded the burst stress criterion minus the 0.5 MPa bias discussed in Section 4.1.

The other case uses the USA4 Evaluation Model where the limiting MAPLHGR was determined when the peak cladding temperature approached 2200°F. Cladding rupture was predicted when the calculated stress exceeded the burst stress criterion minus the 0.5 MPa bias, or contact between adjacent rods was predicted. In both cases, CHACHA-3D accounts for oxidation inside the cladding after rupture is predicted.

As shown in Figure 4-1, the USA4 Evaluation Model results in a higher allowed MAPLHGR than the USA2 Evaluation Model [

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Figure 4-2 compares the hot rod cladding temperature at a cycle exposure of 16,000 MWd/MtU when rod-to-rod contact is prevented and when rupture occurs on rod-to-rod contact. As shown, the USA2 Evaluation Model would have limited the MAPLHGR [

] When cladding rupture is predicted, the metal-water reaction is assumed to occur on the inside, as well as the outside, of the cladding at the rupture location. As a result of the additional heat generation, the rate of change of the cladding temperature increases significantly until it is terminated at a peak cladding temperature of 1204°C (2200°F) by the return of two-phase conditions at that location.

Figure 4-1 Plant-Specific Application of Revised Cladding Rupture Model

Figure 4-2 Hot Rod Cladding Temperature at 16,000 MWd/MtU

5. COMPLIANCE WITH APPENDIX K

5.1 Overview

A description of the compliance of the GOBLIN system of codes with 10CFR50 Appendix K is given in Chapter 5 of ABB Report RPB-90-93-P-A (Reference 3) and Chapter 6 of ABB Report CENPD-293-P-A (Reference 4). This section addresses the impact of the methodology changes presented in this report on the previous assessments of compliance – specifically the revised rupture strain and incidence of rupture model.

5.2 Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

Section I.B of Appendix K reads:

> 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 calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations 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 Evaluation Model compliance with Section I.B of Appendix K is summarized as follows:

> Section 6.2 of CENPD-293-P-A (Reference 4) describes the comparison of the mechanistic swelling and rupture model to the applicable set of data. Section 4.1 of this report describes the revision to the Westinghouse BWR LOCA evaluation model which considers burst to occur when either the burst stress criterion is met or rod-to-rod contact is predicted. 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. Therefore, neither the incidence of rupture nor the degree of swelling is underestimated.

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