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# Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application

Westinghouse

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**Revision 0**

# Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application

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# ABSTRACT

This Licensing Topical Report Supplement describes the updates made to the Boiling Water Reactor (BWR) Loss-of-Coolant Accident (LOCA) Emergency Core Cooling System (ECCS) Evaluation Model. The resulting version of the BWR LOCA ECCS Evaluation Model is identified as the USA6 Evaluation Model.

The updates included in the USA6 Evaluation Model for BWR LOCA analysis do not contain any changes to the approved codes. Furthermore, the updates do not constitute a request for exemption from applicable regulation.

This document provides technical and licensing basis for the methodology updates. These updates include a change to the steam cooling heat transfer coefficient applied to the rod heatup calculations for small- and large-break LOCA analysis. A reference to the Westinghouse BWR Containment Methodology using GOTHIC is also introduced as part of the approved evaluation model for calculating the minimum containment backpressure in ECCS analysis.

In addition to the updates, the report contains a road map to the current methodology and clarification on the application of the methodology.

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# ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| ABB | ASEA Brown Boveri |
| ADS | automatic depressurization system |
| ANS | American Nuclear Society |
| AOR | analysis of record |
| ASME | American Society of Mechanical Engineers |
| BWR | boiling water reactor |
| CCFL | countercurrent flow limit |
| CFR | Code of Federal Regulations |
| CHF | critical heat flux |
| CPR | critical power ratio |
| DBA | design basis accident |
| ECCS | emergency core cooling system |
| FRAD | radial power form factor |
| HPCI | high-pressure coolant injection |
| HPCS | high-pressure core spray |
| HTC | heat transfer coefficient |
| HTCM | heat transfer coefficient multiplier |
| LB | large break |
| LBLOCA | large-break loss-of-coolant accident |
| LOCA | loss-of-coolant accident |
| LPCI | low-pressure coolant injection |
| LPCS | low-pressure core spray |
| MAPLHGR | maximum average planar linear heat generation rate |
| M&E | mass and energy |
| NRC | Nuclear Regulatory Commission |
| PCT | peak cladding temperature |
| SER | Safety Evaluation Report |
| TER | Technical Evaluation Report |
| TLTA | two-loop test apparatus |
| TMOL | thermomechanical operating limit |
| UHI | upper head injection |

# 1 INTRODUCTION

The objective of this Licensing Topical Report is to describe the updates made to the Boiling Water Reactor (BWR) Loss-of-Coolant Accident (LOCA) Emergency Core Cooling System (ECCS) Evaluation Model. The resulting version of the BWR LOCA ECCS Evaluation Model is identified as the USA6 Evaluation Model.

The updates to the Evaluation Model include a change to the steam cooling heat transfer coefficient applied to the rod heatup calculations for small- and large-break LOCA analysis. A reference to the Westinghouse BWR Containment Methodology using GOTHIC is also introduced as part of the approved evaluation model for calculating the minimum containment backpressure in ECCS analysis.

In addition to the updates, the report contains a road map to the current methodology and clarification on the methodology that has been applied on a number of plant-specific reload applications since 1989.

## 1.1 BACKGROUND

The licensing of the Westinghouse BWR reload fuel safety analysis methodology for U.S. applications was begun by the Westinghouse Electric Corporation in 1982, with the submittal of various licensing topical reports. 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 Nuclear Regulatory Commission (NRC). The transfer of the licensing effort was formally facilitated by ABB's resubmittal of NRC-approved licensing topical reports under ABB ownership. The NRC acknowledged the transfer of the Licensing Topical Report approvals in 1992 (Reference 1-1).

After acquisition of Combustion Engineering by the parent company of ABB Atom, the U.S. operations of ABB Atom were consolidated within ABB Combustion Engineering, which became the cognizant organization for BWR reload fuel application in the U.S. Reference 1-2 describes the ABB BWR reload methodology that is currently used for the 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 U.S. application and development of the BWR reload fuel safety analysis methodology within the Westinghouse Electric Company remains unchanged.

The Westinghouse BWR ECCS Evaluation Model was originally approved by the NRC in 1989 and is described in RPB 90-93-P-A and RPB 90-94-P-A (Reference 1-3). This methodology was first revised in 1996 to extend its application to SVEA-96 fuel, which is described in CENPD-283-P-A and CENPD-293-P-A (Reference 1-4). Another revision was made to the methodology in 2003, primarily to improve the fuel rod cladding rupture model, which is described in WCAP-15682-P-A (Reference 1-5). A subsequent revision was made in 2004 to extend its application to SVEA-96 Optima 2 fuel, which is described in WCAP-16078-P-A (Reference 1-6).

The Westinghouse BWR LOCA ECCS Evaluation Model of References 1-3 through 1-6 has been accepted by the NRC and applied in numerous U.S. reload and lead fuel assembly applications since 1989.

## 1.2 OBJECTIVE

The Westinghouse BWR ECCS LOCA methodology employs conservative features as prescribed in 10 CFR 50 Appendix K. Although some of these conservatisms are mandated by the Appendix K regulations, others were assumed solely to simplify the analysis. However, to remain under the regulatory limit, under certain conditions, plant operation and core design may be penalized by an overly conservative LOCA analysis. The post-dryout heat transfer is one of the assumptions that is overly conservative in the Westinghouse methodology, originally introduced in 1989. In this report, some of the experimental data indicating that the actual heat transfer from the bundle would be more favorable than currently assumed is presented. Based on the technical evaluations, the methodology is revised to use a more realistic heat transfer coefficient (HTC) for the steam cooling period instead of the adiabatic assumption for convective heat transfer.

This topical report also updates the text for using containment backpressure calculated by Westinghouse generic methodology for BWR containments, WCAP-16608-P (Reference 1-7). Upon approval of WCAP-16608-P, Westinghouse will use a minimum containment pressure that is equal or less than the pressure calculated by GOTHIC.

## 1.3 REFERENCES FOR SECTION 1

1-1 Letter from A. C. Thadani (NRC) to J. Lindner (ABB Atom), "Designation of ABB Atom Topical Reports Related to Licensing of ABB Atom Reload Fuel," June 18, 1992.

1-2 "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A (Proprietary), CENPD-300-NP-A (Non-Proprietary), July 1996.

1-3 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Report RPB 90-93-P-A (Proprietary), RPB 90-91-NP-A (Non-Proprietary), October 1991.

"Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," Westinghouse Report RPB 90-94-P-A (Proprietary), RPB 90-92-NP-A (Non-Proprietary), October 1991.

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

1-4 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," Westinghouse Report CENPD-283-P-A (Proprietary), CENPD-283-NP-A (Non-Proprietary), July 1996.

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

Letter from R. C. Jones (NRC) to D. Ebeling-Koning (ABB-CE), "CENPD-283-P, 'Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel' and Related Documents (TAC M86126)," October 25, 1995.

Letter from R. C. Jones (NRC) to D. Ebeling-Koning (ABB-CE), "CENPD-293-P, 'Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 1 to Code Description and Qualification,' (TAC NO. M90850)," September 28, 1995.

1-5 "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," Westinghouse Report WCAP-15682-P-A (Proprietary), WCAP-15682-NP-A (Non-Proprietary), April 2003.

Letter from H. N. Berkow (NRC) to P. W. Richardson (Westinghouse), "Acceptance for Referencing Topical Report WCAP-15682-P, 'Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application' (TAC NO. M84276)," March 10, 2003.

1-6 "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," Westinghouse Report WCAP-16078-P-A (Proprietary), WCAP-16078-NP-A (Non-Proprietary), November 2004.

Letter from H. B. Berkow (NRC) to J. A. Gresham (Westinghouse), "Final Safety Evaluation for Topical Report WCAP-16078-P, 'Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel' (TAC NO. MB8908)," October 15, 2004.

1-7 "Westinghouse Containment Analysis Methodology," Westinghouse Report WCAP-16608-P (Proprietary), WCAP-16608-NP (Non-Proprietary), August 2006.

# 2 SUMMARY AND CONCLUSIONS

## 2.1 SUMMARY

The original BWR LOCA Evaluation Model (USA1), which was approved by the NRC in 1989, is described in RPB 90-93-P-A and RPB 90-94-P-A. This methodology was revised in 1996 with the USA2 Evaluation Model, which is described in CENPD-283-P-A and CENPD-293-P-A. The methodology was revised again in 2003 with the USA4 Evaluation Model, which is described in WCAP-15682-P-A, and finally in 2004 with the USA5 Evaluation Model, which is described in WCAP-16078-P-A. The USA3 model was never implemented.

Section 3 of this licensing topical report describes changes to the Westinghouse BWR LOCA Evaluation Model that are identified as the USA6 Evaluation Model. The USA6 model contains a change in post-dryout steam cooling heat transfer application that requires NRC review and approval. Additional updates include referencing the Westinghouse BWR Containment Methodology (WCAP-16608-P) for minimum containment backpressure boundary conditions as part of the Westinghouse LOCA methodology.

In addition, numerous administrative changes not requiring technical review are also described in Section 4 of this licensing topical report to provide clarifying details on the application of the methodology. A road map to various key aspects is also included.

## 2.2 CONCLUSIONS

The USA6 Evaluation Model is an acceptable methodology for establishing BWR MAPLHGR operating limits and demonstrating Emergency Core Cooling System (ECCS) performance for Appendix K reload fuel applications. The USA6 Evaluation Model is a straightforward and fully justified extension of the previously accepted USA1, USA2, USA4, and USA5 Evaluation Models.

Post-Dryout Steam Cooling HTC – The technical basis for the proposed change to the post-dryout heat transfer coefficient (HTC) applied to the rod heatup calculation is based on the [

]$^{a,c}$

Containment Backpressure – The containment pressure boundary condition used in large-break LOCA analysis will be calculated by [

]$^{a,c}$ shall be used once approved by the NRC staff.

# 3 EVALUATION MODEL MODIFICATIONS

The Westinghouse BWR LOCA methodology consists of a set of computer codes, which are described in Section 4.2, and plant-specific application models that utilize those codes to evaluate plant-specific ECCS performance. The following sections describe the methodology modifications included in this report.

## 3.1 CONTAINMENT BACKPRESSURE BOUNDARY CONDITION

The original Westinghouse U.S. BWR LOCA methodology uses an atmospheric pressure boundary condition for containment (Reference 3-1). This conservative assumption is a simplification meeting the 10 CFR 50 Appendix K requirements. However, it is not a requirement of the Safety Evaluation Report (SER) and there are no restrictions placed on this aspect of the methodology. Use of non-atmospheric containment backpressure was introduced in December 2006 and, given the conditions delineated in Reference 3-2, accepted by the U.S. NRC as communicated in a letter dated January 2007 (Reference 3-3). As part of the generic BWR Containment Methodology in WCAP-16608-P (Reference 3-4), Westinghouse has developed a methodology to conservatively calculate the minimum containment backpressure for ECCS calculations. Westinghouse will use NRC-approved containment response methodology for conservative pressure boundary condition in the ECCS evaluation calculations. Upon its review and approval, the containment response will be calculated by GOTHIC according to WCAP-16608-P (Reference 3-4).

This change introduces the necessary reference to the containment methodology topical report, for licensing purposes.

## 3.2 POST-DRYOUT HEAT TRANSFER COEFFICIENTS DURING STEAM COOLING

### 3.2.1 Accident Description

The reference LOCA described here is a double-ended guillotine break of a recirculation suction line in a BWR/3. This reference transient assumes failure of the low-pressure coolant injection (LPCI) valve to open with loss of offsite power assumed coincident with the break. The single failure results in the loss of all LPCI injection, but both low-pressure core spray (LPCS) pumps remain operable. The turbine-driven high-pressure coolant injection (HPCI) pump, while operable, will not function as the pressure decays too rapidly.

Following the postulated pipe rupture, coolant discharges rapidly through both sides of the break, with greater flow from the vessel side. The pump-side flow is restricted by the reduced flow area of the jet pump nozzles and friction losses in the recirculation loop and pump.

In the initial blowdown phase, the loss of electrical power causes the recirculation pumps to begin coasting down and rapid closure of the turbine stop valves (the bypass valves are assumed to remain closed). The feedwater pumps are assumed to coast down rapidly, resulting in a rapid loss of inventory out the break. As shown in Figure 3-1, the sudden closure of the steam line results in an initial increase in reactor vessel pressure and an insertion of positive reactivity. At the same time, the coolant discharging into the drywell results in a rapid increase in drywell pressure above the reactor scram setpoint and a reactor scram signal is generated. As the reactor power decreases due to the insertion of control rods, the combination of the reduction in reactor power and the loss of coolant out the break causes the system pressure to decrease. The two-phase mixture level in the downcomer drops rapidly, first uncovering the top of the jet pumps, which results in steam flow into one side of the break. The mixture level continues to drop until the recirculation suction line is uncovered, which results in steam flow into the other side of the break and an increase in the system depressurization rate.

In the intermediate blowdown phase, the subcooling of the liquid in the lower plenum decreases as the system pressure decreases and, as shown in Figure 3-2, the liquid becomes saturated and steam is generated as the liquid changes phase. This is called the lower plenum flashing phase of the event and it has the effect of decreasing the depressurization rate and extending the period of two-phase cooling in the core by preventing countercurrent flow at the inlet of the core. As shown, countercurrent flow at the hot assembly inlet begins at 40.1 seconds.

At the end of the blowdown phase, the operable ECCS pumps begin to restore inventory to the reactor vessel. The ECCS equipment is actuated by either a high drywell pressure signal or the combination of a low-low reactor water level signal and a low reactor pressure signal. For most breaks inside the containment, the high drywell pressure signal occurs first. Assuming a loss of offsite power, the emergency diesel generators provide electrical power to the ECCS equipment. Although the ECCS pumps may be running, the injection valves will not open until the low pressure permissive signal is satisfied. As shown in Figure 3-3, the system mass begins to recover after the ECCS equipment actuates, which, in this case, consists of two LPCS pumps. For this example case, one must account for LPCS leakage from the piping as the coolant flows through the annulus and onto the inside of the core shroud before the coolant enters the spray sparger. The leakage in the annulus is added to the inventory of water in the annulus, which is lost out the break in this example. The leakage inside the upper plenum is added to the inventory in the upper plenum, but is not credited toward achieving rated spray flow. A signal is generated when rated spray flow is delivered to the spray sparger. As shown in Figure 3-3, this occurs at 49.4 seconds. [

]$^{a,c}$

The recovery phase of the event occurs after ECCS actuation and continues until the system mass is stabilized. For the reference LOCA, the water injected into the upper plenum by the LPCS pumps can accumulate in the upper plenum, flow downward into the core bypass region and down into the active core. As shown in Figure 3-4, some mass is predicted to accumulate in the upper plenum after LPCS injection; however, the amount varies with time and is not significant until later in the transient after two-phase conditions are restored. Figure 3-4 also shows the integrated flow rates between the upper plenum and the inter-assembly (i.e., outside the bundles) and intra-assembly (i.e., SVEA water cross region) bypasses, and the active core regions. As shown, the slope of those curves indicates a downward flow into the inter-assembly bypass and active core regions and a negligible flow into the intra-assembly bypass region during the recovery phase. During the recovery phase, the flow at the exit of the core is countercurrent with steam flowing up and liquid flowing down. Figure 3-4 also shows the midplane cladding temperature in the hot assembly as predicted by GOBLIN. As indicated in Section 4.4.3 of this report, the [

]$^{a,c}$

### 3.2.2 Model Change: [

]$^{a,c}$

10 CFR 50 Appendix K, paragraph D.6 provides the following requirements for convective heat transfer coefficients for boiling water reactor fuel rods under spray cooling:

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

As stated above, the regulation indicates that, absent applicable experimental data, a convective heat transfer coefficient of zero is acceptable during the interval (a). Earlier versions of the Westinghouse Evaluation Model used the adiabatic assumption for this phase of the event as described in §D.6.a above.

However, the calculated heat transfer coefficient during this interval, which is due primarily to steam cooling, is not insignificant. As described in Section 5.2.1 of this report, the post-dryout convective heat transfer coefficient used by GOBLIN is based on mechanistic models. The additional qualification provided in this report for [

]$^{a,c}$ A comparison of the heat transfer coefficient predicted by GOBLIN for the reference LOCA described above to the heat transfer coefficient that will be used as a boundary condition for determining the heatup of an interior fuel rod in the CHACHA code is shown in Figure 3-5. As illustrated in this example, the interval over which this revised convective heat transfer model is applied typically is relatively short [

]$^{a,c}$

For a large-break LOCA, the USA6 evaluation model will use [

]$^{a,c}$

For a small-break LOCA, the USA6 evaluation model will use [

]$^{a,c}$

[

]a,c

## 3.3 FIGURES FOR SECTION 3

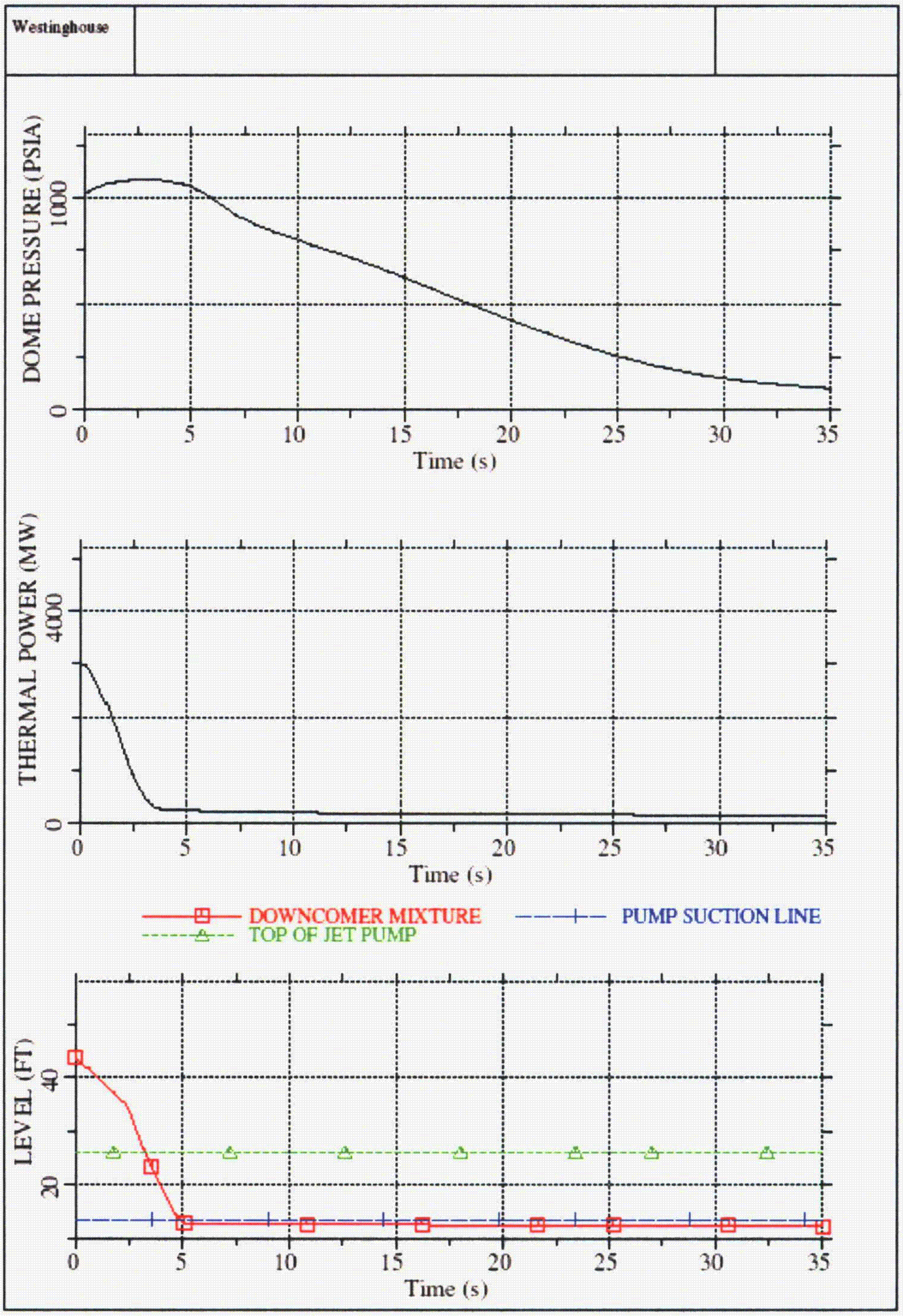


**Figure 3-1 Initial Blowdown Phase of a Large-Break LOCA**

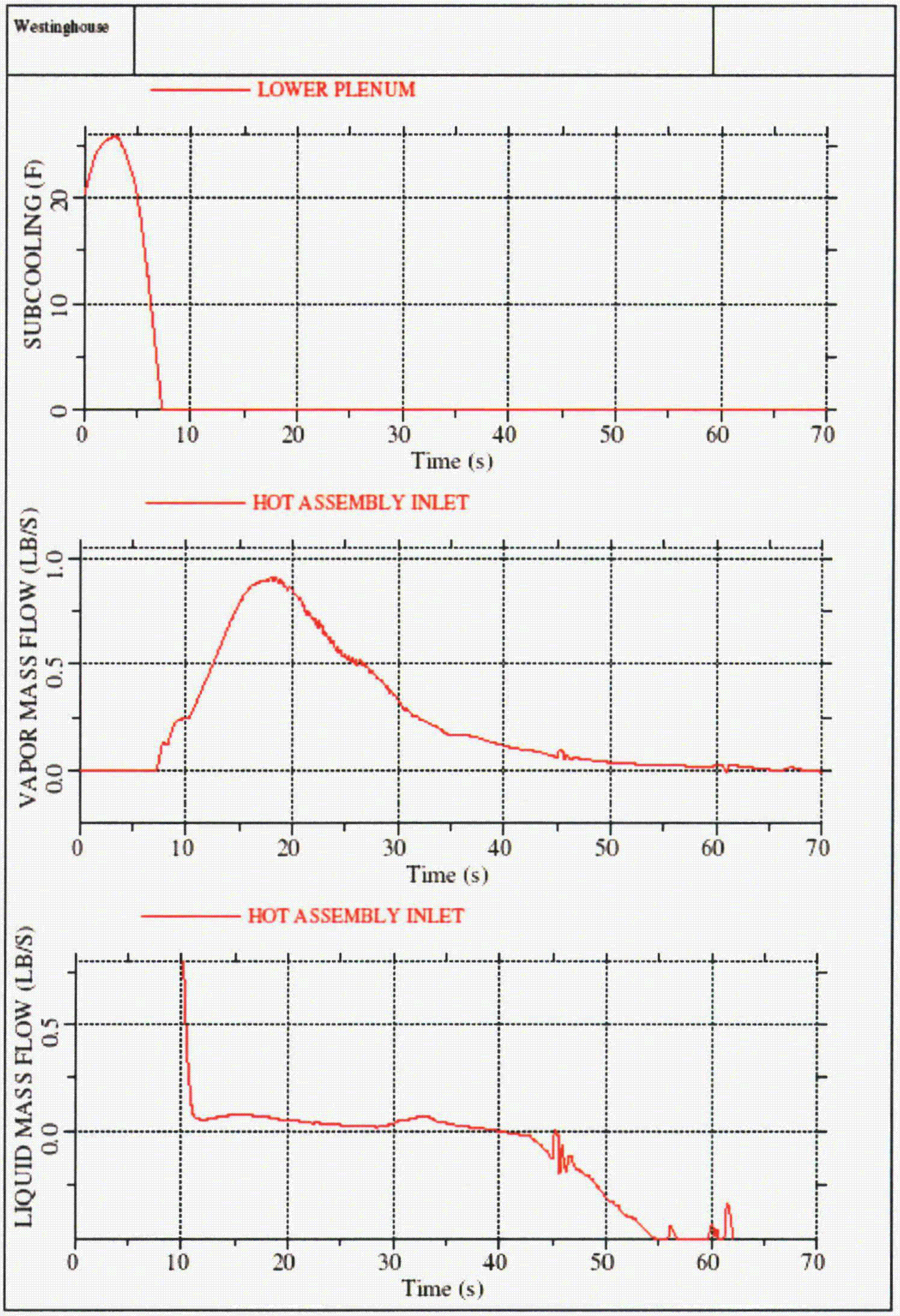


**Figure 3-2 Lower Plenum Flashing Phase of a Large-Break LOCA**

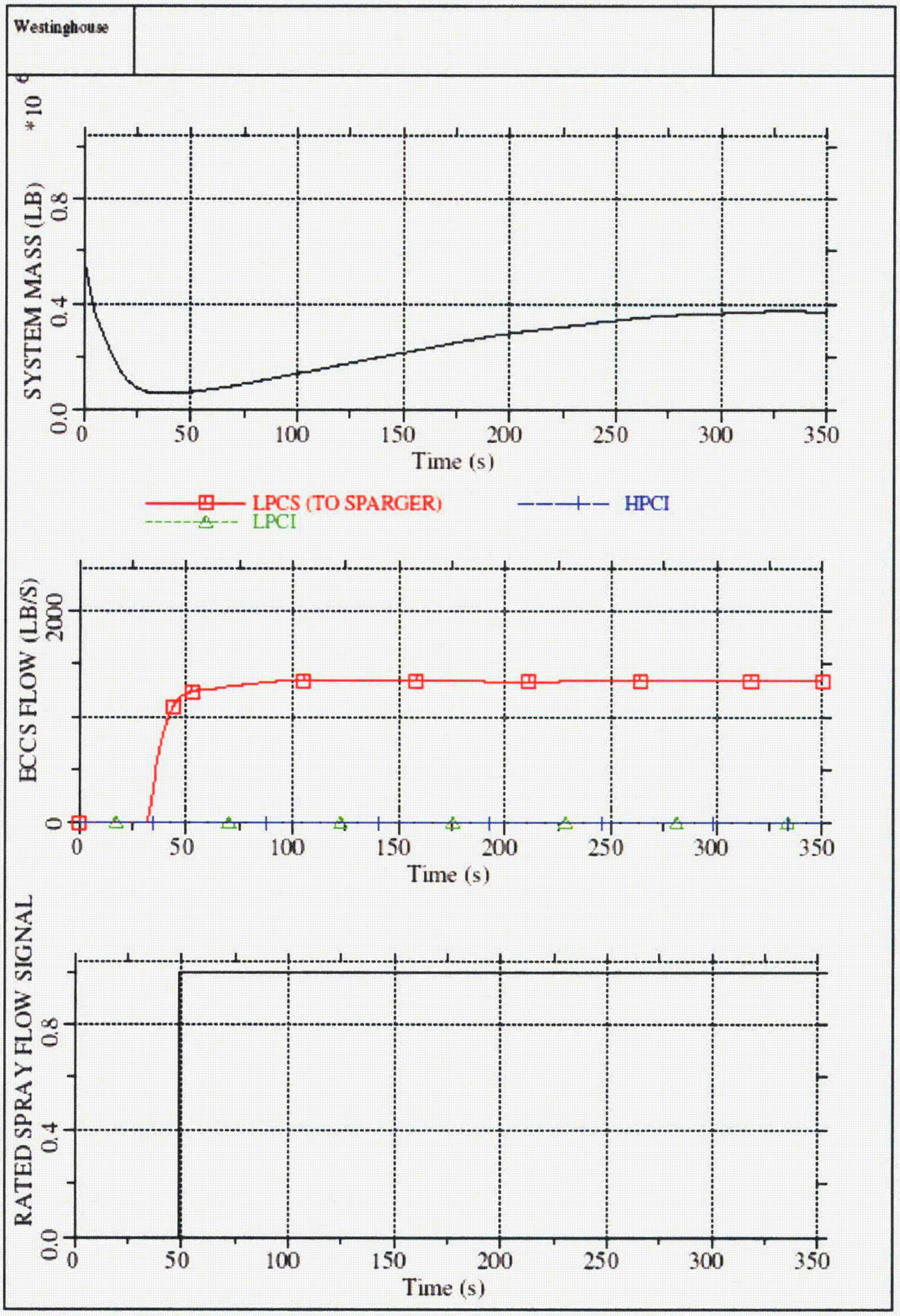


**Figure 3-3 ECCS Actuation Phase of a Large-Break LOCA**

a,c

**Figure 3-4 Recovery Phase of a Large-Break LOCA**

a,c

**Figure 3-5 Revised Convective Heat Transfer Coefficients Used in PCT Calculations**

## 3.4 REFERENCES FOR SECTION 3


3-1 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Report RPB 90-93-P-A (Proprietary), RPB 90-91-NP-A (Non-Proprietary), October 1991.

3-2 Letter from J. A. Gresham (Westinghouse) to U.S. NRC, "Clarification on the Use of Conservatively Calculated Containment Backpressure in Westinghouse BWR ECCS Evaluation Calculations," LTR-NRC-06-67, December 12, 2006.

3-3 Letter from S. L. Rosenberg (NRC) to J. A. Gresham (Westinghouse), "Clarification of the Use of Conservatively Calculated Containment Backpressure in Westinghouse Boiling Water Reactor (BWR) Emergency Core Cooling System (ECCS) Evaluation Calculations," January 8, 2007.

3-4 "Westinghouse Containment Analysis Methodology," Westinghouse Report WCAP-16608-P (Proprietary), WCAP-16608-NP (Non-Proprietary), August 2006.

# 4 BWR ECCS EVALUATION MODEL METHODOLOGY OVERVIEW

This section contains an overview of the Westinghouse BWR LOCA Methodology and repeats information provided in earlier transmittals. However, no changes to the methodology are discussed in this section. An overview of the methodology features, details on application, and clarification of the terms used in the analysis are provided for completeness. The purpose of this section is to assist readers in better understanding the Westinghouse BWR ECCS Methodology.

## 4.1 ECCS DESIGN BASES

A loss-of-coolant accident (LOCA), as defined in Title 10 of the *Code of Federal Regulations*, Section 50.46 (10 CFR 50.46), "Acceptance criteria for emergency core cooling systems for light-water nuclear reactors," is a postulated accident to determine the design acceptance criteria for a plant's ECCS. There are five specific design acceptance criteria for the plant defined in 10 CFR 50.46:

1. Peak cladding temperature (PCT) – "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

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

3. 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."

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 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."

As described in Reference 4-1, 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. Fulfillment of Criterion 4 is assured by meeting Criteria 1 and 2. Criterion 5 is demonstrated during the initial review of the plant's ECCS design.

## 4.2 MAJOR FEATURES OF THE WESTINGHOUSE BWR LOCA EVALUATION MODEL

The GOBLIN series of computer codes uses one-dimensional assumptions and solution techniques to calculate the BWR transient response to both large- and small-break LOCAs. The series is composed of two major computer codes – GOBLIN and CHACHA-3D.

GOBLIN performs the analysis of the LOCA blowdown and reflood thermal-hydraulic transient for the entire reactor, including the interaction with various control and safety systems. The GOBLIN code has what is referred as DRAGON option that can be driven by boundary conditions supplied by the GOBLIN system analysis calculation. This option can be used to analyze the hot assembly in a series fashion rather than as a parallel calculation with the system analysis. The parallel calculation is accomplished by running a two-channel GOBLIN model in which one of the channels represents the hot assembly.

CHACHA-3D performs detailed fuel rod mechanical and thermal response calculations at a specified axial level within the hot fuel assembly previously analyzed by DRAGON or the two-channel GOBLIN model. All necessary fluid boundary conditions are obtained from the hot channel calculation. CHACHA-3D determines the temperature distribution of each rod at a specified axial plane throughout the transient. These results are used to determine the peak cladding temperature and cladding oxidation at the axial plane under investigation. CHACHA-3D also provides input for the calculation of total hydrogen generation.

The flow of information between these codes is shown in Figure 4-1 when a DRAGON analysis is performed. Figure 4-2 shows the flow for a two-channel GOBLIN model. RPB 90-93-P-A (Reference 4-2) provides a detailed description for these codes. CENPD-293-P-A (Reference 4-3), WCAP-15682-P-A (Reference 4-4) and WCAP-16078-P-A (Reference 4-5) describe updates to the various components of the computer codes.

## 4.3 CLARIFICATION OF EVALUATION MODEL APPLICATION DETAILS

### 4.3.1 End of Lower Plenum Flashing

Paragraph D.6 of 10 CFR 50 Appendix K requires the following for convective heat transfer coefficients for BWR fuel rods under spray cooling:

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

As indicated above in the regulation, the timing of the end of lower plenum flashing is an important parameter in the evaluation model as it denotes a transition to a different heat transfer mechanism. Lower plenum flashing occurs during the blowdown process following a large-break LOCA when the saturation temperature becomes equal to the temperature of the fluid in the lower plenum. This results in significant vapor generation in the lower plenum and causes the elevation of the interface between steam and the two-phase mixture in the core to increase. The lower plenum flashing phenomenon has a beneficial effect on the heat transfer between the fuel rods and the coolant in that it delays boiling transition and uncovery.

Previous Westinghouse topical reports have not clearly identified how the end of lower plenum flashing is determined. In the application of the methodology, Westinghouse typically used [

]a,c

In addition to causing an increase in the elevation of the two-phase mixture in the core, the steam flow resulting from lower plenum flashing [

]a,c

Figure 4-3 shows various parameters during the time of lower plenum flashing. From the vapor flow entering the hot assembly, it can be seen that lower plenum flashing starts at 7.4 seconds and that it continues until after 60 seconds. Also shown are the void fractions at three different elevations in the hot assembly between the inlet and the midplane. [

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[

]a,c

### 4.3.2 Rated Spray Flow

As indicated in §D.6 of 10 CFR 50 Appendix K, the time core spray reaches rated flow is another important transition in the analyzing the ECCS performance. Although the code predicts a two-phase mixture to develop in the upper plenum after the initiation of core spray, there is some uncertainty regarding the coverage of all the assemblies by this mixture. Therefore, the Westinghouse evaluation model defines the time of rated spray flow as the time the core spray pump delivers rated spray flow to the spray sparger. Since some plants have identified leakage from spray flow connections in the reactor vessel before the spray flow reaches the spray sparger in the upper plenum, these leakages are also taken into account in the evaluation model. Figure 4-4 shows the ECCS injection flow, its impact on the total system mass, and a signal that shows when the spray flow delivered to the two core spray spargers reaches rated conditions (in this case 4500 gpm). As shown, the spray flow delivered to the spray sparger reaches rated conditions at 49.4 seconds. At this point, the Westinghouse evaluation model transitions to the spray heat transfer coefficients applicable to the particular fuel design (for example, Reference 4-5 for SVEA-96 Optima2 fuel).

### 4.3.3 Two-Phase Reflooding

As indicated in §D.6 of 10 CFR 50 Appendix K, the time of two-phase reflooding at the elevation under consideration is another important transition in the ECCS performance analysis. The transition to two-phase reflooding defines the time of peak cladding temperature (PCT) as the heat transfer coefficient transitions to a value that is large enough to remove more than the local heat generation. Figure 4-5 shows the two-phase recovery phase following a large-break LOCA in a typical BWR/3 assuming a single failure of the LPCI injection valve. The figure shows the predicted void fraction, heat transfer coefficient, and cladding temperature in the hot assembly at three locations from the inlet to the midplane. As shown, the peak cladding temperature at the midplane of the hot assembly occurs at 181.6 seconds. The increase in heat transfer coefficient after the recovery of two-phase conditions at the bottom of the core is a result of increased coolant flow in the channel. The time of PCT at the midplane is also coincident with the time the void fraction at the midplane begins to decrease. The methodology used to determine this transition point was previously described in Section 3.2.2 of Reference 4-7. It acknowledges the effect of the [

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### 4.3.4 Break Spectrum

The break spectrum process includes a series of computer runs performed to identify the most limiting break size, break location, and single failure combination. The ECCS capabilities of the plant have more influence on the outcome compared to the fuel design and the calculation technique. Therefore, existing break spectrum analysis for a given plant is not expected to be very different when repeated by Westinghouse methodology. The Westinghouse methodology verifies the existing plant break spectrum by evaluating the breaks with the potential of becoming limiting. Since the methodology covers both small and large breaks, it is important to employ a consistent methodology to evaluate the break spectrum. In some applications, the break spectrum evaluation was done by [

]$^{a,c}$ This method provides a consistent comparison between different break sizes.

## 4.4 FIGURES FOR SECTION 4

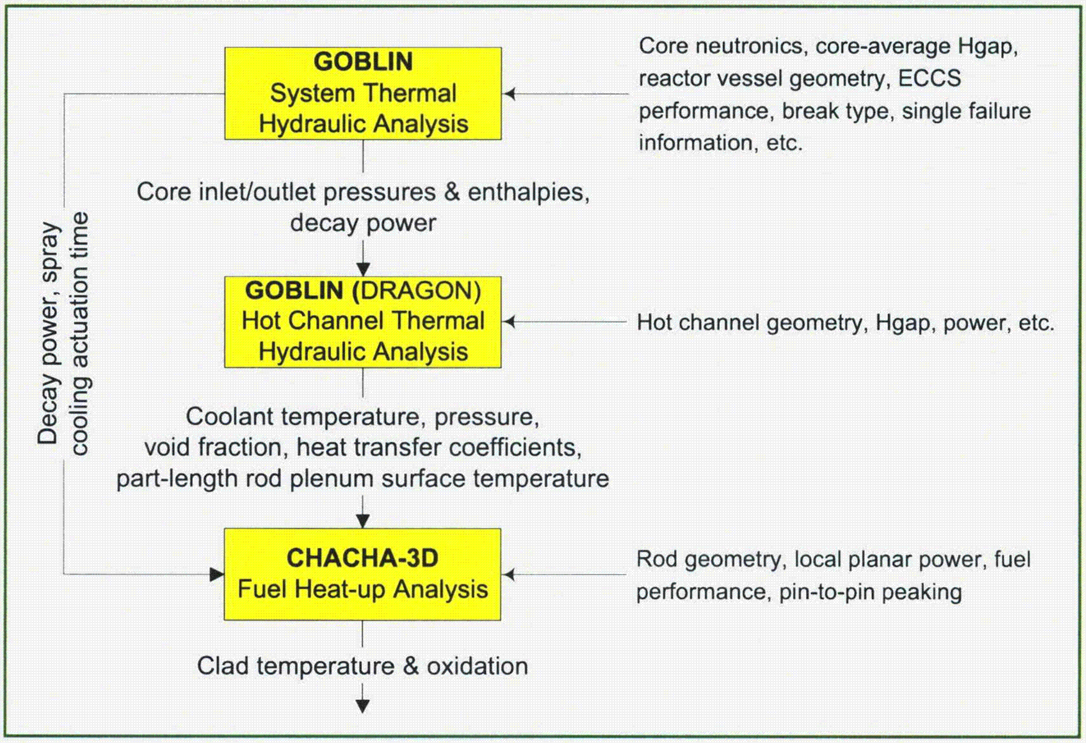


**Figure 4-1 Flow of Information Between Computer Codes for a Single-Channel GOBLIN Model**

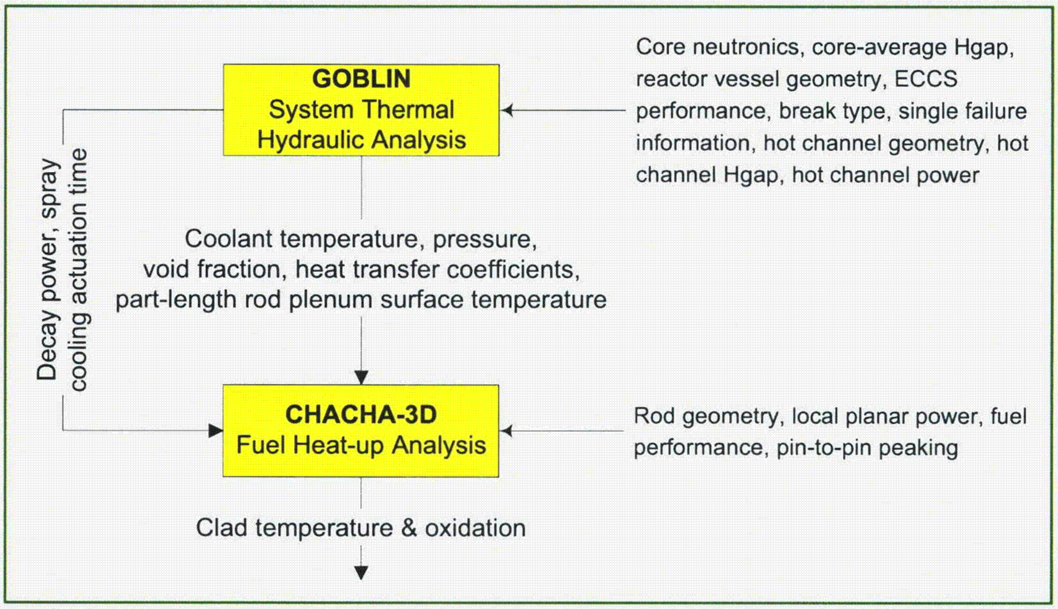


**Figure 4-2 Flow of Information Between Computer Codes for a Two-Channel GOBLIN Model**

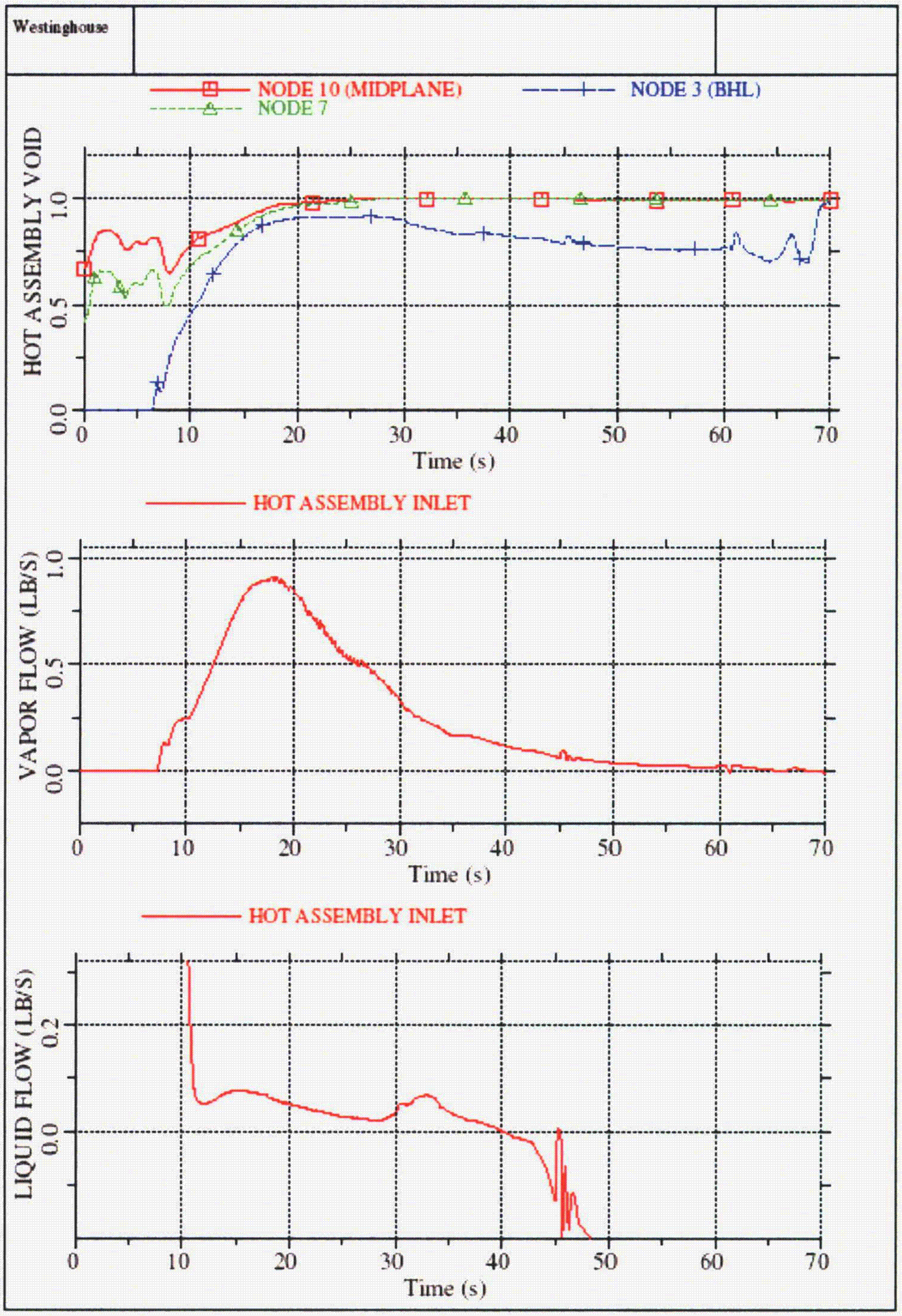


**Figure 4-3 Lower Plenum Flashing**

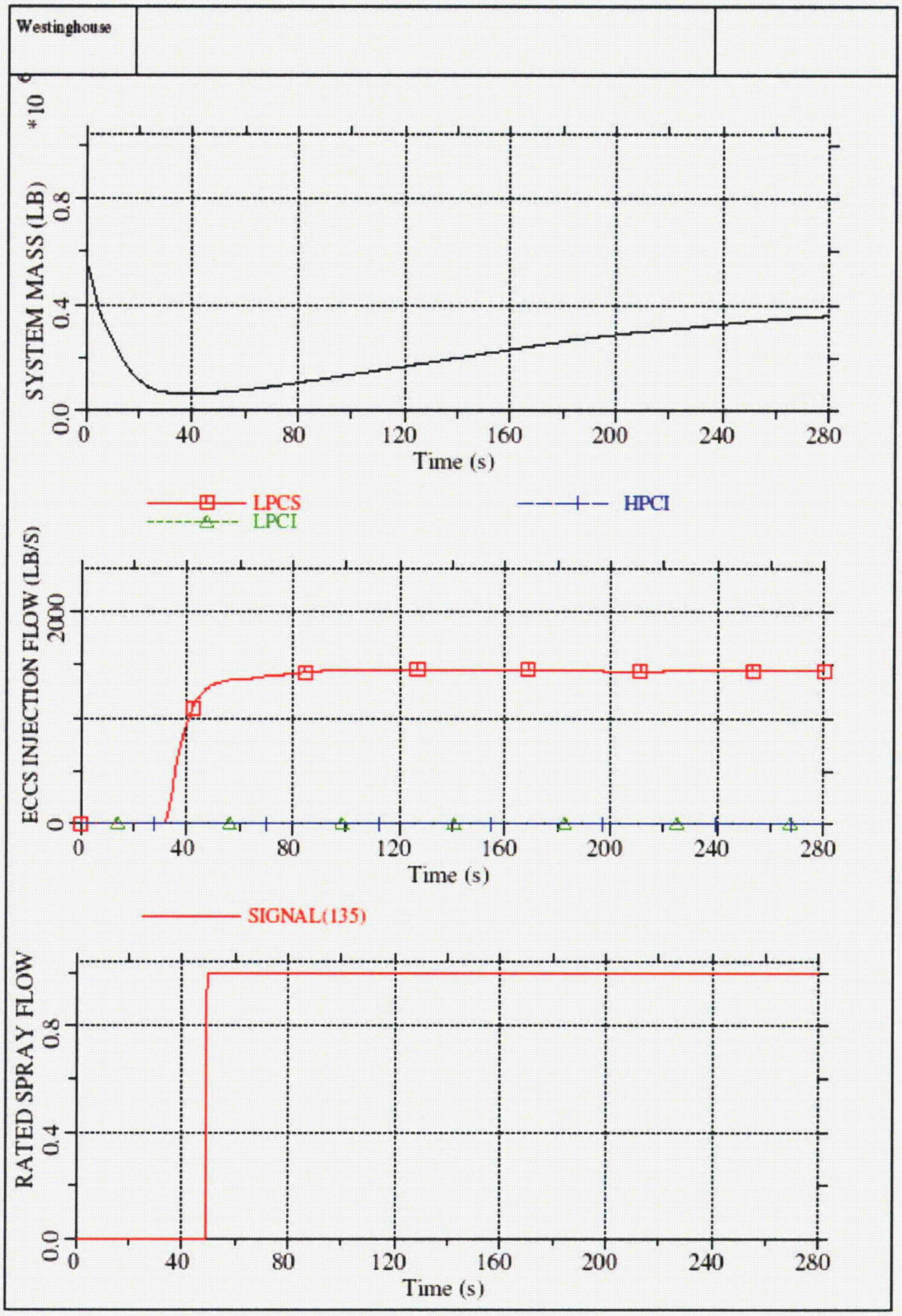


**Figure 4-4 Time of Rated Spray Flow**

a,c

**Figure 4-5 Two-Phase Reflooding**

## 4.5 REFERENCES FOR SECTION 4


4-1 "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A (Proprietary), CENPD-300-NP-A (Non-Proprietary), July 1996.

4-2 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Report RPB 90-93-P-A (Proprietary), RPB 90-91-NP-A (Non-Proprietary), October 1991.

4-3 "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," Westinghouse Report CENPD-293-P-A (Proprietary), CENPD-293-NP-A (Non-Proprietary), July 1996.

4-4 "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," Westinghouse Report WCAP-15682-P-A (Proprietary), WCAP-15682-NP-A (Non-Proprietary), April 2003.

4-5 "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," Westinghouse Report WCAP-16078-P-A (Proprietary), WCAP-16078-NP-A (Non-Proprietary), November 2004.

4-6 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," Westinghouse Report CENPD-283-P-A (Proprietary), CENPD-283-NP-A (Non-Proprietary), July 1996.

4-7 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," Westinghouse Report RPB 90-94-P-A (Proprietary), RPB 90-92-NP-A (Non-Proprietary), October 1991.

4-8 Henry, R. E. and H. K. Fauske, "The Two-Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes," Transactions of the ASME, Journal of Heat Transfer, May 1971, pp. 179-187.

4-9 "SVEA-96 Critical Power Experiments on a Full Scale Sub-bundle," ABB Atom Report UR-89-210-P-A, October 1993.

4-10 Letter from A.C. Thadani (NRC) to W. R. Russell (ABB Atom), "Waiver of CRGR Review of the Safety Evaluation of ABB Supplemental Information Regarding UR 89-210 Safety Evaluation Report," July 12, 1993.

4-11 Letter from B. F. Maurer (Westinghouse) to F. M. Akstulewicz (NRC), "Westinghouse response to Condition 1 in the FINAL SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16078-P, 'Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel' (TAC NO. MB8908), October 21, 2004," LTR-NRC-06-1, January 4, 2006.

4-12 "Fuel Rod Design Methods for Boiling Water Reactors," Westinghouse Report CENPD-285-P-A (Proprietary), CENPD-285-NP-A (Non-Proprietary), July 1996.

4-13 "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Report WCAP-15836-P-A (Proprietary), WCAP-15836-NP-A (Non-Proprietary), April 2006.

4-14 "Fuel Rod Design Methodology for Boiling Water Reactors," Westinghouse Report CENPD-287-P-A (Proprietary), CENPD-287-NP-A (Non-Proprietary), July 1996.

4-15 "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287," Westinghouse Report WCAP-15942-P-A (Proprietary), WCAP-15942-NP-A (Non-Proprietary), March 2006.

# 5 QUALIFICATION OF THE MODEL CHANGES

## 5.1 CONTAINMENT BACKPRESSURE EFFECTS

The containment pressure impacts the cooling effectiveness and flow rates of the ECCS during reflood and spray cooling. [

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The Westinghouse methodology does not apply separate codes to large and small breaks. Therefore, the entire break spectrum is primarily analyzed using GOBLIN, including small, large, and intermediate break sizes. [

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[

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[

]$^{a,c}$

# 5.2 STEAM COOLING VALIDATION

## 5.2.1 GOBLIN Post-Dryout Heat Transfer Model

The GOBLIN post-dryout heat transfer model is described in Section 3.5 of Reference 5-1. The information is repeated here for convenience. Figure 5-1 shows the post-dryout flow regime map used in the GOBLIN code. Of particular interest in this section is the high void fraction regime in which the model makes use of three heat transfer correlations: [

]$^{a,c}$. As shown in Figure 5-1, the code transitions from one correlation to another based on the calculated Reynolds number. These transitions occur where the heat transfer for the two regimes becomes equal.

The post-dryout heat transfer model was qualified in Reference 5-1 by comparison to the Westinghouse G-1 and G-2 tests and the FLECHT low-flow reflooding tests. These comparisons showed that the GOBLIN heat transfer package conservatively calculates the post-dryout heat transfer during a LOCA transient.

The heat transfer model is further qualified in this report as described below. Data from the Two-Loop Test Apparatus (TLTA) large-break simulation tests (Reference 5-5) and the TLTA bundle uncovery tests (Reference 5-6) were used.

## 5.2.2 Large Break

### 5.2.2.1 TLTA Facility Description for Large-Break Tests

A schematic of the overall TLTA facility is shown in Figure 5-2. The facility simulated an 8 x 8 fuel bundle in a scaled BWR/6 with a 624-assembly core. The facility simulates one full-length fuel assembly. All the major regions in a BWR/6 system were modeled in the TLTA. These regions include the lower plenum, core regions, upper plenum, steam separator region, annular downcomer regions, steam dome, and two recirculation loops. While the scaling of the TLTA in most respects simulates typical BWR geometry, the depth of the lower plenum and the elevation of the jet pump inlets relative to the core inlet are not scaled well. For example, the jet pump inlet nozzle in a typical BWR is approximately 9 feet above the bottom of the core. In the TLTA facility, the jet pump inlet nozzle is only 2 feet above the bottom of the heated length. Since the height of the jet pump is very important in the recovery phase of a large-break LOCA, this phase of the TLTA test is not representative of a BWR – especially at the higher elevations of the test section.

As shown in Figure 5-3, the single-assembly core is simulated by 62 electrically heated rods with 2 centrally located water rods in an 8 x 8 array. Note that the water rods in the facility were solid and did not permit water to flow through them. The heater rods were constructed of variable-thickness cladding. The heater rods were powered by passing an electric current through the cladding. The variable thickness of the cladding permitted the simulation of a chopped-cosine axial power shape (1.40 nominal axial peaking factor). The interior of the heater rods was filled with a ceramic insulating material between the inside surface of the cladding and the center portion of the heater rod, which was hollow. Several thermocouples were placed against the inside cladding surface of the heater rods at several elevations to measure the inside surface temperature of the cladding. In addition, a series of pressure transducers measured the axial pressure drop along the heated length.

#### 5.2.2.2 GOBLIN Model Used for TLTA Large-Break Test 6423 Run 3

The heated section of the TLTA facility was represented by 9 nodes in the GOBLIN model described in the code validation section of RPB 90-93-P-A (Reference 5-1). Since the TLTA facility is instrumented extensively with thermocouples and differential pressure measurements, a more detailed model was used in this effort to obtain a more precise estimate of the fluid properties at the location of the instrumentation. As shown in Figure 5-4, an axial noding arrangement whereby the upper and lower nodes were 7.5 inches high and the interior nodes were 5 inches high causes the center of many of the nodes to be aligned with thermocouples. Since this noding resulted in flow paths that did not align precisely with the elevation of the spacer grids, the spacer grids were placed in the flow path at the nearest nodal elevation. The resulting 29-core-node GOBLIN Model is shown in Figure 5-5.

#### 5.2.2.3 Comparison of Predicted and Measured Parameters

Test 6423 Run 3 simulated a large double-ended break of the recirculation suction line. The test simulated an operable high-pressure core spray (HPCS) pump, an operable low-pressure core spray (LPCS) pump, and an operable low-pressure coolant injection (LPCI) pump. As shown in Figure 5-6, the prediction of the dome pressure was very good, which resulted in the actuation of the ECC pumps at approximately the same times as in the test.

The GOBLIN predictions of the cladding surface temperatures at the 71- and 35-inch elevations are compared to the measured data in Figure 5-7 and Figure 5-8.

As shown in Figure 5-7, the measured cladding temperature at the upper elevation indicates significant radial variation in coolant conditions, causing the cladding heatups and cooldowns to start at different times. The data also indicates that the two-phase mixture was never fully restored at this elevation as some of the thermocouples were heating up when the test ended. Since the typical GOBLIN model represents the entire assembly cross-section with a single sub-node, it will not predict radial variations in conditions. However, similar to the data, GOBLIN also predicted that the two-phase mixture was never fully restored as it indicated an initial heatup, followed by a recovery lasting about 75 seconds, which was followed by a heatup that continued until the simulation ended. This behavior occurs because the TLTA jet pumps, being only 24 inches high, cannot establish a two-phase mixture at the 71-inch elevation. Thus, the cooling at this elevation is a result of the spray flow from above and steam flow from below. This behavior is not predicted at the midplane of a BWR because the tops of the jet pumps are typically at 2/3 core height, well above the midplane. Therefore, it is more useful to compare the GOBLIN prediction to the TLTA data at a lower elevation.

As shown in Figure 5-8, the TLTA data at the 35-inch elevation do not indicate the same behavior as the data at the 71-inch elevation. Although only two thermocouple measurements are available, the convective heat transfer coefficient improved dramatically at approximately 118 seconds, when the PCT was measured, followed by a complete quench at approximately 140 seconds. As shown in Figure 5-8, one of the thermocouples in a heater rod next to the channel wall was quenched shortly after the actuation of the LPCS pump. Evidently, the spray water flowed down the channel wall near the location of this heater rod and provided good cooling throughout the test. The GOBLIN prediction is conservative with respect to the measurements, indicating an earlier heatup, a higher predicted PCT, and a later two-phase recovery.

The phase of the event that is of interest in this report is between the end of lower plenum flashing and the time that the spray pumps achieved rated flow. [

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For TLTA Test 6423 Run 3, the period between the end of lower plenum flashing and the time the core spray pumps achieve rated flow is defined in Table 5-1. In this case, the end of lower plenum flashing is defined as described in Section 4.4.1 of this report and the time the core spray pumps achieve rated flow is estimated as 112 seconds, since the exact value of the rated spray flow for the test facility pump is not known. As shown in Figure 5-9, [

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A calculation was performed to predict the cladding temperature at the 35-inch elevation using the following heat transfer coefficients:

- [
- ]a,c
- Time of spray flow to quench: Appendix K spray heat transfer coefficient
- Time of quench to end of test: Appendix K reflood heat transfer coefficient

The time of interest for this study is the second bullet above. The predicted cladding temperatures using this approach are shown as the dashed black line in Figure 5-10. These results are also compared to the traditional Appendix K approach of using a convective heat transfer coefficient of zero between the end of lower plenum flashing and the time the spray pumps reach rated flow. As shown in Figure 5-10, the proposed approach results in an adequately conservative estimate of the TLTA test measured cladding temperature.

### 5.2.3 Small Break

10 CFR 50 Appendix K §D.6 provides the following requirements for convective heat transfer coefficients for BWR fuel rods under spray cooling:

*Following the blowdown period, convective heat transfer shall be calculated based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:*

a. *During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.*

b. *During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5 and 1.5 Btu-hr$^{-1}$-ft$^{-2}$ °F$^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.*

c. *After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu-hr$^{-1}$-ft$^{-2}$ °F$^{-1}$ shall be applied to all fuel rods.*

As the wording of the requirements is intended for application to large breaks, the application of this requirement to small-break analysis is difficult as most small breaks do not present distinct phases of blowdown and lower plenum flashing, which are large-break phenomena. Typically, the system pressure will increase following a small break after steam line isolation. The system pressure is controlled between the open and close setpoints of the safety/relief valves. In the event the high-pressure ECCS fails to actuate, the system pressure is reduced intentionally by actuation of the automatic depressurization system (ADS). Actuation of this system will reduce the system pressure until the low-pressure ECCS can actuate. Flashing of liquid in the lower plenum will occur for an extended period of time during the ADS depressurization and the core may slowly uncover prior to actuation of the low-pressure ECCS.

The high-pressure uncovery process was investigated in the TLTA facility as described in Reference 5-5. These data were used to validate the GOBLIN post-dryout heat transfer model for application to small breaks.

#### 5.2.3.1 TLTA Facility Description for Bundle Uncovery Test 6441

The overall facility description is provided in Section 5.2.2.1. In bundle uncovery test 6441, the feedwater, ECCS, and recirculation systems were not used and the tests were run using constant bundle powers at constant pressures. The primary objective of the test series was to obtain data for evaluating heat transfer in a partially uncovered bundle. A series of five tests was reported in Reference 5-6. These tests were run at three different nominal system pressures (800, 400, and 200 psia) and bundle powers (150, 250, and 400 kW).

The system was brought to the initial test pressure and temperature conditions using the auxiliary heaters in the lower plenum and the recirculation pumps. The system conditions were held steady using manual control of the pressure regulation while the bundle power was turned on in place of the auxiliary heaters.

Subsequently, the recirculation pumps were turned off, allowing the two-phase level within the core region to drop from its upper plenum location due to the boil-off of the bundle inventory. A natural circulation between the heated bundle and the downcomer and bypass was established. The primary measured quantities include system pressure, node differential pressures, fluid temperatures, bundle power, and cladding temperatures on the insides of the heater rods. As the system inventory is boiled away and the level drops to the upper tie plate, the liquid overflow from the bundle drains into the bypass region. A summary of the initial conditions for the various tests is given in Table 5-2. A schematic of the TLTA test facility is shown in Figure 5-2.

#### 5.2.3.2 GOBLIN Model Used for TLTA Bundle Uncovery Test Simulation

Many features of the GOBLIN large-break model were not needed for the simulation of the bundle uncovery tests. Although the same basic model was used, features such as the ECCS delivery systems and the break modeling were deactivated. The approach taken was to establish the prescribed initial conditions for the test and hold these conditions constant until a steady-state situation existed. Then the feedwater flow was stopped and the facility was allowed to boil off at constant power and dome pressure. The time scale of the GOBLIN results was adjusted to be zero when the top node started to uncover. The three tests highlighted in Table 5-2 were simulated. Note that these three tests were run with the same bundle power with the initial two-phase level above the top of the heated length. The dome pressure was the major difference between the three tests simulated.

#### 5.2.3.3 Comparison of Measured and Predicted Parameters

Selected figures from Reference 5-6, which showed pressure drop and thermocouple measurements, were digitized to facilitate comparison to code predictions. A selection of the digitized data from Run 7, Test Point 5 is shown as an example in Figure 5-11 and Figure 5-12.[1] The decreasing differential pressure between two elevations is an indication of the two-phase mixture elevation moving downward as the coolant is boiled away. The cladding thermocouples also provide a measure of the location of the two-phase mixture elevation as they indicate cladding heatup when the flow regime at a particular elevation transitions from mostly liquid to mostly steam. The data for the other simulated tests are similar.

The position of the two-phase mixture height can be estimated by determining when the pressure drop measurement between two elevations begins to decrease and when it reaches a more or less steady low value. That is, one can estimate the two-phase mixture height as being at the top pressure tap when the pressure drop starts to decrease and at the bottom pressure tap when the pressure drop reaches its low value. Similarly, one can estimate the position of the top of the two-phase mixture when the cladding thermocouple begins to show an increase in temperature. This was done for all tests evaluated and is shown in Figure 5-13 for the 790 psia test. Note that all the cladding thermocouple data available was used to develop this figure. As shown, the thermocouple data suggest a higher two-phase mixture than the pressure drop data – especially at the higher elevations. This suggests that there is a region above the relatively low void fraction two-phase mixture that contains sufficient liquid (liquid froth or dispersed droplets) to prevent the cladding from heating up. Above the "froth" region, the heat transfer coefficient

[1] Note that the process used to digitize the data was to select as many points as necessary to reproduce the shape of the original curve. Thus, the spacing between data points in the digitized plot is not uniform.

degrades to steam-only cooling, which does not remove sufficient heat from the cladding to prevent heatup.

The predicted nodal void fraction response for the 790 psia test is shown in Figure 5-14. Note that the time scale shown in the plot is the unadjusted time from the actual GOBLIN run. Note also that the void fraction in node 4,29 (the top node) begins to increase at 866 seconds. This corresponds to the start of bundle uncovery in the GOBLIN prediction. When the nodal void fraction goes to 1.0, one can say that the two-phase mixture has moved past the node lower boundary. These results are also shown in Figure 5-13 with the time scale adjusted by 866 seconds. As shown, the predicted results agree very closely with the estimated two-phase mixture elevation using the differential pressure measurements.

Figure 5-15 compares the measured and predicted cladding temperatures at three elevations for the 790 psia test (a time shift of 866 seconds was applied to the GOBLIN results). As shown, the predicted heatup begins sooner than the observed – especially at the higher elevations. The predicted time of heatup correlates very well with the predicted void fraction in the node approaching 1.0. However, close inspection of Figure 5-15 indicates that the GOBLIN heatup rate is less than the observed heatup rate. This indicates that GOBLIN predicts a more effective heat transfer than existed.

A study was performed to evaluate the ability of GOBLIN in predicting the local heat transfer coefficient. This required the development of a small GOBLIN model, which is shown schematically in Figure 5-16. This model used the so-called DRAGON option in GOBLIN, whereby boundary conditions are provided from the actual GOBLIN run and the heat transfer coefficient at the node in question is supplied by the user as a table. In this case, the predicted heat transfer coefficients from the GOBLIN run at a particular location can be written to a table and the values in the table can be adjusted as desired. [

]a,c

This study was performed at two different elevations for each of the three TLTA tests evaluated. The results are summarized in Figure 5-18. The figure was generated by adjusting the GOBLIN time scale for each location so that the heatup started at the same time. Then the cladding temperatures predicted by GOBLIN at the times corresponding to the digitized TLTA data were determined from the start of heatup to a point in time prior to termination of the test. These results were plotted as predicted versus measured values as shown. Although there was no attempt to bias the result, these results should not be considered as a rigorous statistical treatment as the digitized data were not selected rigorously. [

]a,c

---

[

]a,c

## 5.3 TABLES FOR SECTION 5

**Table 5-1 TLTA LB Test 6432 Run 3: Predicted Conditions Pertaining to 35-Inch Elevation**

| Condition | Time (sec) |
|---|---|
| Transition from Nucleate Boiling | 38.1 |
| Transition to Steam Cooling ($\alpha$= 0.995) | 38.7 |
| Sustained Reverse Liquid Flow at Inlet | 35.0 |
| End of Lower Plenum Flashing | 38.7 |
| Core Spray Flow at Rated Conditions | 112* |

*Estimate

**Table 5-2 TLTA Bundle Uncovery Test 6441: Initial Conditions**

| Test Run | Test Point | System Pressure (psia) | Bundle Power (kW) | Initial 2-Phase Level |
|---|---|---|---|---|
| 7 | 5 | 790 | 250 | Bundle Top |
| 3 | 4 | 393 | 150 | BHL* + 103 in |
| 6 | 1 | 395 | 250 | Bundle Top |
| 3 | 5 | 394 | 400 | Bundle Top |
| 5 | 9 | 195 | 250 | Bundle Top |

* BHL: bottom of heated length

## 5.4 FIGURES FOR SECTION 5

a,c

**Figure 5-1 GOBLIN Post-Dryout Heat Transfer Flow Regime Map**

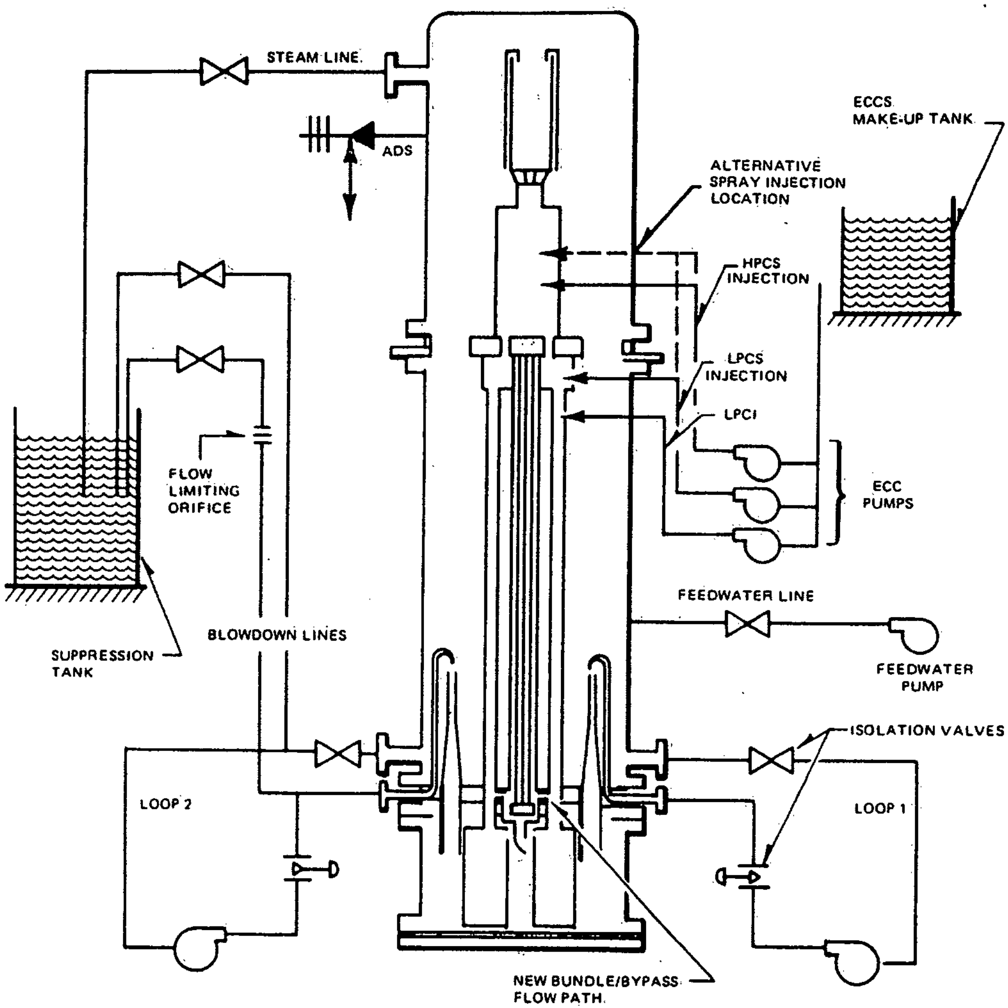


**Figure 5-2 Schematic of the TLTA-5B Facility**

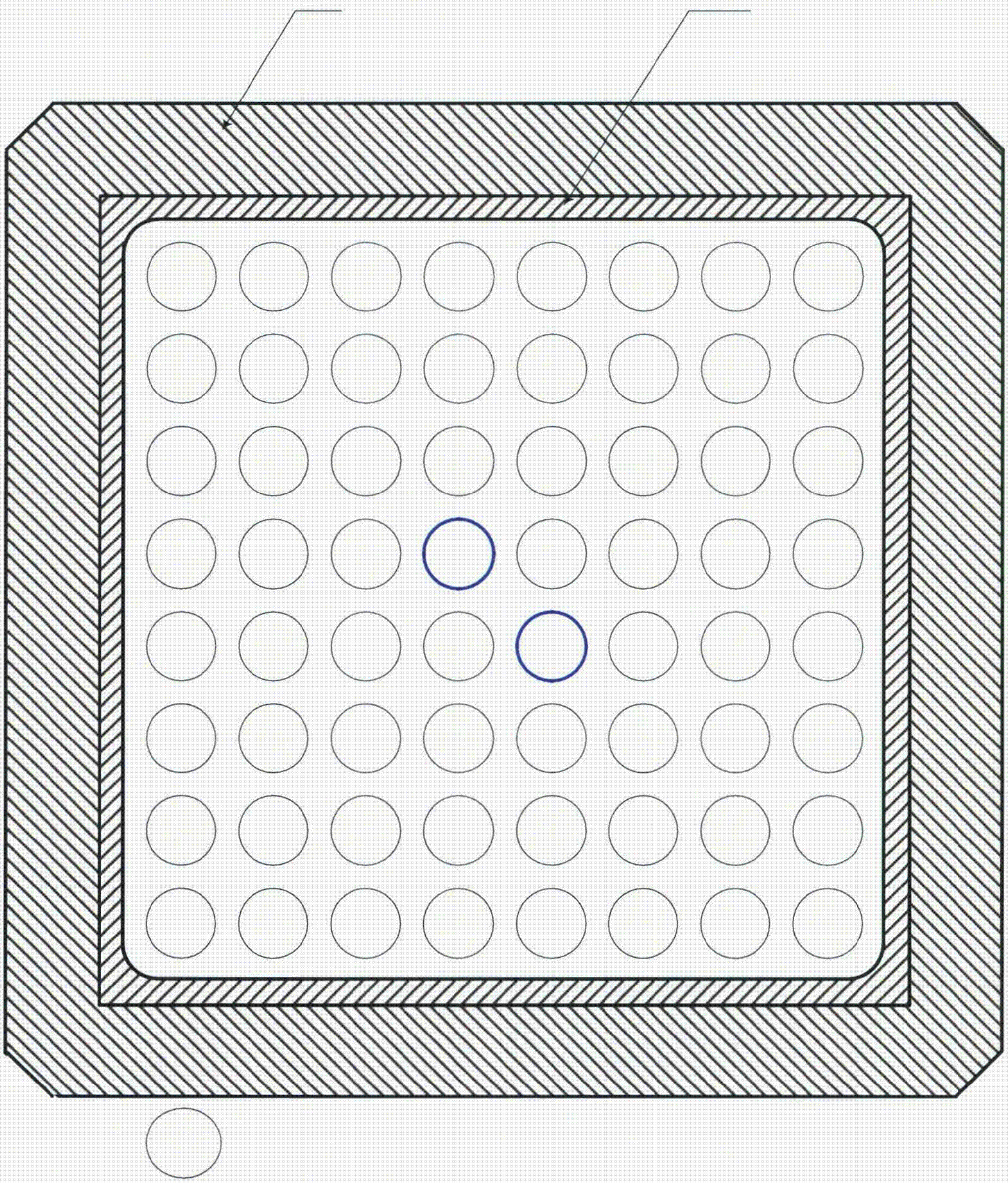

**Figure 5-3 Cross-Section of TLTA Test Section**



1

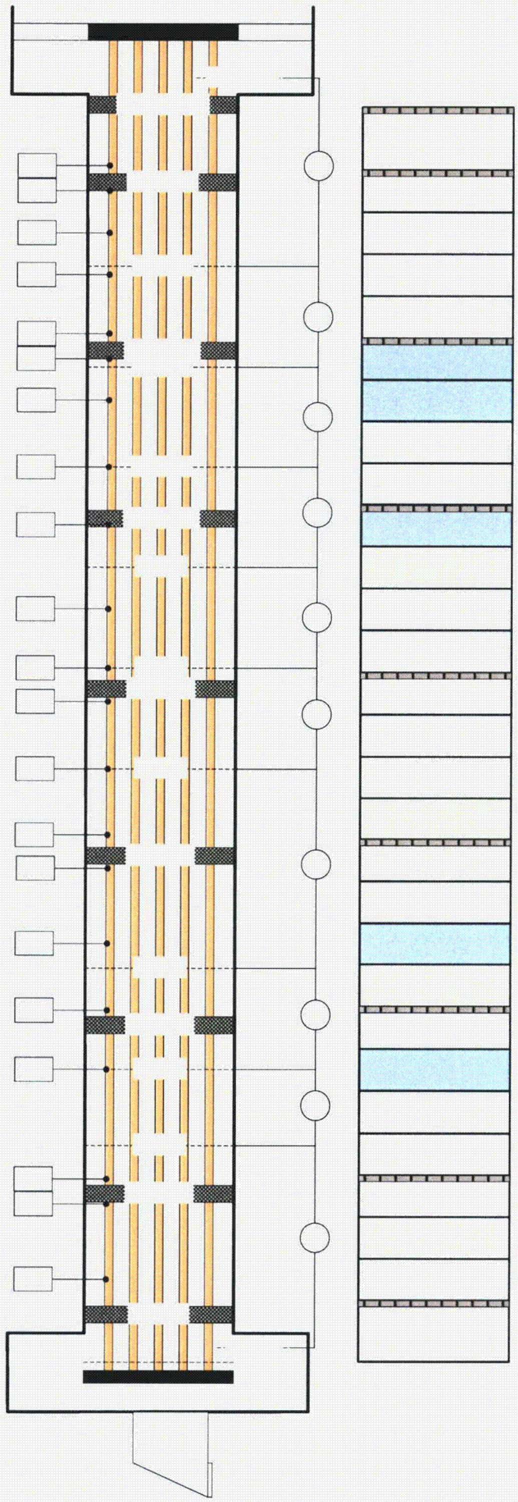

**Figure 5-4 Schematic of TLTA Test Section and Location of Instrumentation**

a,c

**Figure 5-5 GOBLIN Noding Diagram Used for TLTA Facility**

a,c

**Figure 5-6 TLTA LB Test: Dome Pressure**

a,c

**Figure 5-7 TLTA LB Test: Cladding Temperature at 71-Inch Elevation**

a,c

**Figure 5-8 TLTA LB Test: Cladding Temperature at 35-Inch Elevation)**

a,c

**Figure 5-9 TLTA LB Test: End of Lower Plenum Flashing (35-Inch Elevation)**

a,c

Figure 5-10 TLTA LB Test: Predicted vs. Measured Cladding Temperature at 35-Inch Elevation

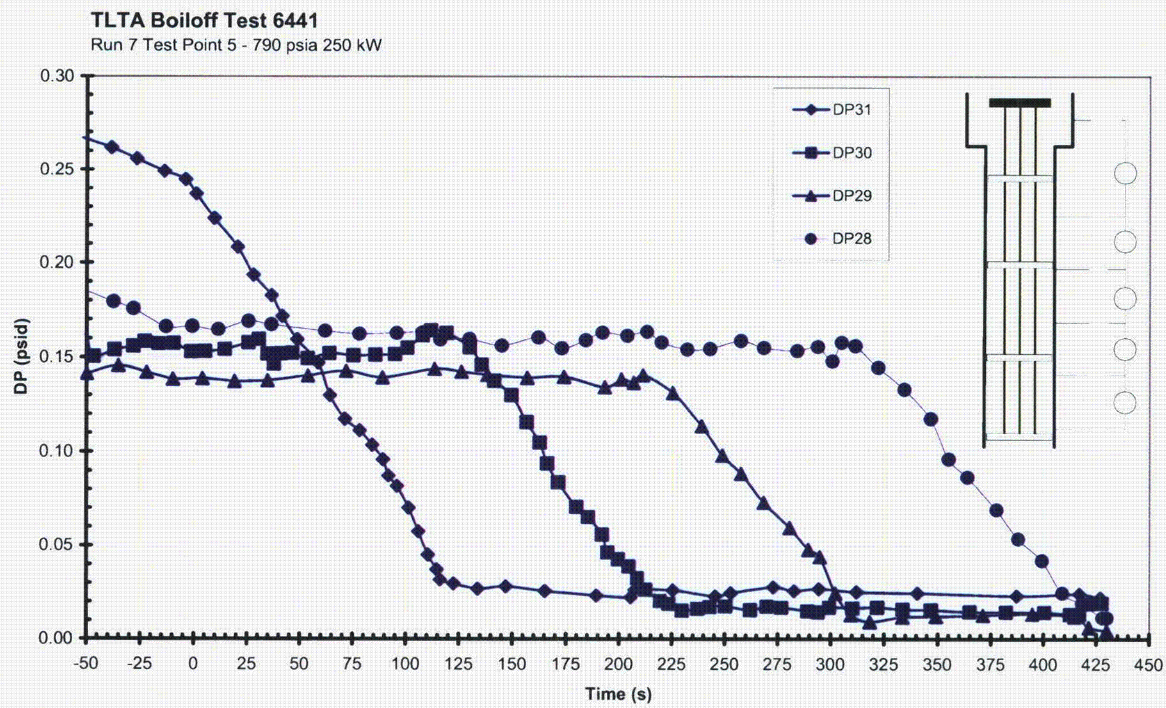


**Figure 5-11 TLTA Boil-Off Test: TLTA Differential Pressure Data (790 psia, 250 kW)**

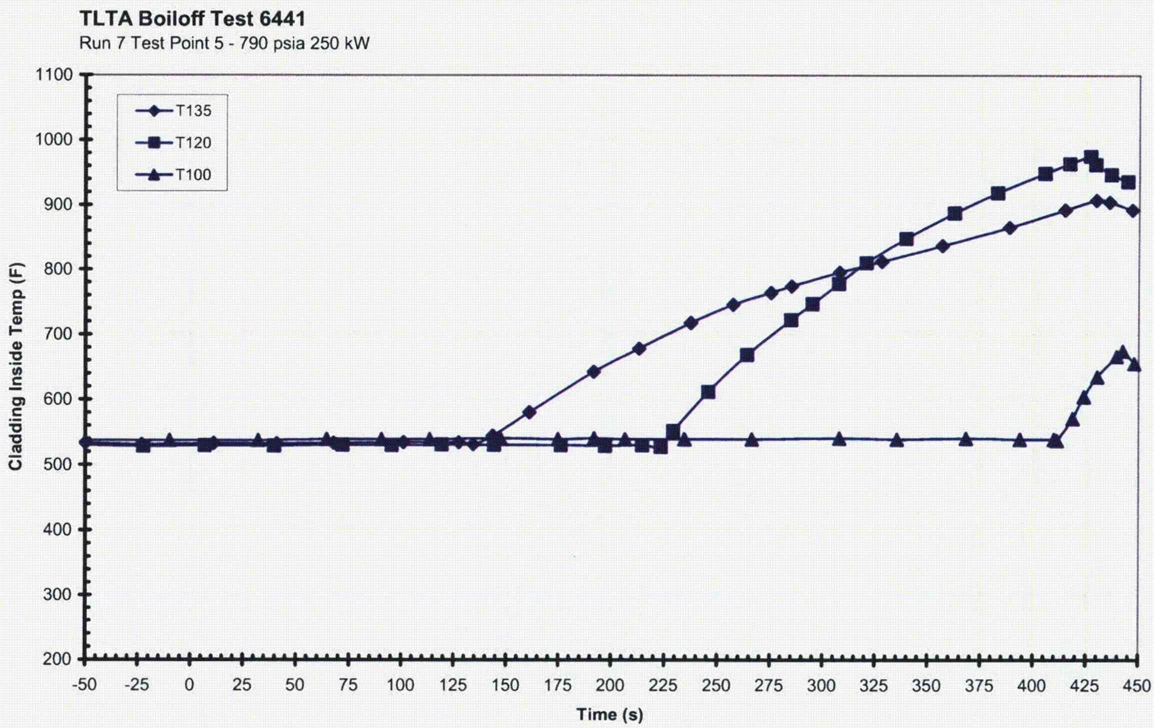


**Figure 5-12 TLTA Boil-Off Test: Cladding Temperature Data (790 psia, 250 kW)**

a,c

**Figure 5-13 TLTA Boil-Off Test: Two-Phase Mixture Heights (790 psia, 250 kW)**

a,c

**Figure 5-14 TLTA Boil-Off Test: Predicted Nodal Void Fractions (790 psia, 250 kW)**

a,c

**Figure 5-15 TLTA Boil-Off Test: Measured/Predicted Cladding Temp. (790 psia, 250 kW)**

a,c

**Figure 5-16 Small GOBLIN Model Used to Test the Local Heat Transfer Coefficient**

a,c

**Figure 5-17 TLTA Boil-Off Test: Effect of HTC Multiplier at 135 Inches (790 psia)**

a,c

**Figure 5-18 TLTA Boil-Off Tests: Predicted vs. Measured Cladding Temp. [ ]a,c**

## 5.5 REFERENCES FOR SECTION 5


5-1 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Report RPB 90-93-P-A (Proprietary), RPB 90-91-NP-A (Non-Proprietary), October 1991.

5-2 Dittus, F. W., and L. M. K. Boelter, Pub. Eng. 2, University of California, Berkeley, 1930.

5-3 Seider, E. N., and G. E. Tate, "Heat Transfer and Pressure Drop of Liquids in Tubes," Ind. Eng. Chem. 28, 1936.

5-4 Jakob, M., "Heat Transfer," Vol. 1, Wiley, 1949.

5-5 "BWR Large Break Simulation Tests – BWR Blowdown/Emergency Core Cooling Program, Report GEAP-24962, Vol. 1 and 2, March 1981.

5-6 "BWR Low-Flow Bundle Uncovery Test and Analysis," Report GEAP-24964, June 1982.

5-7 "Westinghouse Containment Analysis Methodology," Westinghouse Report WCAP-16608-P (Proprietary), WCAP-16608-NP (Non-Proprietary), August 2006.

# 6 SAMPLE APPLICATION OF THE MODEL

A representative BWR/3 model was used to quantify the impact of the containment backpressure and the post-dryout heat transfer methodology changes on the LOCA ECCS analyses. The results of this example application and sensitivity studies are presented in the following sections.

## 6.1 CONTAINMENT BACKPRESSURE BOUNDARY CONDITIONS

This section discusses the revised LOCA analysis utilizing the conservatively calculated containment backpressure.

### 6.1.1 Original GOBLIN Model and Low Mass and Energy Bias

The GOBLIN model used in this calculation is based on a BWR/3 model from a recent LOCA analysis (Reference 6-1). The differences include plant-related input and the main update, and a two-channel model instead of a separate system model and hot channel model (i.e., DRAGON).

The limiting break is double-ended guillotine break of the recirculation line at the pump suction side. The single failure assumption is failure of the LPCI injection valve. In this case, only the ECCS flow from two LPCS pumps is credited. As shown in Figure 6.1-2, the reactor vessel blowdown is over by [ ][a,c] As shown in Figure 6.1-5, ECCS water from LPCS pumps begins to flow into the reactor vessel at [ ][a,c] As shown in Figure 6.1-6, the total mass in the reactor coolant system begins to recover shortly afterward since the mass flow rate of the injected water exceeds the mass flow rate lost to the break.

The mass and energy (M&E) releases from GOBLIN results are used in the GOTHIC calculation of the minimum containment backpressure. The limiting PCT model is biased to ensure that the minimum containment pressure in case of a design basis accident (DBA) LOCA can be conservatively predicted.

As described by the process given in WCAP-16608-P (Reference 6-2), the following modifications are made to the GOBLIN BWR ECCS evaluation model to calculate conservative break mass and energy release input data for the GOTHIC containment analyses for the minimum ECCS backpressure analyses:

1. [

2.

][a,c]

3. [

]$^{a,c}$

### 6.1.2 Original GOTHIC Model and Low Containment Backpressure Bias

The original GOTHIC model is the same as the final model used in the Westinghouse Containment Model Methodology (Reference 6-2).

[

]$^{a,c}$

[

]$^{a,c}$

### 6.1.3 Mass & Energy and Containment Biasing Results

The break M&E output data from the biased M&E release calculation were input to the biased containment model. [ ]$^{a,c}$ The large blowdown M&E release causes the drywell pressure to increase rapidly to [ ]$^{a,c}$. After the vent path clears, the drywell pressure decreases and remains at [ ]$^{a,c}$. The injection of the cooler ECCS water into the reactor vessel condenses steam and reduces break flow to a point where it completely stops since a constant backpressure is assumed. The containment drywell pressure decreases and remains at [ ]$^{a,c}$.

### 6.1.4 GOBLIN Results

Figures 6.1-2 through 6.1-8 compare the system response for a case assuming atmospheric containment backpressure (Case 1) to a case assuming a conservatively calculated backpressure (Case 2). Figure 6.1-2 compares the conservatively calculated drywell pressure boundary condition. [

]a,c

The major impact of increased containment backpressure is [

]a,c

As shown in the results, the impact of the containment backpressure [

]a,c

## 6.2 POST-DRYOUT HTC DURING STEAM COOLING

This section discusses the revised LOCA analysis utilizing the revised heat transfer coefficient (HTC) during post-dryout steam cooling.

### 6.2.1 Base GOBLIN Model and Input Modification

The base GOBLIN model used for the HTC multiplier sensitivity is the model described in Section 6.1 herein, containing the conservatively calculated containment backpressure.

[

]a,c

### 6.2.2 PCT Sensitivity Results

The results of the HTC multiplier sensitivity runs for a constant bundle peaking factor (FRAD) of 1.80 are presented in this section. These results quantify the impact of the convective HTC multiplier on PCT (which are applied using the containment backpressure improvement case as a base case). Table 6.2-1 shows that the effect of changing the HTC from [

]a,c

Figure 6.2-1 illustrates the PCT impact as a result of the differing convective heat transfer modeled for the base case and the HTC multiplier cases. As expected, the temperatures begin to deviate around the time that the HTC begins to differ (Figure 6.2-2). Figure 6.2-2 shows that the HTC multipliers have been applied as intended.

### 6.2.3 FRAD Sensitivity Results

The results of a sensitivity study performed for [ ]$^{a,c}$ are summarized in this section. Both the base case and the sensitivity run employ non-atmospheric containment backpressure, as discussed in earlier sections. Table 6.2-2 shows that updating from an [

]$^{a,c}$

Figure 6.2-3 illustrates that the CHACHA calculations were performed for a prescribed PCT and that the temperatures between the base case and the HTC multiplier cases, as expected, begin to deviate around the time that the HTC begins to differ (Figure 6.2-4). Figure 6.2-4 shows that the HTC multipliers have been applied as intended.

## 6.3 TABLES FOR SECTION 6

[ ]$^{a,c}$

## 6.4 FIGURES FOR SECTION 6

a,c

**Figure 6.1-1 Minimum Containment Back Pressure vs. Time**

a,c

**Figure 6.1-2 Comparison of Drywell Pressure Boundary Condition**

a,c

**Figure 6.1-3 Comparison of Steam Dome Pressure**

a,c

**Figure 6.1-4 Comparison of Steam Dome to Drywell Pressure Difference**

a,c

**Figure 6.1-5 Comparison of Break Mass Flow Rate**

a,c

**Figure 6.1-6 Comparison of Core Spray Flow Rate Delivered**

a,c

Figure 6.1-7 Comparison of Reactor Coolant System Total Mass

a,c

**Figure 6.1-8 Comparison of Cladding Heatup Rate Targeting 2150°F (1176°C) PCT**

a,c

**Figure 6.2-1 PCT Comparison for HTC Multiplier Sensitivity Runs (FRAD = 1.80)**

a,c

**Figure 6.2-2 HTC Comparison for HTC Multiplier Sensitivity Runs (FRAD = 1.80)**

a,c

**Figure 6.2-3 PCT Comparison for HTC Multiplier Sensitivity Runs (PCT = 1176°C = 2150°F)**

a,c

**Figure 6.2-4 HTC Comparison for HTC Multiplier Sensitivity Runs (PCT = 1176°C = 2150°F)**

## 6.5 REFERENCES FOR SECTION 6


6-1 "Task Report for TSD DQW04-21 LOCA Analysis for Quad Cities 1 & 2 and Dresden 2 & 3," Westinghouse Report NF-BEX-06-8-P, Revision 1 (Proprietary), NF-BEX-06-8-NP, Revision 1 (Non-Proprietary), February 2006.

Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Additional Information Supporting Request for License Amendment Regarding Transition to Westinghouse Fuel," Exelon Letter RS-06-033, February 22, 2006.

Letter from M. Banerjee (NRC) to C. M. Crane (Exelon), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Issuance of Amendments Re: Transition to Westinghouse Fuel and Minimum Critical Power Ratio Limits (TAC NOS. MC7323, MC7324, MC7325 and MC7326)," April 4, 2006.

6-2 "Westinghouse Containment Analysis Methodology," Westinghouse Report WCAP-16608-P (Proprietary), WCAP-16608-NP (Non-Proprietary), August 2006.

# 7 COMPLIANCE WITH 10 CFR 50 APP. K

This section provides excerpts summarizing how the Westinghouse BWR ECCS Evaluation Model complies with the requirements of 10 CFR 50 Appendix K.

## 7.1 SOURCES OF HEAT DURING THE LOCA

*Adapt from WCAP-16078 (Reference 7-1) - From WCAP-16078:*

Section I.A of Appendix K reads as follows:

> *"For the heat sources listed in paragraphs I.A.1 to 4 of this appendix it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks* and *single failures that are analyzed."*

Westinghouse Evaluation Model Compliance with Section I.A:

> Section I.A of Appendix K has changed relative to Westinghouse's statement of compliance in References 7-2 and 7-3 in that an assumed power level less than 1.02 times the licensed power level may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. Westinghouse may account for power level instrumentation uncertainties less than 2 percent, but no less than the power level uncertainty that has been demonstrated.

## 7.2 INITIAL STORED ENERGY IN THE FUEL

*From WCAP-16078 (Reference 7-1):*

Section I.A.1 of Appendix K reads as follows:

> *"The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep."*

Westinghouse Evaluation Model compliance with Section I.A.1:

The revised model for the [

]$^{a,c}$

## 7.3 FISSION HEAT

*From RPB 90-93 (Reference 7-4):*

Section I.A.2 of Appendix K reads as follows:

*"Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur."*

Westinghouse Evaluation Model Compliance with Section I.A.2:

The fission heat is calculated by a reactor point kinetics model as described in Section 3.7 (of Reference 7-4). The model includes feedback effects from voiding, Doppler broadening, moderator temperature, and control rod worth. The point kinetics parameters shall be generated from the nuclear design code PHOENIX for a range of power distributions, peaking factors, and void fractions throughout the fuel life. Conservative values for these parameters, i.e., those which yield the highest fission heat generation, shall be used in the GOBLIN model. Specifically:

- The delayed neutron fraction ($\beta$) will be given its highest calculated value, typically corresponding to beginning-of-life conditions.
- The void and Doppler reactivity coefficients will be given their highest calculated value (lowest absolute value).
- The reactivity worth of the control rods will be given a conservative (low) value.
- The reactor scram will be assumed to occur at a conservative (late) time in the LOCA transient.

The methodology and sensitivity studies demonstrating the conservatism of the fission heat generation are provided in the topical report WCAP-11427 (Reference 7-5).

The average fission and decay heat generation curve calculated by the GOBLIN code is used with appropriate peaking factors in the DRAGON hot channel analysis and CHACHA-3C peak axial plane heatup analysis.

## 7.4 DECAY OF ACTINIDES

*From RPB 90-93 (Reference 7-4):*

Section I.A.3 of Appendix K reads as follows:

> *"The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA."*

Westinghouse Evaluation Model Compliance with Section I.A.3:

> The heavy-element (i.e., actinides) decay-energy release contribution is determined by calculating the equilibrium concentrations of the isotopes $U^{239}$ and $Np^{239}$, and then using the energy per disintegration and half-life for these isotopes to evaluate the time dependence of the energy release after shutdown. The elements $U^{239}$ and $Np^{239}$ are the only significant activation products that contribute to the decay energy release in the time range of interest for LOCAs. The energy release from the activation products of $U^{235}$ (namely $U^{236}$ and $U^{237}$) are insignificant, approximately a factor of 20-30 less, when compared to the energy release of $U^{239}$ and $NP^{239}$ for this time range.
>
> The actinide decay power is determined from the decay rate equations as described in the American Nuclear Society Standard 5.1 (Reference 7-6) and is modeled in GOBLIN as the decay power groups 12, 13, and 14 (see Section 3.7.1 of Reference 7-4).
>
> The $U^{239}$ production per fission, $C_r\sigma_{25}/\sigma f_{25}$, is chosen to yield the highest actinide decay power throughout the fuel life, typically end of life.

## 7.5 FISSION PRODUCT DECAY

*From RPB 90-93 (Reference 7-4):*

Section I.A.4 of Appendix K reads as follows:

> *"The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards--"Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors." Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971). This standard has been approved for incorporation by reference by the Director of the Federal Register. A copy of the standard is available for inspection at the NRC Library,*

*11545 Rockville Pike, Rockville, Maryland 20852-2738. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation."*

Westinghouse Evaluation Model Compliance with Section I.A.4:

Decay of $U^{235}$ fission products is computed by a relationship in the form of the summation of eleven decay equations. The fission product decay model is described in Section 3.7.1 of Reference 7-4. Comparison with the tabulated 1971 ANS proposed standard (Reference 7-7) is shown in Table 3-3 of Reference 7-4. The agreement is excellent. The local decay heat power calculated by this model is multiplied by 12 in accordance with the requirement in paragraph I.A.4. The fraction of gamma energy deposited in the fuel along with gamma and neutron and gamma deposition in the coolant may be specified with time through the transient. The actual deposition fractions are described and justified in the topical report WCAP-11427 (Reference 7-5).

## 7.6 METAL-WATER REACTION RATE

*From RPB 90-93 (Reference 7-4):*

Section I.A.5 of Appendix K reads as follows:

*"The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L.C., "Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, page 7, May 1962). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of the publication is available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852-2738. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less that 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited."*

Westinghouse Evaluation Model Compliance with Section I.A.5:

The requirement to account for the effects of local metal-water reaction as described in Section I.A.5 is met. This additional heat source is considered in the cladding temperature calculation as described in Section 3.7.2 (Reference 7-4) for GOBLIN/DRAGON and Section 4.4 (Reference 7-4) for CHACHA-3C. In these models, the reaction between Zircaloy cladding and steam is assumed to follow the parabolic rate law of Baker and Just as specified by the criterion.

## 7.7 REACTOR INTERNALS HEAT TRANSFER

*From RPB 90-93 (Reference 7-4):*

Section I.A.6 of Appendix K reads as follows:

> *"Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account."*

Westinghouse Evaluation Model Compliance with Section I.A.6:

> Heat transfer from piping, vessel walls, and non-fuel internal hardware is accounted for, according to the methods described in Sections 3.5 and 3.6 of Reference 7-4.

## 7.8 SWELLING AND RUPTURE OF THE CLADDING AND FUEL ROD THERMAL PARAMETERS

*From WCAP-15682 (Reference 7-8):*

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."*

Westinghouse Evaluation Model Compliance with Section I.B:

> Section 6.2 of CENPD-293-P-A (Reference 7-3) describes the comparison of the mechanistic swelling and rupture model to the applicable set of data. Section 4.1 of WCAP-15682-P-A (Reference 7-8) 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.

## 7.9 BREAK SPECTRUM ANALYSIS

*From RPB 90-93 (Reference 7-4):*

Section I.C.1.a of Appendix K reads:

> *"In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe."*

Westinghouse Evaluation Model Compliance with Section I.C.1.a:

> The LOCA sensitivity study topical report (Reference 7-5) reports the results of a break spectrum analysis including the double-ended guillotine break of the largest pipe in a typical boiling water reactor design for which this model is employed. The study will also include various break locations and will be used to justify the selection of the worst case in a plant-specific LOCA analysis.

## 7.10 DISCHARGE MODEL

*From RPB 90-93 (Reference 7-4):*

Section I.C.1.b of Appendix K reads:

> *"For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F.J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of this publication is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852-2738. The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperatures calculated by this variation has been achieved."*

Westinghouse Evaluation Model Compliance with Section I.C.1.b:

> The Moody model is used to calculate the two-phase discharge rate. The application and integration of the Moody model into the complete break flow model for all regimes is described in Section 3.3.6 of Reference 7-4.

The results of a study showing PCT sensitivity to break area are presented in the Evaluation Model sensitivity topical report (Reference 7-5). The sensitivity study results will be used to justify the worst-case break flow area and discharge coefficients used in plant-specific evaluation model LOCA analyses.

## 7.11 CRITICAL HEAT FLUX

*From WCAP-16078 (Reference 7-1):*

Section I.C.4 of Appendix K reads:

"a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *W 3*. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution," *Journal of Nuclear Energy*, Vol. 21, 241-248, 1967.

(2) *B&W-2*. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," *Two-Phase Flow and Heat Transfer in Rod Bundles*, ASME, New York, 1969.

(3) *Hench-Levy*. J. M. Healzer, J. E. Hench, E. Janssen, S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, GE Company Private report, July 1966.

(4) *Macbeth*. R. V. Macbeth, "An Appraisal of Forced Convection Burnout Data," *Proceedings of the Institute of Mechanical Engineers*, 1965-1966.

(5) *Barnett*. P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R 463, 1966.

(6) *Hughes*. E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia," IN-1412, Idaho Nuclear Corporation, July 1970."

Westinghouse Evaluation Model Compliance with Section I.C.4:

The critical heat flux in the system and hot assembly analyses is determined using an NRC-approved CPR correlation that is applicable to the fuel design. For SVEA-96 Optima2 fuel, that correlation was submitted to NRC for approval prior to its use for ECCS performance analyses. The process of implementing the CPR correlation in the GOBLIN code is described in WCAP-16078 (Reference 7-1).

## 7.12 POST-CHF HEAT TRANSFER CORRELATIONS

*From RPB 90-93 (Reference 7-4):*

Section I.C.5 of Appendix K reads:

*"a. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.*

*b. The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime, "AECL-3281, revised December (1969), the Dougall-Rohsenow flow film boiling correlation (R.S. Dougall and W.M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities, "MIT Report Number 9079-26, Cambridge, Massachusetts, September 1963), and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation, "U.S.A.E.C. Docket RM-50-1, page 25.1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes.*

*In addition the transition boiling correlation of McDonough, Mi l ich, and King (J.B. McDonough, W. Milich, E.C. King, "Partial Film Boiling with Water at 2000 psig in a Round Vertical Tube" MSA Research Corp., Technical Report 62 (NP-6976), (1958) is suitable for use between nucleate and film boiling. Use of all of these correlations shall be restricted as follows:*

*(1) The Groeneveld correlation shall not be used in the region near its low-pressure singularity,*

*(2) the first term (nucleate) of the Westinghouse correlation and the entire McDonough, Mi l ich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300°F,*

*(3) transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300°F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions."*

Westinghouse Evaluation Model Compliance with Section I.C.5:

The convective heat transfer correlations and regimes modeled in GOBLIN are described in detail in Section 3.6. The post-critical heat flux (dryout) convective heat transfer coefficient is calculated using the Groeneveld 5.7 correlation, NRC-approved Westinghouse upper-head injection (UHI) correlation, modified Bromley correlation, and single-phase steam correlations. The Groeneveld correlation is used for flow film boiling in the higher pressure range. For lower pressures, where the Groeneveld correlation has a singularity, a transition is made to the Westinghouse UHI correlation. This NRC-approved correlation is more conservative than the Doughall-Rohsenow correlation, which is nonconservative when compared against some heat transfer data.

The lower limit to the heat transfer coefficient is calculated using the modified Bromley correlation, which is based on zero flow. The modified Bromley correlation has been demonstrated to be a conservative lower limit when compared against a wide range of tests. A more detailed discussion of the applicability of this correlation is given in the qualification section (see Section 6.1.7).

Once dryout is calculated to occur, the heat transfer is conservatively forced to remain in the post-dryout regime, even if rewet and transition boiling are calculated to occur.

## 7.13 PUMP MODELING

*From RPB 90-93 (Reference 7-4):*

Section I.C.6 of Appendix K reads:

*"The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump head may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent."*

Westinghouse Evaluation Model Compliance with Section I.C.6:

The recirculation pump model is described in Section 3.4.1 of Reference 7-5. An angular momentum balance is solved for the pump, including all contributing torques. Single-phase and degraded two-phase pump performance is modeled through user-specified performance curves. The justification of specific pump resistances, single-phase, and two-phase performance curves shall be addressed in a future topical report.

The two-phase performance degradation to zero pump head, once the pump suction has a quality of one, can be specified by the user through the two-phase performance curves. However, it is expected that for the jet pump BWRs, the pump performance will degrade to zero before saturation conditions reach the pump suction, due to the draining of the downcomer and uncovery of the jet pump suction. The degradation of pump performance will be addressed on a plant-specific basis.

"(Appendix K Section I.C.7 – Not applicable to BWR)"

## 7.14 SINGLE FAILURE CRITERION

*From RPB 90-93 (Reference 7-4):*

Section I.D.1 of Appendix K reads:

*"An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place."*

Westinghouse Evaluation Model Compliance with Section I.D.1:

The evaluation of the loss-of-coolant accident is performed assuming the single active component failure that results in the most severe consequences. The combination of ECC subsystems assumed to be operating are those remaining after the component failure has occurred.

The topical report WCAP-11427 (Reference 7-5) includes results showing LOCA peak clad temperature sensitivity to various single failure assumptions. Previous evaluations by the NSSS vendor identifying the worst single failure in the ECC systems will also be reviewed to determine the limiting component failure assumed in a plant-specific LOCA analysis.

## 7.15 CONTAINMENT PRESSURE

*From RPB 90-93 (Reference 7-4):*

Section I.D.2 of Appendix K reads:

*"The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes."*

Westinghouse Evaluation Model Compliance with Section I.D.2:

The GOBLIN analyses for LBLOCA will use a pressure calculated conservatively by GOTHIC 7.2a as described in WCAP-16608-P. For SBLOCA, the GOBLIN analyses will assume atmospheric pressure in the containment volume throughout the LOCA transient. This assumption adequately addresses the requirements for this feature of Appendix K."

## 7.16 CONVECTIVE HEAT TRANSFER COEFFICIENTS FOR BWR FUEL RODS UNDER SPRAY COOLING

*From RPB 90-93 (Reference 7-4):*

Section I.D.6 of Appendix K reads:

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

Westinghouse Evaluation Model Compliance with Section I.D.6 (revised in this report):

The rod surface heat transfer coefficients used by CHACHA-3C for the period prior to the end of lower plenum flashing are those calculated by GOBLIN/DRAGON using the models described in Section 3.5 of Reference 7-4. After this period, the GOBLIN/DRAGON values are replaced as follows:

(1) For the period following lower plenum flashing but prior to the core spray reaching rated flow, [

]$^{a,c}$

(2) For the period after core spray reaches rated flow but prior to reflooding, the spray cooling convective heat transfer coefficients in Table 4.3 of Reference 7-4 will be used. These heat transfer coefficients are derived from the Appendix K recommended values. Experimental data will be used to verify their applicability (Section 6.1.12 of Reference 7-4). The time needed for the ECCS to reach the rated spray flow rate is calculated in GOBLIN/DRAGON.

(3) For the period after the two-phase reflooding fluid reaches the core elevation of interest, the reflood heat transfer coefficient specified in Appendix K will be used. The time of reflooding the core elevation of interest will be based on the local void fraction.

## 7.17 THE BOILING WATER REACTOR CHANNEL BOX UNDER SPRAY COOLING

*From RPB 90-93 (Reference 7-4):*

Section I.D.7 of Appendix K reads:

> *"Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7 x 7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.*
>
> *a. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.*
>
> *b. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 Btu-hr$^{-1}$-ft$^{-2}$-°F$^{-1}$ shall be applied to both sides of the channel box.*
>
> *c. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis ("Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Company Report NEDO-10329, April 1971). This report was approved for incorporation by reference by the Director of the Federal Register. A copy of the report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852-2738."*

Westinghouse Evaluation Model Compliance with Section I.D.7 (revised in current report):

Calculation of the channel wall temperature is described in Section 4.2 of Reference 7-4. The following channel wall heat transfer coefficients and rewet model are used:

- For the period prior to the end of lower plenum flashing, the convective heat transfer coefficients calculated by GOBLIN/DRAGON will be used (Section 3.5 of Reference 7-4).
- For the period after lower plenum flashing but prior to core spray reaching rated flow, the [ ]$^{a,c}$
- For the period after core spray reaches rated flow but prior to wetting of the channel, the convective heat transfer coefficient in Table 4-3 of Reference 7-4 will be applied to both sides of the channel. This heat transfer coefficient is derived from the Appendix K recommended value. Experimental data will be used to verify its applicability (Section 6.1.12 of Reference 7-4).
- The channel wetting time will be determined based on the modified Yamanouchi correlation plus 60 seconds, as described in Section 4.7 of Reference 7-4.

## 7.18 REFERENCES FOR SECTION 7


7-1 "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," Westinghouse Report WCAP-16078-P-A (Proprietary), WCAP-16078-NP-A (Non-Proprietary), November 2004.

7-2 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," Westinghouse Report CENPD-283-P-A (Proprietary), CENPD-283-NP-A (Non-Proprietary), July 1996.

7-3 "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," Westinghouse Report CENPD-293-P-A (Proprietary), CENPD-293-NP-A (Non-Proprietary), July 1996.

7-4 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Report RPB 90-93-P-A (Proprietary), RPB 90-91-NP-A (Non-Proprietary), October 1991.

7-5 "Westinghouse Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," Westinghouse Report WCAP-11427-P-A, October 1989.

7-6 Proposed American Nuclear Society Standard 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," October 1971, revised October 1973.

7-7 Code of Federal Regulations, 10 Part 50, Office of the Federal Register, National Archives and Records Administration, 1986.

7-8 "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," Westinghouse Report WCAP-15682-P-A (Proprietary), WCAP-15682-NP-A (Non-Proprietary), April 2003.

# APPENDIX A
# ROADMAP TO THE METHODOLOGY CHANGES

## A.1 INTRODUCTION

The original BWR LOCA Evaluation Model (USA1), which was approved by the NRC in 1989, is described in RPB 90-93-P-A (Reference A-1) and RPB 90-94-P-A (Reference A-2). This methodology was revised in 1996 with the USA2 Evaluation Model, described in CENPD-283-P-A (Reference A-3) and CENPD-293-P-A (Reference A-4); in 2003 with the USA4 Evaluation Model, described in WCAP-15682-P-A (Reference A-5); and in 2004 with the USA5 Evaluation Model, described in WCAP-16078-P-A (Reference A-6). The USA6 Evaluation Model is described in this topical report.

Reference A-1 provides a detailed description of the BWR ECCS Evaluation Model. Reference A-2 describes the application of the evaluation model to reactor plant system analyses and sensitivities to plant parameters. References A-3 through A-6 describe changes to the evaluation model as well as applications of the model to different fuel designs.

Table A-1 provides a roadmap to changes to various elements of the BWR ECCS Evaluation Model. The changes to each element are cross-referenced to a brief description of the changes in the following sections and to the appropriate section of the reference cited.

## A.2 MAJOR ASPECTS OF THE EVALUATION METHODOLOGY

### A.2.1 Momentum Equation

As described in Reference A-4, Section 4.2:

> The spatial acceleration term in the momentum equation has been modified to account more accurately for uneven velocities of water and steam. This change has insignificant effects on typical LOCA transients and has been introduced to improve the consistency of the fluid flow model as shown in Section 7.4 of Reference A-4. The new formulation was assessed and qualified by repeating pertinent cases in the GOBLIN qualification.

Per the NRC SER of Reference A-4:

> The Technical Evaluation attached to the SER acknowledged the change to the momentum equation and cited a study showing that the GOBLIN code with the new formulation resulted in an early prediction of dryout time and in a slightly higher temperature than in old analysis. No limitations or conditions were placed upon the use of this modification.

### A.2.2 Countercurrent Flow Limitation (CCFL) Model

As described in Reference A-3, Section 7.1:

> The BWR LOCA evaluation model has a comprehensive Countercurrent Flow Limitation (CCFL) model for determining the rate of liquid drainage into the SVEA-96 fuel assembly. The correlation is documented in Reference A-1, Section 3.3. The CCFL correlation was originally developed for 8 x 8 fuel assemblies. Since its original development, the correlation has been generalized and validated for many geometries. Further, the correlation, with its general geometric dependence, has been confirmed valid for QUAD+ fuel through comparisons with experimental data (see response to Question 8 in Reference A-1). The SVEA-96 geometry is basically the same as the SVEA-64 and QUAD+ geometry. Differences in area of the flow restrictions are accounted for in the CCFL correlation. In the LOCA evaluation model, the CCFL correlation with the appropriate geometric parameters for SVEA-96 fuel will be used.

Per the NRC SER of Reference A-3:

- The coefficients in the CCFL correlation that were shown to be insensitive for the SVEA-96 fuel should not be extended to other fuels without being validated by experimental data.
- In addition, Westinghouse was required to demonstrate the acceptability the CCFL coefficient in any instance when the calculated PCT is greater than 2100°F. In this case, the CCFL correlation shall include a conservative bias that bounds the scatter in the database. The bias introduced to the base CCFL correlation will be such that conservative bounding predictions are obtained from the database of all fuel assembly components that were used to derive the basic CCFL correlation.

As described in Reference A-6, Section 5.4.2:

> The change to the CCFL model removes the restriction placed on the USA2 Evaluation Model. This change was made to the CCFL correlation to apply a conservative bias such that it bounds all scatter in the correlation database.
>
> The present CCFL correlation replaces the wetted perimeter term in one of the correlation coefficients with one composed of an effective diameter relation that is a function of the local cross-sectional flow area. The more restrictive effective diameter relation better represents the observed data and eliminates the restriction placed on earlier versions of the Evaluation Model.
>
> To qualify the applicability of the modified CCFL model to SVEA-96 Optima2 fuel, Westinghouse performed a sensitivity study demonstrating the effect of the new fuel design and the modified CCFL correlation on the overall LOCA response.

Per the NRC SER of Reference A-6:

- The staff concurred that the change to the CCFL acts in the same manner as the imposed restriction required by the SER of Reference A-3 and found that the CCFL model with appropriate geometric parameters is acceptable for applications involving SVEA-96, SVEA-96+, and SVEA-96 Optima2 fuel designs.

### A.2.3 Two-Phase Level Tracking

As described in Reference A-3, Section 6.1.2:

> As a result of a sensitivity study on the use of level tracking in the upper plenum, it was determined that it is conservative to deactivate the level tracking option and that the additional accuracy of tracking the upper plenum level is not warranted.

Per the NRC SER of Reference A-3:

> The TER attached to the SER acknowledged the level tracking sensitivity study. The SER made no limitations or conditions regarding the use of level tracking in the upper plenum.

As described in Reference A-6, Section 5.1.1:

> The level tracking model calculates the motion of the control volume interface such that it moves with the two-phase mixture level. The intent of the level tracking model is to capture the transient interaction of the mixture level with flow paths or with ECCS injection when it is impractical to do so by additional noding detail. Sensitivity studies were presented on the level tracking feature in Section 4.1.3 of RPB 90-94-P-A (Reference A-2) and Section 6.1.2 of CENPD-283-P-A (Reference A-3). The focus of these sensitivity studies was the use of level tracking in the upper plenum. Section 6.1.2 of CENPD-283-P-A (Reference A-3) provided the basis for not using the level tracking feature in the upper plenum.
>
> The focus of the sensitivity study presented in this section is on the use of level tracking in the lower plenum. Cases were run with the level tracking feature activated and deactivated in the lower plenum. The results were virtually identical .... Since level tracking in the lower plenum does not affect the timing of these key events, the heat transfer coefficients that are used to determine the cladding temperature response of the hot plane will be identical. As a result, the cladding temperature response will also be identical. This sensitivity shows that the use of the level tracking feature in the lower plenum is not warranted. Therefore, standard practice will be to not use level tracking in the lower plenum of the USA5 Evaluation Model unless warranted by the specific application.
>
> However, level tracking remains an option to capture important thermal-hydraulic phenomena when it is not practical to do so with fixed control volumes. Level tracking continues to be used in the reactor vessel annulus to ensure that conditions upstream of the break are determined correctly.

Per the NRC SER of Reference A-6:

> The SER indicated that the use of the optional level tracking model in the lower plenum of the GOBLIN vessel model is acceptable.

### A.2.4 Convective Spray Heat Transfer Coefficients

As described in Reference A-3, Section 4.3:

> Convective spray heat transfer coefficients are specified in 10 CFR 50 Appendix K applicable for 7 x 7 fuel designs. Convective heat transfer coefficients have been derived for open lattice 8 x 8, SVEA-64/QUAD+ fuel from the coefficients prescribed in 10 CFR 50 Appendix K. These coefficients also were confirmed by experimental tests for 8 x 8 and SVEA-64/QUAD+. An extension of this application provides spray heat transfer coefficients for SVEA-96 fuel.
>
> The approved values of convective heat transfer coefficients per Reference A-3 are presented in Table A-2.

Per the NRC SER of Reference A-3:

- The SER concurred with the procedure to show conservatism in the method used to determine the spray cooling heat transfer coefficients. However, since the procedure was not supported by experimental data, it should not be extended to other fuels without experimental verification.

As described in Reference A-6, Section 6.1.1:

> The convective spray heat transfer coefficients described in Section 7.2 of CENPD-283-P-A (Reference A-3) are applied without modification to analyses determining the hot plane heatup response for a reactor containing SVEA-96 Optima2 fuel. The spray cooling heat transfer coefficients are given in Table A-3.

Per the NRC SER of Reference A-6:

- Because of the similarity of the lattice layout to the SVEA-96/96+ fuel design, the staff found applying SVEA-64 spray coefficients to the Optima2 fuel to be conservative and acceptable.

### A.2.5 Critical Power Ratio (CPR) Correlation

As described in Reference A-3, Section 4.2:

> The SVEA-96 CPR correlation was developed through a full-scale thermal-hydraulic verification program in the ABB Atom FRIGG loop. The resultant correlation is documented in Reference A-7, which has been approved by the U.S. NRC. This correlation, denoted by XL-S96, is implemented into the GOBLIN/DRAGON code. The implementation is analogous to the previous approved QUAD+ CPR correlation application.

Per the NRC SER of Reference A-3:

- The SER (Reference A-8) on UR-89-210-P-A (Reference A-7) approved the use of the XL-S96 CPR correlation with the BISON computer code. However, Reference A-8 requires that when this correlation is implemented in other computer codes, the vendor must submit documentation of adequate implementation to the NRC. The SER also requires that the correlation be used to evaluate the SVEA-96 fuel assemblies for the revised range of applicability.

- The adequacy of implementation of the XL-S96 CPR correlation into the GOBLIN series will be reviewed with CENPD-293-P (Reference A-4), since this version of GOBLIN/DRAGON/CHACHA-3C was viewed to be an intermediary state.

As described in Reference A-4, Section 4.1:

> A critical power ratio (CPR) correlation using the "critical quality-boiling length" formulation has been introduced in the thermal-hydraulic code system GOBLIN/DRAGON.
>
> The GEXL correlation (also described in Reference A-7) is chosen as the basis for all "critical quality-boiling length" type CPR correlations. The implementation of this base correlation is described in Section 4.1.2 of Reference A-4. As a sample case of the critical quality-boiling length CPR correlation, the implementation of the XL-S96 correlation is described in Section 4.1.3 of Reference A-4, and verification of proper implementation of the correlation is given in Section 7.1 of Reference A-4.

Per the NRC SER of Reference A-4:

> The SER placed a condition on the use of the XL-S96 CPR correlation requiring that it be subject to the SER conditions in UR 89-210-P-A (Reference A-7) and Reference A-8.

As described in Reference A-6, Section 5.4.1:

> CPR correlations are part of the heat transfer model in GOBLIN and DRAGON. The CPR correlation is used to determine the initial power of the hot assembly. The CPR correlation may also determine when boiling transition occurs during the LOCA transient if the fluid conditions are within the range of applicability of the correlation. The Westinghouse USA5 BWR ECCS Evaluation Model will use the SVEA-96 Optima2 CPR correlation that is approved by the NRC for applications involving the SVEA-96 Optima2 fuel design.
>
> An NRC-approved CPR correlation that is applicable to the fuel-design being analyzed is used in the ECCS Evaluation Model (RPB 90-93-P-A [Reference A-1]). At the time this topical report was written, the CPR correlation for SVEA-96 Optima2 had not been approved by NRC. Qualification of the SVEA-96 Optima2 CPR correlation was provided subsequently. The CPR correlation was installed in the GOBLIN code in accordance with the process described in Section 3.3.2.2 of Reference A-6 and has been used for licensing. The NRC will be informed of the resulting change to the GOBLIN code via the 10 CFR 50.46 reporting process.

Per the NRC SER of Reference A-6:

- The SER indicated that the SVEA-96 Optima2 CPR correlation was being reviewed by the staff. After it is approved, Westinghouse may implement it into the USA5 EM model and report to the NRC through the 10 CFR 50.46 annual report process.

- A new fuel design normally requires a specific CPR correlation approved by the NRC. The implementation of a new CPR correlation into GOBLIN has become a routine code update process, which includes the source code development, new CPR correlation validation, and non-Westinghouse fuel justifications. Westinghouse requested that this process be evaluated through the 10 CFR 50.46 annual report process. Therefore, the staff does not necessarily review the details of the implementation process. The staff has previously reviewed the proposed process from Reference 10 and determined that the requested process is acceptable as long as the new CPR correlation has been approved by the NRC and the change to the LOCA method is reported to the NRC through the 10 CFR 50.46 reporting process.

- For version USA5, the currently approved CPR correlations (i.e., XL-S96, ABBD1.0, ABBD2.0) can still be used within the approved ranges of applicability. However, the new CPR correlation for SVEA-96 Optima2 fuel has not yet been approved by the NRC. Therefore, the current version of the SVEA-96 Optima2 fuel CPR correlation in USA5 cannot be used until it has received the approval of the staff.

- On December 9, 2004, the NRC staff issued its safety evaluation (SE) approving Topical Report WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2." As provided in the FSER for NRC license amendments issued on April 4, 2006 for the transition of the Quad Cities and Dresden units to Westinghouse Fuel, the NRC staff verified that all the conditions and limitations of the NRC-approved BWR LOCA methods were satisfied for this application.

## A.2.6 Fuel Rod Conduction Model

As described in Reference A-4, Section 5.1:

> The fuel rod conduction model described in Section 4.1 of Reference A-1 is unchanged. An optional feature is added to explicitly model the heat resistance due to crud on the cladding surface. The effective outside surface heat transfer coefficient is a function of the previous coefficient, the depth of the crud layer, and the thermal conductivity of the crud layer.
>
> The depth of the crud layer is calculated using an NRC-approved fuel rod performance code and the thermal conductivity of crud used is consistent with the fuel performance code properties. For example, applications in the foreseen future shall use a crud depth from the STAV6.2 code and a crud thermal conductivity of 0.5 (W/m/°K), which is also consistent with STAV6.2.

Per the NRC SER of Reference A-4:

> The Technical Evaluation attached to the SER acknowledged the addition of the crud resistance model. The SER placed no limitations or conditions on its use.

As described in Reference A-6, Section 5.5.2.3:

A model has been introduced in the STAV7.2 code to describe the burnup-induced degradation of the fuel pellet conductivity. This model has replaced the STAV6.2 fuel pellet conductivity model in CHACHA-3D.

Per the NRC SER of Reference A-6:

The SER acknowledged the addition of the revised fuel pellet conductivity model but placed a condition upon its use until the NRC approval of the STAV7.2 code was complete. Westinghouse provided a letter to the NRC (Reference A-9), as required by the condition, after NRC approval of the STAV7.2 code was complete. The letter indicated that the applicable STAV7.2 models have been implemented in CHACHA-3D.

### A.2.7 Heat Generation Model

As described in Reference A-4, Section 5.2:

The heat generation model, as described in Section 4.3 of Reference A-4, is unchanged except for the radial power distribution within the fuel pellet, and will be supplied from the appropriate NRC-approved fuel performance code. For example, results from the STAV6.2 code will be used as input to CHACHA-3. In addition, options are available in CHACHA-3 to assume:

- Uniform radial power distribution, or
- Bessel function based radial power distribution

Per the NRC SER of Reference A-4:

The Technical Evaluation attached to the SER acknowledged the use of the uniform radial power distribution model and the Bessel function model in CHACHA. No conditions or limitations were placed upon their use in the SER.

As described in Reference A-6, Section 5.5.2.3:

The burnup-dependent TUBRNP model in STAV7.2 has been implemented in CHACHA-3D and will be used in the USA5 Evaluation Model. This model takes into account power generation by plutonium isotopes, resulting in a more precise radial power distribution in the pellet rim region.

Per the NRC SER of Reference A-6:

The SER acknowledged the change to the pellet heat generation model, but placed a condition upon its use until NRC approval of the STAV7.2 code was complete. Westinghouse provided a letter to the NRC (Reference A-9), as required by the condition, after NRC approval of the STAV7.2 code was complete. The letter indicated that the applicable STAV7.2 models have been implemented in CHACHA-3D.

### A.2.8 Metal-Water Reaction

As described in Reference A-4, Section 5.3:

> The metal-water reaction model remains unchanged from that described in Section 4.4 of Reference A-1. The initial oxide depth on the cladding outer surface is calculated using an NRC-approved fuel performance code. For example, the STAV6.2 code (Reference A-10) will replace the PAD fuel rod performance code, identified in Reference A-2.

Per the NRC SER of Reference A-4:

> The NRC SER indicated that acceptance of this change should be determined in the review of CENPD-285-P (Reference A-10). The NRC SER of Reference A-10 placed no restrictions on the use of the initial oxide depth except for limiting the rod average burnup range of STAV6.2 to 50 GWd/MTU.

As described in Reference A-6, Section 5.5.2.3:

> The initial oxide depth of the cladding outer surface is determined by the STAV7.2 fuel performance code.

Per the NRC SER of Reference A-6:

> The SER acknowledged the change from STAV6.2 to STAV7.2, but placed a condition on the use of this feature in that it could not be used until the staff's review of STAV7.2 (Reference A-11) was complete. The SER of Reference A-11 did not place any condition on the use of STAV7.2 except for limiting the rod average burnup range of STAV7.2 to 62 GWd/MTU. Westinghouse provided a letter to the NRC (Reference A-9), as required by the condition, after NRC approval of the STAV7.2 code was complete.

### A.2.9 Thermal Radiation Model

As described in Reference A-4, Section 5.4:

> The basic model of thermal radiation, as described in Section 4.5 and 4.5.1 of Reference A-1, remains unchanged.
>
> The gray body factors used in the radiation model are still calculated with the BILBO code as described in Section 4.5.2 of Reference A-1. However, the gray body factors are now calculated throughout the transient. To facilitate this, the BILBO code has been incorporated into CHACHA-3. The change was done to make the radiation model consistent with the new rod deformation model (described in Section 5.6 of Reference A-4), which calculates individual time-dependent dimensions for each fuel rod. The gray body factors are first calculated by BILBO at the beginning of the CHACHA-3 calculation using the initial geometry. They are updated transiently when a significant change in geometry or emissivity has occurred.

Per the NRC SER of Reference A-4:

> The SER acknowledged the change to the CHACHA code and placed no limitations or conditions on the use of the change.

### A.2.10 Gas Plenum Temperature and Pressure Model

As described in Reference A-6, Section 5.5.1:

> The detailed fuel heatup computer code (CHACHA-3D) has been revised to provide a new plenum type that permits a conservative prediction of the plenum temperature of the part-length rods. For this plenum type, the gas temperature in the rod plenum is determined conservatively by equating it to the maximum of the plenum cladding outer surface temperature, which is calculated in the hot channel analysis, and the gas temperature determined using the conventional plenum model.

Per the NRC SER of Reference A-6:

> The SER acknowledged the new part-length rod plenum model and found it acceptable. No limitations or conditions were placed on its use.

### A.2.11 Pellet-Cladding Gap Heat Transfer Model

As described in Reference A-4, Section 5.5:

> Due to the replacement of the PAD code with the STAV6.2 code, the pellet-cladding gap heat transfer model in the original CHACHA code was replaced with that from the STAV6.2 code with one modification. The contact conductance term is neglected in the CHACHA model to ensure conservatism when the clad and fuel are computed to be in contact.

Per the NRC SER of Reference A-4:

> Since the model was identical to that in STAV6.2, a detailed review of this model was not performed as it was to be done as part of the review of CENPD-285-P (Reference A-10). The SER for CENPD-285-P-A did not place any restriction on the use of the STAV6.2 gap heat transfer model except for the limitation to a rod average burnup of 50 GWd/MTU.

As described in Reference A-12, Appendix A:

> The input to CHACHA from STAV6.2 consists of all the data describing the fuel conditions at the initiation of the LOCA. The fuel rod heatup calculations used to derive the inputs to the CHACHA gap heat transfer model make use of bounding input for model parameters, fuel geometry, and power history. A conservative representation of a reference core limiting power history was used.

Per the NRC SER for Reference A-12:

> The SER adopted the TER evaluation, which acknowledged the conservative approach and concluded that the use of the STAV6.2 initialization for LOCA was acceptable. No limitations or conditions were placed on the use of STAV6.2 to provide initial conditions for CHACHA using the approach described.

As described in Reference A-6, Section 5.5.2.3:

> No changes were made to the CHACHA gap heat transfer model as a result of the change to the STAV7.2 fuel performance except that initial conditions for the model would be taken from the STAV7.2 calculations.

Per the NRC SER for Reference A-6:

> The SER acknowledged that CHACHA would receive inputs from the STAV7.2 code to initialize the gap heat transfer model, but placed a condition on approval of this change pending completion of the NRC review of the STAV7.2 code. Westinghouse provided a letter to the NRC (Reference A-9), as required by the condition, after NRC approval of the STAV7.2 code was complete.

As described in Reference A-13, Section 4.4.4:

> The fuel rod heatup calculations used to derive the inputs to the CHACHA gap heat transfer model are based on bounding segmented power histories and conservative fuel parameters.

Per the NRC SER for Reference A-13:

> The SER acknowledged the use of the segmented power history approach to bound all operation defined by the thermomechanical operating limit (TMOL) and concluded that the LOCA initialization methods were acceptable. No limitations or conditions were placed upon the use of this methodology other than to limit its application to a peak rod average burnup of 62 GWd/MTU.

### A.2.12 Cladding Strain and Rupture Model

As described in Reference A-4, Section 5.6:

> The mechanistic models described in Section 5.6 replaced the empirical correlations presented in Section 4.9 of Reference A-1. The mechanical models for the fuel rod cladding are used to determine the geometry of the fuel rods (outside diameter of the rods, size of the gap between the $UO_2$ pellet and the cladding, and cladding thickness).
>
> The mechanistic model for cladding burst, which gives a burst stress as a function of material properties and temperature, accounts for the influence of surface oxide and oxygen that has diffused into the Zircaloy. The burst stress is compared to the true, actual stress to detect a

rupture. The true, actual stress is calculated as a function of the pressures inside and outside the rod and the strained dimensions of the rod.

Per the NRC SER of Reference A-4:

The SER acknowledged the revision to the cladding strain and rupture model but placed a condition on its use that requires a bias of -0.5 MPa to be placed on the burst stress.

As described in Reference A-5, Section 4.1:

In addition to the cladding burst criterion described in Section 5.6 of Reference A-4, a second criterion was added to require cladding burst upon rod-to-rod contact.

Thus, the criteria for determining fuel rod rupture became 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.

Per the NRC SER of Reference A-5:

The NRC acknowledged the change and concluded that the change 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. No limitations or conditions were placed on the application of this change.

### A.2.13 Fuel Bundle Material Properties

As described in Reference A-4, Appendix A:

The fuel properties in CHACHA were changed to be consistent with fuel performance models derived from the STAV6.2 fuel performance code of Reference A-10. This includes density, thermal conductivity, and specific heat for uranium oxide (with and without $Gd_2O_3$), Zircaloy-2 and Zircaloy-4, and zirconium oxide.

Per the NRC SER of Reference A-4:

The SER placed a condition on the revisions to CHACHA that were based on STAV6.2 to be subject to the review findings of Reference A-10. The only condition resulting from this review was that applications must be to peak rod average burnups less than or equal to 50 GWd/MTU.

As described in Reference A-6, Section 5.5.2:

A model to account for the burnup-induced degradation of fuel pellet conductivity was introduced in CHACHA. This model was consistent with the STAV7.2 fuel performance code.

Per the NRC SER of Reference A-6:

> The SER placed a condition on the revisions to CHACHA that were based on STAV7.2 to be subject to the review findings of STAV7.2, which were ongoing at the time. Westinghouse provided a letter to the NRC (Reference A-9), as required by the condition, after NRC approval of the STAV7.2 code was complete. No limitations were placed on the model other than to limit applications to peak rod average burnups less than or equal to 62 GWd/MTU.

## A.3 APPLICATION OF THE EVALUATION MODEL TO A NEW FUEL DESIGN

The U.S. version of the Westinghouse BWR ECCS Evaluation Model has been qualified and approved for application to several fuel designs. The specific designs are QUAD+, SVEA-96, SVEA-96+, and SVEA-96 Optima2. The same methodology has been applied in Europe to additional fuel designs (e.g., open lattice 8 x 8, SVEA-64, SVEA-100, and SVEA-96 Optima).

The qualification process described for various fuel designs, which is discussed in Reference A-14, is shown in Figure A-1 and summarized below.

### A.3.1 Methodology

If all the qualification criteria are met, the ECCS Evaluation Model is acceptable for application to the specific fuel mechanical design. If any step described below does not fulfill the qualification criteria, then the LOCA ECCS Evaluation Model may not be applied for the new fuel mechanical design prior to specific NRC review and approval.

1. Nodalization – Fuel design-specific models are developed for the GOBLIN, DRAGON, and CHACHA-3D codes that capture the geometrical characteristics of the fuel design that are important to the key phenomena of a LOCA event.

2. CPR Correlation – The critical power ratio (CPR) correlation used is an NRC-approved CPR correlation that has been shown to conservatively predict early boiling transition in a LOCA event for the specific fuel design.

3. CCFL Correlation – The countercurrent flow limit (CCFL) model used is demonstrated as conservative relative to applicable experimental data for the specific mechanical fuel design.

4. Spray Cooling Convective HTC – The spray cooling heat transfer coefficients used are demonstrated as conservative relative to applicable experimental data for the specific mechanical fuel design.

5. Transition Cores – A full core configuration of the specific fuel design is used in LOCA ECCS performance evaluation applications. Acceptability for transition cores is confirmed by comparing the following reactor system responses for analyses performed assuming a full core of the applicable co-resident fuel designs:

    a. time of reactor trip

b. time of boiling transition at the midplane of the hot assembly
c. time of dryout at the midplane of the hot assembly
d. times of ECCS actuation
e. time of reflood of the midplane of the hot assembly

The following sections provide discussion of each item above in the methodology statement.

### A.3.2 Nodalization

The GOBLIN average reactor core and hot channel nodalization are selected to represent the fuel design features important to ECCS performance analysis. These features include the fuel rod dimensions, fuel assembly active cross-sectional flow areas, locations and characteristics of inter- and intra-assembly flow paths, grid spacers, and tie plates. Axial node size in the GOBLIN models is selected to ensure there is sufficient detail to characterize thermal-hydraulic conditions along the channel and at the hot plane. When it is impractical to reduce axial node size sufficiently to capture important mixture level dynamics, GOBLIN's two-phase level tracking feature may be used to determine the position of the mixture level more precisely.

The CHACHA-3D geometric model is selected to represent fuel design-specific rod or rod lattice configuration, channel configuration, fuel pellet, cladding and gap dimensions, and fuel rod plenum dimensions.

### A.3.3 CPR Correlation

The CPR correlation is used to (1) determine the initial power of the hot assembly that will have it operating at bounding operating conditions, and (2) determine the time of boiling transition during the blowdown phase of the LOCA. GOBLIN has several CPR correlations available to the user. The CPR correlation that is applicable to the fuel design being evaluated or demonstrated to be conservative relative to a NRC-approved correlation for that fuel design is selected by the analyst to ensure that the hot assembly power and the time of dryout are predicted conservatively. To ensure that the critical power is calculated conservatively, a modified pool boiling correlation is also used to determine the critical power. The code then determines the critical power by selecting the smaller of the two calculated values. The following NRC-approved CPR correlations are currently available to the user:

| CPR Correlation | Application |
|---|---|
| XL-S96 | SVEA-96 |
| ABBD1.0 | SVEA-96 |
| ABBD2.0 | SVEA-96+ |
| D4.1.1 | SVEA-96 Optima 2 |

CPR correlations are applicable to specific fuel designs or a group of fuel designs. The SER for RPB 90-93-P-A (Reference A-1) requires that an appropriate NRC-approved CPR correlation be used when GOBLIN is used in a licensing analysis. The NRC-approved correlation may be one that has been developed specifically for the fuel design, or shown to be conservative relative to an NRC-approved correlation for that fuel design. Changes to GOBLIN are necessary when a new CPR correlation is implemented. The process described below is used by Westinghouse to install and test NRC-approved CPR correlations. Changes to GOBLIN following this process do not require specific NRC review and approval. Such changes will be communicated to the NRC via the 10 CFR 50.46 annual reporting process.

The process used to install and qualify a CPR correlation in GOBLIN is as follows:

1. Develop coding to represent the new correlation. The coding includes checks on correlation parameters to ensure that inputs to the correlation are within valid ranges of those parameters. If a parameter is outside its range of validity, the [ ]$^{a,c}$

2. Validation of the implemented CPR correlation is performed by:

   a. Transient code simulation of transient experimental data, or

   b. Transient code to transient code comparisons where the reference transient code implementation of the CPR correlation has been qualified against transient experimental data.

3. Ensure NRC approval of CPR correlation for the fuel design prior to its use in licensing applications.

4. Inform the NRC of the change to GOBLIN via the 10 CFR 50.46 annual reporting process.

If a LOCA analysis of non-Westinghouse fuel is required, Westinghouse may not have direct access to the accepted correlation for the resident fuel. In this case, sufficient information is obtained from the utility to either:

1. Allow renormalization of an NRC-approved Westinghouse CPR correlation for Westinghouse fuel to describe the CPR performance of the fuel, or

2. Show that the NRC-approved Westinghouse CPR correlation for Westinghouse fuel is conservative.

CPR correlations are valid within specified ranges of parameters (e.g., system pressure, core mass flux, inlet subcooling). When a CPR correlation is implemented in GOBLIN, it is only applied when conditions in the core are within its range of applicability. If any parameter is outside its valid range, a pool boiling CHF correlation is used. Since the system pressure and core flow decrease very rapidly following a large-break LOCA, the prediction of boiling transition is often the result of exceeding the [ ]$^{a,c}$. Experience has shown that the fuel-specific CPR correlation selected [ ]$^{a,c}$

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[ ]$^{a,c}$

[

]a,c

The process for developing the renormalized CPR correlation is described in Section 5.3.2.5 of Reference A-14. Implementation of the renormalized CPR correlation in GOBLIN follows the process outlined above.

### A.3.4 CCFL Model

The CCFL model has been approved for a variety of fuel designs. In accordance with CENPD 283-P-A (Reference A-3), this correlation will not be extended to fuel designs outside the range of approved applicability without being supported by experimental data. NRC review and approval of the new CCFL model is required prior to its use in licensing applications.

The change to the CCFL model in GOBLIN that is described in Section 5.4.2 of Reference A-6 removes a restriction placed on the USA2 Evaluation Model.[2]

### A.3.5 Spray Cooling Convective Heat Transfer

A methodology to extrapolate spray cooling heat transfer coefficients for application to a variety of fuel designs has been approved. In accordance with CENPD-283-P-A (Reference A-3), this methodology will not be extended to fuel designs outside the range of applicability without being supported by experimental data. If the spray cooling heat transfer coefficients cannot be demonstrated as applicable, spray cooling heat transfer coefficients must be determined either from a detailed analysis that has been validated by experimental data or directly from applicable data. NRC review and approval of the new spray cooling heat transfer coefficients are required prior to their use in licensing applications.

### A.3.6 Transition Cores

The BWR fuel channel and fuel mechanical designs are established to ensure hydraulic compatibility with co-resident fuel. This means that the system response to a LOCA event for one core of mixed fuel designs will be very similar hydraulically to that of a full core of a single fuel design. This observation has been demonstrated for several fuel designs in References A-2 and A-3. It became a requirement to specifically analyze a transition core during the first reload analysis following the NRC acceptance of WCAP-16078-P-A (Reference A-6). If it is confirmed that a full core of Westinghouse fuel is bounding, then the Evaluation Model can be performed using the full-core Westinghouse fuel approach. Otherwise, the mixed-core model must be used. The Westinghouse Evaluation Model may not be used to calculate the MAPLHGR limits for non-Westinghouse fuel for a mixed-core analysis. If the transition core analysis indicates that the system performance of the mixed core is more limiting than the full-core analysis of the

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2 In responding to a request for additional information relative to the NRC review of CENPD-283-P-A (Reference A-3), Westinghouse committed to applying a conservative bias to the CCFL correlation to bound all the scatter in the correlation database for LOCA applications in which the calculated peak cladding temperature exceeded 2100°F.

legacy fuel, Westinghouse must request the utility to contact the legacy fuel vendor for an evaluation of the impact of the mixed core on the MAPLHGR limits for their fuel.

## A.4 TABLES FOR APPENDIX A

**Table A-1 Roadmap to Evaluation Model Changes**

| Evaluation Model Element | Reference No. | | | | | Road Map Section |
|---|---|---|---|---|---|---|
| | A-1 | A-4 | A-3 | A-5 | A-6 | |
| ***Thermal-Hydraulic Model – GOBLIN*** | | | | | | |
| Mass Conservation Equations | 3.1.1 | | | | | |
| Energy Conservation Equations | 3.1.2 | | | | | |
| Momentum Conservation Equations | 3.1.3 | 4.2 | | | | A.2 |
| Fluid Properties | 3.2.1 | | | | | |
| Equation of State | 3.2.2 | | | | | |
| Two-Phase Energy Flow Model | 3.3.1 | | 7.1 | | 5.4.2 | A.2.2 |
| Two-Phase Level Tracking | 3.3.2 | | 6.1.2 | | 5.1.1 | A.2.3 |
| Frictional Pressure Drop Correlations | 3.3.3 | | | | | |
| Form Pressure Drop Correlations | 3.3.4 | | | | | |
| Injection Flow – Fluid Interaction | 3.3.5 | | | | | |
| Critical Flow Model | 3.3.6 | | | | | |
| Recirculation Pump Model | 3.4.1 | | | | | |
| Jet Pump Model | 3.4.2 | | | | | |
| Separator and Dryer Model | 3.4.3 | | | | | |
| Feedwater and Steamline Systems | 3.4.4 | | | | - | |
| Reactor Measurement and Protection Systems | 3.4.5 | | | | | |
| Heat Transfer Regimes | 3.5.1 | | | | | |
| Convective Heat Transfer Coefficients | 3.5.2 | | 4.3 | | 6.1 | A.2.4 |
| Critical Power Ratio Correlation | 3.5.3 | 4.1 | 4.2 | | 5.4.1 | A.2.5 |
| Transition Boiling | 3.5.4 | | | | | |
| Radiation Heat Transfer | 3.5.5 | | | | | |
| Fuel Rod Conduction Model | 3.6.1 | | | | | |
| Plate Conduction Model | 3.6.2 | | | | | |
| Material Properties | 3.6.3 | | | | | |
| Point Kinetics Model | 3.7.1 | | | | | |
| Metal-Water Reaction Model | 3.7.2 | | | | | |
| Point Kinetics Solution | 3.8.1 | | | | | |

| Table A-1 Roadmap to Evaluation Model Changes (cont.) | | | | | | |
|---|---|---|---|---|---|---|
| | **Reference No.** | | | | | **Road Map Section** |
| **Evaluation Model Element** | **A-1** | **A-4** | **A-3** | **A-5** | **A-6** | |
| Hydraulic Model Solution | 3.8.2 | | | | | |
| Heat Conduction and Transfer Solution | 3.8.3 | | | | | |
| Nodalization | 3.9 | | | | | |
| ***Rod Heatup Model – CHACHA*** | | | | | | |
| Fuel Rod Conduction Model | 4.1 | 5.1 | | | 5.5.2 | A.2.6 |
| Channel Temperature Model | 4.2 | | | | | |
| Heat Generation Model | 4.3 | 5.2 | | | 5.5.2 | A.2.7 |
| Metal-Water Reaction Model | 4.4 | 5.3 | | | 5.5.2 | A.2.8 |
| Thermal Radiation Model | 4.5 | 5.4 | | | | A.2.9 |
| Gas Plenum Temperature and Pressure Model | 4.6 | | | | 5.5.1 | A.2.10 |
| Channel Rewet Model | 4.7 | | | | | |
| Pellet-Cladding Gap Heat Transfer Model | 4.8 | 5.5.1 | | | 5.5.2.3 | A.2.11 |
| Cladding Strain and Rupture Model | 4.9 | 5.6 | | 4.1 | | A.2.12 |
| Fuel Bundle Material Properties | App. 4.A | App. A | | | 5.5.2 | A.2.13 |

| Table A-2 Extrapolated Spray Cooling Heat Transfer Coefficients per Reference A-3 | | | | |
|---|---|---|---|---|
| | **Extrapolated from Appendix K Values ($W/m^2$-K)** | | | |
| **Geometry** | **Corner Rods** | **Side Rods** | **Inner Rods** | **Channel** |
| Appendix K 7 x 7, 8 x 8, Isotropic Radiation | 17.0 | 19.9 | 8.5 | 28.4 |
| Appendix K 8 x 8, Anisotropic Radiation | 16.8 | 19.4 | 11.9 | 28.4 |
| SVEA-64, Anisotropic Radiation | 15.0 | 17.3 | 10.6 | 25.3 |
| SVEA-96, Anisotropic Radiation | 15.0 | 17.3 | 10.6 | 25.3 |

| Table A-3 Extrapolated Spray Cooling Heat Transfer Coefficients per Reference A-6 | | | | |
|---|---|---|---|---|
| | Extrapolated from Appendix K Values (W/m$^2$-K) | | | |
| | Corner Rods | Side Rods | Inner Rods | Channel |
| SVEA-96, Anisotropic Radiation | 15.0 | 17.3 | 10.6 | 25.3 |
| SVEA-96 Optima2 | 15.0 | 17.3 | 10.6 | 25.3 |

## A.5 FIGURES FOR APPENDIX A

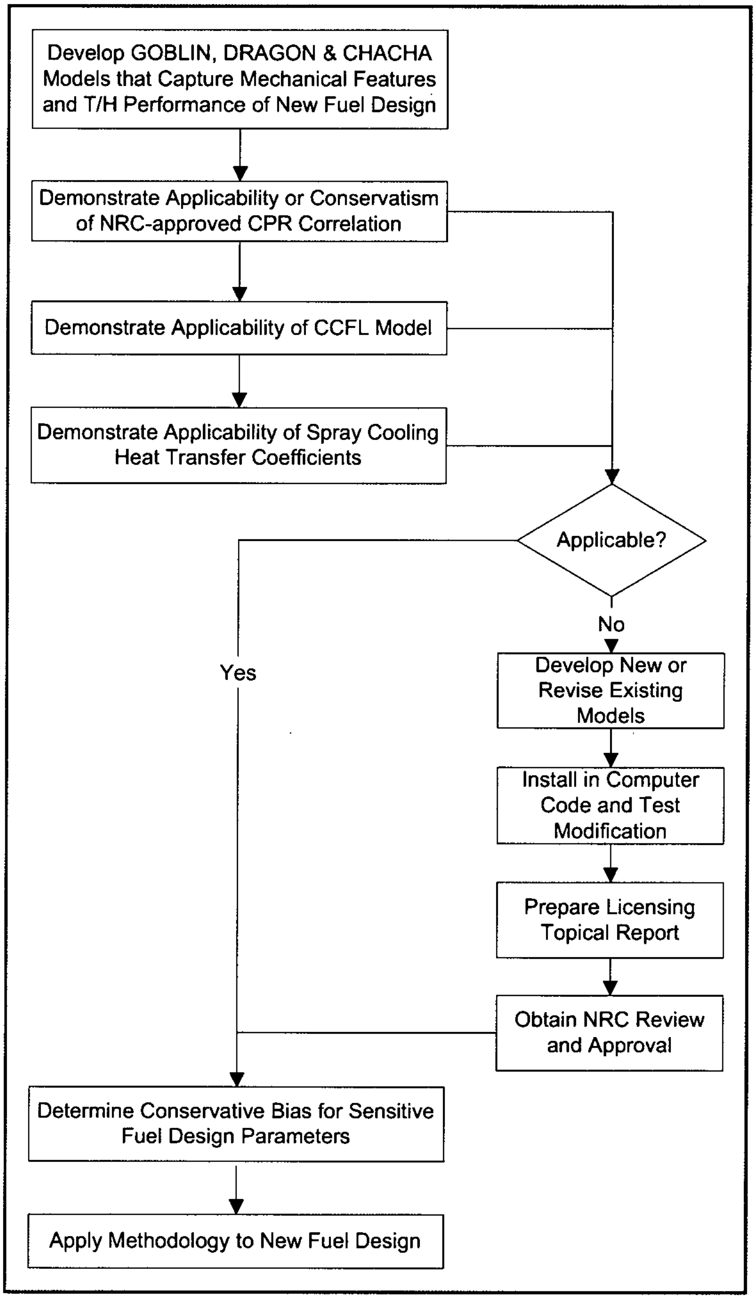


**Figure A-1 Process for Applying Evaluation Model to New Fuel Mechanical Design**

## A.6 REFERENCES FOR APPENDIX A


A-1 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," Westinghouse Report RPB 90-93-P-A (Proprietary), RPB 90-91-NP-A (Non-Proprietary), October 1991.

A-2 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," Westinghouse Report RPB 90-94-P-A (Proprietary), RPB 90-92-NP-A (Non-Proprietary), October 1991.

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

A-4 "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," Westinghouse Report CENPD-293-P-A (Proprietary), CENPD-293-NP-A (Non-Proprietary), July 1996.

A-5 "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," Westinghouse Report WCAP-15682-P-A (Proprietary), WCAP-15682-NP-A (Non-Proprietary), April 2003.

A-6 "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," Westinghouse Report WCAP-16078-P-A (Proprietary), WCAP-16078-NP-A (Non-Proprietary), November 2004.

A-7 "SVEA-96 Critical Power Experiments on a Full Scale Sub-bundle," ABB Atom Report UR-89-210-P-A, October 1993.

A-8 Letter from A.C. Thadani (NRC) to W. R. Russell (ABB Atom), "Waiver of CRGR Review of the Safety Evaluation of ABB Supplemental Information Regarding UR 89-210 Safety Evaluation Report," July 12, 1993.

A-9 Letter from B. F. Maurer (Westinghouse) to F. M. Akstulewicz (NRC), "Westinghouse response to Condition 1 in the FINAL SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16078-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" (TAC NO. MB8908), October 21, 2004," LTR-NRC-06-1, January 4, 2006.

A-10 "Fuel Rod Design Methods for Boiling Water Reactors," Westinghouse Report CENPD-285-P-A (Proprietary), CENPD-285-NP-A (Non-Proprietary), July 1996.

A-11 "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Report WCAP-15836-P-A (Proprietary), WCAP-15836-NP-A (Non-Proprietary), April 2006.

A-12 "Fuel Rod Design Methodology for Boiling Water Reactors," Westinghouse Report CENPD-287-P-A, July 1996.

A-13 "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287," Westinghouse Report WCAP-15942-P-A (Proprietary), WCAP-15942-NP-A (Non-Proprietary), March 2006.

A-14 "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A (Proprietary), CENPD-300-NP-A (Non-Proprietary), July 1996.