SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
LICENSING TOPICAL REPORT WCAP-17116-P
"WESTINGHOUSE BWR ECCS EVALUATION MODEL: SUPPLEMENT 5 - APPLICATION TO THE ABWR"
PROJECT NO. 0772

1.0 INTRODUCTION

South Texas Project Nuclear Operating Company submitted WCAP-17116-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 5 – Application to the ABWR" (hereafter referred to as Supplement 5) for U.S. Nuclear Regulatory Commission (NRC) review and approval by letter dated September 30, 2009 (Ref. 1). Supplement 5 is an extension of the Westinghouse boiling-water reactor (BWR) emergency core cooling system (ECCS) evaluation model (EM) to the advanced boiling-water Reactor (ABWR) loss-of-coolant accident (LOCA) analysis. The Westinghouse ABWR ECCS EM, as described in Supplement 5, is identified as USA7. It consists of the GOBLIN and CHACHA-3D proprietary computer codes, as well as the parameters and input required to select features and enable the use of the desired models. GOBLIN is a one-dimensional, two-phase thermal-hydraulic code, and CHACHA-3D is a one-dimensional code that performs detailed temperature calculations at a particular axial level within a fuel assembly.

GOBLIN analyzes the time-dependent thermal-hydraulic response of the reactor coolant system during the blowdown and the re-flood phases of loss of coolant accidents (LOCAs), including the interactions with various control and safety systems. GOBLIN calculates the pressure and enthalpy at the core inlet and outlet, using the core power generation, system geometry, ECCS performance, and the break specification. The GOBLIN code calculations are performed by modeling the reactor core with two parallel flowpaths, with one flowpath representing the hot assembly, and the other flowpath representing the remaining assemblies. CHACHA-3D performs detailed temperature calculations at a specified axial level within the hot assembly previously analyzed by the GOBLIN code. All necessary fluid boundary conditions are obtained from the GOBLIN calculation. CHACHA-3D determines the temperature distribution of each rod throughout the accident and ultimately determines the peak cladding temperature (PCT) and cladding oxidation at the axial plane under investigation. CHACHA-3D also provides input for the calculation of total hydrogen generation by supplying the local oxidation at various axial and radial core locations.

BACKGROUND

The staff first approved the Westinghouse BWR ECCS EM, incorporating the then-current versions of GOBLIN/DRAGON and CHACHA codes, in 1989. The initial version of the Westinghouse BWR ECCS EM was identified as USA1 (Refs. 2, 3).

DRAGON is a mode or option within GOBLIN that uses the GOBLIN results (i.e., the plenum to plenum flow boundary conditions), to analyze the response of hot channel. The DRAGON option determines the response of the hot channel to the LOCA event (e.g., boiling transition, dryout, and refill). Since in USA7 the GOBLIN code calculations are performed by modeling the reactor core with two parallel flowpaths, the DRAGON option of GOBLIN is not exercised for ABWR applications.

ASEA Brown Boveri (ABB)/ Combustion Engineering (ABB/CE) supplemented the original BWR ECCS licensing topical report (LTR) with CENPD-293-P, "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," for staff review in 1994. Approval for Supplement 1 to the EM (known as USA2) was granted in 1996 (Ref. 4). The purpose of CENPD-293-P-A was to update the EM to enable analysis of cores containing SVEA-96 fuel.

Westinghouse Electric Company (Westinghouse) submitted WCAP-15682-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," for NRC review in 2002.

The purpose of Supplement 2 was to introduce improved fuel cladding rupture criteria and provide qualification bases for the improvement while maintaining the overall conservatism of the previously approved versions of the EM. The staff approved Supplement 2 to the EM (known as USA4) in 2003 (Ref. 5).

In 2003, Westinghouse submitted WCAP-16078-P, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," for NRC review. The purpose of Supplement 3 was to update the EM to enable analysis of cores containing SVEA-96 Optima 2 fuel design. The staff approved Supplement 3 in 2004, with the EM identified as USA5 (Ref. 6).

Westinghouse submitted WCAP-16865-P, "Westinghouse BWR ECCS Evaluation Model Updates: Supplement 4 to Code Description, Qualification and Application," for NRC review in 2007. The updates to the EM in Supplement 4 included a change to determine the end of lower plenum flashing calculations for small- and large-break LOCA analysis. The staff approved Supplement 4, with the EM identified as USA6 in 2011 (Ref. 7).

The previously approved BWR ECCS evaluation models were developed for application to BWR/2 to BWR/6 designs. The design of the ABWR is quite different to that of BWR/2 to BWR/6. The differences include the presence of internal recirculation pumps in the ABWR and the lack of any external recirculation piping. Therefore, the response of the ABWR system to a LOCA event is quite different than for a BWR with external recirculation pumps. In ABWR, the core flow rate decreases quickly because of the rapid coastdown of the reactor internal pumps (RIPs) following pump trip (if the loss of offsite power is assumed). Also the elevations of large pipe breaks are above, not below, the top of active fuel. As a result, different features of the evaluation model become more important and require additional qualification.

The stated objective of Supplement 5 is to provide a basis for extending the applicability of the EM to the ABWR.

2.0 REGULATORY EVALUATION

The staff used the review guidance provided in Standard Review Plan (SRP) Section 15.0.2, "Review of Transient and Accident Analysis Methods," (SRP 15.0.2, Ref. 8) in the conduct of its review of Supplement 5. The review covered the areas of: (1) documentation, (2) evaluation model, (3) accident scenario identification process, (4) code assessment, (5) uncertainty analysis, and (6) quality assurance plan. SRP 15.0.2 incorporates the requirements expressed in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors" and those in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix K, "ECCS Evaluation Models" (Ref. 9).

10 CFR 50.46

A LOCA, as defined in 10 CFR 50.46, is a postulated accident to determine the design acceptance criteria for a plant's ECCS (Ref. 9). The calculated maximum fuel element cladding temperature shall not exceed 1,204 degrees Celsius (C) (2,200 degrees Fahrenheit (F) or 1,477 degrees Kelvin (K)).

To establish that the ECCS design performance satisfies the aforementioned criteria, SRP 15.0.2 (Ref. 8) and 10 CFR 50.46 (Ref. 9) provide guidance to ensure that the ECCS analyses is performed using an acceptable ECCS EM. Regulations in 10 CFR 50.46 state that the EM must identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria are not exceeded. Alternatively, the EM may be conservatively developed in conformance with the required and acceptable features of 10 CFR Part 50, Appendix K. Supplement 5 states that USA7 is compliant with the regulatory requirements of 10 CFR 50.46.

10 CFR Part 50, Appendix K

Appendix K to 10 CFR Part 50 specifies the required and acceptable features for ECCS evaluation models. The Appendix K requirements and features that are relevant to the present review are listed below:

Section I.A: Sources of heat during the LOCA

Section I.A.1: The initial stored energy in the fuel

Section I.A.2: Fission heat

Section I.A.3: Decay of actinides

Section 1.A.4: Fission product decay

Section 1.A.5: Metal-water reaction rate

Section I.A.6: Reactor internals heat transfer

Section I.A.7: Pressurized water reactor primary-to-secondary heat transfer (applies only to PWR)

Section I.B: Swelling and rupture of the cladding and fuel rod thermal parameters

Section I.C.1.a: Break characteristics and flow

Section I.C.1.b: Discharge model

Section I.C.1.c: End of blowdown (applies only to PWR)

Section I.C.1.d: Noding near the break and the ECCS injection points

Section I.C.2: Frictional pressure drops

Section I.C.3: Momentum equation

Section I.C.4.(a–e): Critical heat flux

Section I.C.5.(a and b): Post-CHF heat transfer correlations

Section I.C.6: Pump modeling

Section I.C.7: Core flow distribution during blowdown (applies only to PWR)

Section I.D.1: Single failure criterion

Section I.D.2: Containment pressure

Section I.D.3: Calculation of reflood rate for pressurized water reactors (applies only to PWR)

Section I.D.4: Steam interaction with emergency core cooling water in pressurized water reactors (applies only to PWR)

Section I.D.5: Refill and reflood heat transfer for pressurized water reactors (applies only to PWR)

Section I.D.6: Convective heat transfer coefficients for boiling water reactor fuel rods under spray cooling (in ABWR, the core flooding is used rather than spray cooling; this requirement, therefore, does not apply to ABWR)

Section II (1.a, 3, and 4): Required documentation

3.0 TECHNICAL EVALUATION

The staff performed its review of Supplement 5 to the Westinghouse BWR ECCS EM as reported in WCAP-17116-P under a technical assistance contract with Energy Research, Inc. (ERI). Detailed descriptions of the evaluation and findings of the review are described in the ERI technical evaluation report (TER).

A key element of this review is to recognize that SRP 15.0.2 (Ref. 8) provides an evaluation methodology that may be used to determine if a change to an EM should be subjected to a graded or a full review. The SRP recommends judging the submittal against the following criteria: (1) novelty, (2) complexity, (3) degree of conservatism in the evaluation model, and (4) extent of plant or operational changes requiring reanalysis. With respect to this guidance, the staff makes the following observations:

- Novelty: The changes to the EM in Supplement 5 are not particularly novel. They consist primarily of changes to noding and structure commensurate with ABWR component dimensions, elevations, and descriptions. The staff believes that consideration of this aspect does not indicate that a full review of the EM is required.

- Complexity: There are no significant changes to the complexity of the EM resulting from Supplement 5 changes. The staff believes that consideration of this aspect does not indicate that a full review of the EM is required.

- Degree of conservatism in the EM: This consideration is not applicable as the conservatism is addressed fully by using the 10 CFR Part 50, Appendix K requirements.

- Extent of changes requiring reanalysis: Since ABWR has not yet been licensed, it is not necessary to consider the issue of reanalysis.

On the basis of the evaluation above, the staff believes that a graded review is appropriate for Supplement 5, and the staff conducted its review in this manner. The criteria used for this focused review are the same as for a full review. The EM elements reviewed, however, are limited to those introduced by the LTR for which approval is sought or to those elements that are related because of their importance to ABWR analysis but not specifically discussed in sufficient detail to allow the staff to conclude that they are equally applicable to BWR/2 through BWR/6.

To accomplish the required review objectives, the staff issued several requests for additional information (RAIs 1 to 33). RAIs were answered in a series of seven responses. In addition, the staff conducted an audit in February 2011, during which several RAIs were discussed in considerable detail (Ref. 25).

3.1 Documentation

The staff reviewed Supplement 5 to determine the adequacy of the documentation relative to the review guidance provided in SRP 15.0.2.

The documentation in Supplement 5 and that was submitted in response to the staff's questions included the following in accordance with the requirements stated in SRP 15.0.2:

- an overview of the evaluation model
- a complete description of the accident scenario
- a complete description of the code assessment
- a determination of the code uncertainty
- a quality assurance plan

The staff finds the submitted documentation to be acceptable. Based on the review of Supplement 5, responses to RAIs and questions during the audit, the staff determined that the graded review did not require inspection of the theory and user manuals. The staff concluded that the previously reviewed versions of these documents are acceptable.

3.2 Evaluation Model

USA7 is the ECCS evaluation model based on 10 CFR Part 50, Appendix K. Therefore, the review in this section is focused on assessing the conformance of USA7 to the requirements of 10 CFR Part 50, Appendix K. The applicability of USA7 to ABWR is determined based on the review of Supplement 5 and the evaluation of responses to RAIs and discussions during the audit. Section 3.2.1 presents discussion of important elements of the USA7 model. The conformance of USA7 to each applicable item of the requirements established in 10 CFR Part 50, Appendix K is evaluated in Section 3.2.2.

3.2.1 Assessment of USA7 Model Elements

Table 1 shows various elements of USA7 as documented in Table A-1 of Supplement 5 and the relevant Appendix K criteria. Since the basic thermal-hydraulic phenomena are the same for the ABWR and the operating BWRs, many elements of USA7 that were found to be applicable for the BWR/2 through BWR/6 are still applicable for the ABWR. Therefore, only those aspects of USA7 that are unique for the ABWR analysis are addressed here. These include the following:

- nodalization
- use of the two-phase level tracking model
- ECCS injection flow
- break flow and ADS flow models
- conservative model assumptions

The relevant USA7 model elements that are addressed in this SER are highlighted in Table 1. Table 1 also shows the associated RAIs and sections of this SER.

Table 1 - Applicability of USA7 to ABWR

EM Element	Appendix K Criteria	Applicability to ABWR		
		Y/N	Basis*	SER Sections and RAI
Thermal-Hydraulic Models in GOBLIN				
Mass Conservation Equations	-	Y	A1	-
Energy Conservation Equations	-	Y	A1	-
Momentum Conservation Equations	I.C.3	Y	A1, A2	Section 3.2.2
Fluid Properties	-	Y	A1	-
Equation of State	-	Y	A1	-
Two-Phase Energy Flow Model	-	Y	A1	-
Two-Phase Level Tracking	-	Y	A1, A3	-
Frictional Pressure Drop Correlations	I.C.2	Y	A1, A2	Section 3.2.2
Injection Flow – Fluid Interaction	I.C.1.d	Y	A1, A2, A3	Section 3.2.2, RAI-13 -14, -24
Critical Flow Model	I.C.1.b	Y	A1, A2, A3	Section 3.2.2, RAI-7, -8, -15
Recirculation Pump Model	I.C.6	Y	A2, A3	Section 3.2.2, RAI-5, 6, -10, -11, -12
Jet Pump Model	I.C.6	N	A4	Section 3.2.2
Separator and Dryer Model	-	Y	A1	-
Feedwater and Steam line Systems	-	Y	A1, A3	RAI-2
Reactor Measurement and Protection Systems	-	Y	A1, A3	RAI-18
Heat Transfer Regimes	I.A.6	Y	A1, A2, A3	Section 3.2.2, RAI-26
Convective Heat Transfer Coefficients	I.A.6	Y	A1, A2, A3	Section 3.2.2, RAI-26
Critical Power Ratio Correlation	I.C.4.c	Y	A1, A2, A3	Section 3.2.2, RAI-20, -21
Transition Boiling	I.C.5	Y	A1, A3, A2	Section 3.2.2, RAI-20
Radiation Heat Transfer	I.A.6	Y	A1, A2	Section 3.2.2
Fuel Rod Conduction Model	I.A.1	Y	A1, A2	Section 3.2.2
Plate Conduction Model	I.A.6	Y	A1, A3, A2	Section 3.2.2, RAI-16, -23
Material Properties	I.A.1 I.A.6, & I.B	Y	A1, A2	Section 3.2.2
Point Kinetics Model	I.A.2	Y	A1, A2	Section 3.2.2

EM Element	Appendix K Criteria	Applicability to ABWR		
		Y/N	Basis*	SER Sections and RAI
Metal-Water Reaction Model	I.A.5	Y	A1, A2, A3	Section 3.2.2, RAI-29
Point Kinetics Solution	I.A.2 & I.A.3	Y	A1, A2	Section 3.2.2
Hydraulic Model Solution	-	Y	A1	-
Heat Conduction and Transfer Solution	I.A.6	Y	A1, A2	Section 3.2.2
Nodalization	I.C.1.d	Y	A2, A3	Section 3.2.2, RAI-1, -13, -14, -24, -30
Rod Heat-up Models in CHACHA				
Fuel Rod Conduction Model	I.A.1 & I.B	Y	A1, A2	Section 3.2.2
Channel Temperature Model	-	Y	A1	-
Heat Generation Model	I.A	Y	A1, A2	Section 3.2.2
Metal-Water Reaction Model	I.A.5	Y	A1, A2, A3	RAI-29
Thermal Radiation Model	-	Y	A1	-
Gas Plenum Temperature and Pressure Model	I.A.1	Y	A1, A2	Section 3.2.2
Channel Rewet Model	I.D.7	Y	A1, A2	Section 3.2.2
Pellet-Cladding Gap Heat Transfer Model	I.B	Y	A1, A2	Section 3.2.2
Cladding Strain and Rupture Model	I.B	Y	A1, A2	Section 3.2.2
Fuel Bundle Material Properties	I.A and I.B	Y	A1, A2	Section 3.2.2

* A1: Fundamental phenomenon remains the same for ABWR. A2: In conformance with the 10 CFR Part 50, Appendix K requirements. A3: Based on the information in the LTR and applicant's responses to RAIs. A4: Not Applicable to the ABWR design.

3.2.1.1 Nodalization

The nodalization of the ABWR model used for USA7 is different from that used in previous submittals because of the differences in the design between the ABWR and BWR/2 through BWR/6. As compared to the previous Westinghouse BWR EM, the major change in the ABWR model is the increased number of nodes to represent the active core region. The core region is modeled using multiple parallel control volumes representing the average core and the hot channel. Section 4.3.1.1 of Supplement 5 indicates that 25 axial nodes are used to model the ABWR active core, as compared to 6 nodes that were used in the GOBLIN nodalization for BWR (Ref. 2). The steam dome is represented by a single control volume in USA7. The upper and lower downcomer regions are represented by nine one-dimensional control volumes (i.e., there is no azimuthal or radial nodalization). Feedwater, as well as reactor core isolation cooling (RCIC) and low pressure flooder (LPFL) injection are directed into the downcomer at the appropriate elevations. The break flow paths for the feedwater line break (FWLB) and the residual heat removal system (RHR) suction and discharge line breaks also are connected directly to the downcomer control volumes. [[ ]] The High Pressure Core Flooder (HPCF) injection is directed into the upper plenum. The noding near the break in the ABWR LOCA analysis is similar to that used in the BWR/2 through BWR/6 LOCA models (Ref. 2).

Based on the review of Supplement 5, the staff raised the following issues regarding the nodalization used in USA7:

- The nodalization sensitivity studies that result in the choice of 25 axial nodes for modeling the ABWR core in GOBLIN are not present in Supplement 5. Therefore, the justification for the use of 25 nodes, as required by Section II.3 of Appendix K to 10 CFR Part 50, is not evident. (RAI-1).

- The one-dimensional nodalization of the upper plenum and downcomer can lead to the prediction of incorrect thermal-hydraulic behavior. Because of the nodalization of the upper plenum, the mass inflow from the HPCF would be automatically distributed uniformly throughout the upper plenum node volume. The HPCF mass flow also would immediately reach thermal equilibrium. Furthermore, if the HPCF flow is available from only one sparger, the resulting asymmetric effects on core cooling would not be accounted for by the one-dimensional nodalization. Similarly, there is a potential for nonuniform delivery of LPFL flow in the downcomer. The one-dimensional downcomer nodalization may not be appropriate for possible imperfect mixing. There also is a potential for ECCS injection exiting through a line break without providing any cooling (e.g., FWLB). The justification for the use of one-dimensional nodalization for the downcomer and upper plenum regions, as required by Section II.3 of Appendix K to 10 CFR Part 50, is not provided.

- Supplement 5 states that noding near the break in the ABWR LOCA analysis is similar to that used in the BWR/2 through BWR/6 LOCA models. However, the BWR/2 through BWR/6 break location noding sensitivity studies were performed for a break in the recirculation line piping, which is not present in the ABWR. As noted in Supplement 5, the maximum ABWR line break size is only approximately 15 percent of the size of the double-ended recirculation line in a BWR. For the ABWR design, the effects of break location noding on break flow and system inventory may be different than for BWR/2 through BWR/6 designs. As required by Section 1.C.1.d of Appendix K, justification for the selected break location noding is necessary to determine its adequacy. Section 1.C.1.d of Appendix K also requires justification for the noding in the vicinity of ECCS injection point. Therefore, additional sensitivity studies are required to assess the effect of noding near the ECCS injection points and breaks in the upper plenum and in the downcomer (RAI-24).

In response to RAI-1, Westinghouse provided results from GOBLIN simulations of FRIGG tests using 15, 25, and 50 axial nodes. The predicted dryout times were similar for the 25- and 50-node cases, while the 15-node case showed slight over-prediction for some tests. Since FRIGG has similar scaling as the ABWR for a single bundle, the selected core noding is considered acceptable for the evaluation of an ABWR LOCA.

The drift flux model, as implemented in USA7, is a one-dimensional mixture model. It cannot resolve the exact location of the mixture level within a given nodal region. Consequently, if the node (or computational control volume) containing a mixture level is large, the calculated mixture void fraction (and quality) for the node would be different from the actual void fraction (and quality) below and above the mixture level. Therefore, the finer nodalization is essential to provide better estimation of the nodal void fraction and steam quality.

There is a mixture level tracking option available in USA7 that can be activated to calculate the actual position of the mixture level in a given node. Supplement 5 notes that the two-phase level tracking feature can be introduced when it is impractical to reduce the node size below a certain limit. However, this model cannot be activated in the core region. In the absence of the level tracking model in the core region, the finer nodalization is essential to improve the prediction of local void fraction and quality, which can also improve the prediction of boiling transition (dryout). Therefore, the finer nodalization selected in USA7 based on the FRIGG experiments also is considered to be acceptable for prediction of boiling transition under ABWR LOCA conditions.

In response to RAIs 13 and 14, Westinghouse noted that since the HPCF flow mixes with the upward flow from the reactor core and overflows into the downcomer via the steam separators, details of how the HPCF flow is distributed in the upper plenum are not important. Furthermore, the HPCF is not a core spray system and, therefore, the radial distribution of HPCF flow in the upper plenum is not important for safety analysis. The staff finds this explanation to be acceptable only if the flow at the core exit is in upward direction during the HPCF injection period under all LOCA conditions. If, in future applications of USA7, downward flow into the core is predicted during the HPCF injection period, the assumption of one-dimensional upper plenum nodalization should be reconsidered. Therefore, this imposes a limitation on the use of USA7 to restrict the one-dimensional nodalization in the upper plenum to conditions where the flow is always from the core to the upper plenum region during the HPCF injection period.

Westinghouse also stated that at the time of the initiation of LPFL injection, the LPFL fluid must fall through approximately 2 meters of a saturated steam environment in the downcomer. This provides ample opportunity to interact with and condense the steam. Consequently, the LPFL fluid is expected to be saturated upon mixing with the downcomer fluid. If subcooled LPFL flow is able to penetrate the steam environment and reach the downcomer mixture level, interaction and mixing in the downcomer region is expected to result in thermal equilibrium. Westinghouse also performed a conservative analytic calculation demonstrating that flow rates in the downcomer region are insufficient to entrain LPFL fluid out to the break and bypass core cooling by failing to reach the downcomer mixture level. The staff finds the justifications and explanations provided by Westinghouse to be acceptable. Consequently, the one-dimensional nodalization downcomer is considered acceptable for the ABWR LOCA analysis.

To justify the noding near the break, Westinghouse presented the results of nodalization sensitivity studies performed for the RHR suction line break and the main steam line break (MSLB) scenarios. In both sensitivity studies, the results showed no significant effect on calculated peak cladding temperature. Westinghouse also presented the results of a sensitivity study for more detailed noding in the upper plenum for the HPCF line break case. The upper node [[ ]] For the sensitivity case, the break flow, system mass, and downcomer water level were virtually identical to the base case. The staff finds the results of the sensitivity studies and, consequently, the noding in the vicinity of the break and the ECCS injection points as used in USA7 acceptable.

Westinghouse also clarified that the use of multiple parallel control volumes in GOBLIN representing the average core and the hot channel eliminated the need to use the DRAGON option (RAI-28). The DRAGON option has been used in previous LOCA analysis (for BWR/2 through BWR/6) to provide hot channel predictions based on boundary conditions determined from the GOBLIN simulation. Since the DRAGON option has not been reviewed for ABWR LOCA analysis, the staff imposes a restriction that before there is any use of the DRAGON option in GOBLIN in conjunction with the USA7 EM, appropriate prior approval shall be sought by the applicant from the NRC.

3.2.1.2 Two-Phase Level Tracking Model

As noted earlier, the one-dimensional drift flux model as implemented in USA7 cannot resolve the exact location of the mixture level within a given nodal region. Therefore, finer nodalization is essential to provide better estimation of the nodal void fraction and steam quality near the expected mixture level. Alternately, the two-phase level tracking option that is available in USA7 can be activated to calculate the location of a distinctive mixture level where an abrupt void fraction distribution occurs within a control volume containing a two-phase mixture. The intent of the level tracking model is to capture the time-dependent interaction of the mixture level with flow paths or with ECCS injection within coarse mesh control volumes (Ref. 6).

The GOBLIN code has the capability to specify a series of control volumes in which a two-phase level is calculated and tracked. Supplement 5 states that the two-phase mixture level tracking model is activated throughout the downcomer. The use of the two-phase level tracking model was questioned by the staff:

- To assess the adequacy of two-phase level tracking in USA7, justification is necessary regarding its inclusion or exclusion at various locations (RAI-17).

- Feedwater flow, as well as RCIC and the LPFL injection, is directed into the lower downcomer control volumes in which the two-phase mixture level tracking model is active. From the discussion provided in WCAP-11284-P-A (Ref. 2), it appears that the two-phase level tracking model assumes saturated conditions below the two-phase mixture level interface. However, after the break initiation and before ECCS injection, the downcomer fluid below the mixture level is expected to be subcooled. The mixture level and the interfacial mass and energy exchange calculations under such conditions are not clear (RAI-17S01).

In response to the above concerns, Westinghouse clarified that the two-phase mixture level tracking model is not activated in the lower plenum, core, upper plenum, and steam dome. It was explained that the position of the two-phase mixture level in the downcomer is important since it affects the break flow and system depressurization for both the FWLB and MSLB LOCAs. Furthermore, the results of a sensitivity study showing the effect of activating the two-phase level tracking in the upper plenum on a HPCF line break LOCA also were provided. The results show that the activation of two-phase tracking in the upper plenum results in reduced break flow rate and less inventory loss. These studies indicated some effect on calculated inventory loss. However, no impact was observed on the peak cladding temperature (PCT).

The ABWR model in USA7 uses the same nodalization in the downcomer as previous BWR models. Westinghouse explained that in the two-phase level tracking model, the nodal boundaries change depending upon the elevation of the mixture level. When the mixture level decreases below the bottom of a node, the volume of the vapor region from the lower node is added to the volume of the node above, and the volume of the lower node is decreased to include only the liquid. Westinghouse indicated that in this situation, there is no energy exchange between the steam above the interface and the liquid below the interface. Any water injection above the interface results in homogenization with the two-phase mixture above the interface, with mass and energy transfer addressed through the solution of the conservation equations.

Westinghouse also noted that the mixture level tracking model is not capable of calculating the two-phase mixture level within the core. As a result, the determination of core uncovery is based upon nodal void fractions. However, the nodal void fraction defining whether or not the node is assumed to be uncovered was not provided. During the audit, Westinghouse clarified that the core uncovery is assumed to occur when the nodal void fraction in the first unheated node above the core increases to unity. Using this void fraction criterion, Westinghouse showed a comparison of the GOBLIN predictions to a two-loop test apparatus (TLTA) boil-off test. The mixture level calculated using this criterion was compared against the experimental data. The comparison showed satisfactory prediction of the mixture level.

The staff finds the use of the two-phase level tracking model as it exists in USA7 to be acceptable. The staff also notes that the determination of core uncovery in not essential for the calculation of boiling transition (dryout) and, subsequently, cladding heat-up in USA7. Therefore, core uncovery in USA7 has only qualitative significance. To use a drift flux model to precisely calculate core uncovery in the absence of two-phase level tracking, it is necessary to use finer noding near the top of the core. However, in response to RAI-1, Westinghouse showed that increasing the number of core nodes to more than 25 does not result in better prediction of dryout time in using the FRIGG test data. Therefore, the staff concluded that the modeling of the core region in USA7 is adequate for the proper prediction of PCT in the ABWR.

3.2.1.3 <u>ECCS Injection Flow</u>

The ABWR ECCS is comprised of the RCIC, HPCF system, and LPFL mode of RHR system. The HPCF delivers coolant to the upper plenum. The RCIC and the LPFL inject coolant to the downcomer. The calculation of mixing of ECCS injection flow and the amount of steam condensation caused by ECCS injection are important. The issues related to the nodalization and mixing of the ECCS injection flows are addressed in this safety evaluation report (SER). The injection flow–fluid interaction model is a special model available in GOBLIN to calculate the steam condensation caused by ECCS injection flow. Since Supplement 5 lacks detailed discussion on modeling of ECCS injection flow; the staff requested additional information in RAI-13 and RAI-14.

In response, Westinghouse clarified that since the two-phase tracking model is not active in the upper plenum region of USA7, the injection flow–fluid interaction model cannot be used for HPCF injection flow. [[ ]] This modeling approach was justified by noting that there is always upward flow from the reactor core to upper plenum. However, the staff finds this explanation acceptable only if the flow at the core exit is in upward direction during the HPCF injection period. If in the future applications of USA7 it is observed that the flow at core exit is in downward direction during the HPCF injection period, the assumptions of one-dimensional nodalization, no two-phase level tracking, and no use of "injection flow–fluid interaction model for the HPCF injection in the upper plenum should be reconsidered. Therefore, the staff imposes limitation on the use of one-dimensional nodalization in upper-plenum without the activation two-phase level tracking and injection flow-fluid interaction model, unless it is demonstrated that the flow at core exit is in upward direction during the HPCF injection period.

Westinghouse also clarified that the injection flow–fluid interaction model is not activated for the LPFL injection in the downcomer.

It was noted that at the time of the initiation of LPFL injection, the LPFL fluid must fall through approximately 2 meters of a saturated steam environment, providing ample opportunity to realistically interact with the steam environment and condense steam. Consequently, LPFL fluid is expected to saturate upon mixing with the downcomer fluid. If subcooled LPFL flow is able to penetrate the steam environment and reach the downcomer mixture level, interaction and mixing in the downcomer region is expected to result in thermal equilibrium. The staff finds the justifications and explanations provided by Westinghouse to be acceptable.

3.2.1.4 Break Flow and ADS Flow Models

The calculation of break flow discharge is essential for the LOCA analysis. Based on the review of the break flow and ADS flow models used in USA7, the staff raised the following reservations:

- Additional details and sensitivity calculations regarding the modeling and effect of longitudinal split type breaks were necessary to determine the compliance with Section 1.C.1.a of Appendix K (RAI-8).

- Activation of the ADS is typically one of the most important features for the recovery from a hypothetical LOCA event. Therefore, the conditions at the ADS valve choke point and the model used to represent the flow through the ADS valves are important. Additional details regarding the model used to determine ADS flow were necessary (RAI-15).

In response to the above concerns, Westinghouse clarified that GOBLIN cannot simulate the characteristic effects of entrainment or vapor-pull-through in a longitudinal split break. One might expect longitudinal split and double-ended guillotine breaks to behave similarly in cases of single phase steam flow (for example, main steam line break). However, if the break location were to contain liquid and gas phases in close proximity (e.g., feedwater line break), then entrainment of one phase into the other could result in different break flow characteristics (also dependent upon whether the break is in the top, bottom, or side of the pipe). Westinghouse stated that since the feedwater sparger nozzles have a diameter of only 1.75 inches (0.0445 meters), the characteristic effects of entrainment or vapor-pull-through in a longitudinal split break are not expected to have a significant effect on the analysis results assuming the flow choking occurs at the nozzles on the feedwater spargers.

To further clarify this issue, Westinghouse presented the results of FWLB sensitivity calculations using GOBLIN for a break location downstream of the feedwater sparger nozzle. The noding was revised to represent the feedwater lines [[ ]], and the break was modeled in the feedwater line node connected to the downcomer fluid node. It was shown that when the break area is smaller than 50 percent of the feedwater line flow area, choking would occur at the break rather than at the feedwater sparger nozzles. Therefore, the staff restricts the use of USA7 for modeling of longitudinal FWLB, unless it can be demonstrated that the break area is greater than 50 percent of the feedwater line flow area.

In response to questions about the ADS modeling, Westinghouse clarified that the flow area is determined based on the safety valve design flow conditions using a homogeneous equilibrium model (HEM) for critical flow. Upon ADS actuation, flow through the valve is based on the upstream (steam line) stagnation pressure, stagnation enthalpy, ADS flow area, and the HEM critical flow model.

According to 10 CFR Part 50, Appendix K (Ref. 10), the Moody model is to be used after the discharging fluid through the postulated break has been calculated to be two-phase. It is worth noting that since the flow through the ADS valves will be choked in the same manner as the flow through the postulated break in a LOCA, the ADS flow calculated using the HEM critical flow model would be different than that calculated using the Moody critical flow model if two-phase discharge occurs through the ADS. Westinghouse provided a comparison between the HEM and the Moody critical flow to show that the two models show very similar results, with HEM predicting a slightly lower flow as compared with Moody's critical flow model (which is conservative in terms of ADS flow). Also, Westinghouse informed the staff that validation studies for HEM using FIX-II data have been performed. The use of HEM results in a slightly lower discharge flow through the ADS valves, which would result in a slightly slower depressurization rate.

The staff finds the break flow and ADS flow modeling in USA7 to be acceptable with the following restriction:

- GOBLIN cannot simulate the characteristic effects of entrainment or vapor-pull-through in a longitudinal split break. Consequently, GOBLIN is not applicable for modeling longitudinal breaks. Therefore, the modeling of longitudinal feedwater line breaks is restricted to a break area demonstrated to be greater than 50 percent of the feedwater line flow area.

3.2.1.5 Other Conservative Assumptions in USA7

Certain modeling assumptions used in USA7— including feedwater flow isolation, reactor internal pump (RIP) operation, reactor power, and power profile—were evaluated by the staff. The staff expressed the following concerns regarding these assumptions in USA7:

- The feedwater flow is assumed to terminate in 1 second after the initiation of a break (because of loss of offsite power (LOOP) coincident with the LOCA), even though the pump coastdown may take longer because of their inertia. However, sensitivity analyses were not presented to demonstrate that the assumption for feedwater flow termination time is conservative from the standpoint of PCT. Maintaining the design feedwater flow will result in lower enthalpy fluid entering the core until isolation occurs. PCT in ABWR is a function of very early formation and collapse of voids in the reactor coolant rather than long-term coolant inventory makeup. The fast isolation of feedwater may not lead to a conservative estimation of PCT (RAI-2).

- For most of the break spectrum cases documented in Supplement 5, the initial core flow rate showed the greatest impact on PCT. This is because of the effect of the flow rate on early occurrence of boiling transition. The low (90 percent) core flow rate usually yields the highest PCT. However, no justification or sensitivity studies were provided in Supplement 5 for the selected core flow rates (i.e., 90 percent and 111 percent) (RAI-5).

- The ABWR LOCA analysis assumes a LOOP in combination with failure of one emergency diesel generator (EDG). Furthermore, all RIPs were assumed to coastdown rapidly as a result of the LOOP. The rapid reduction in core flow leads to boiling transition and excursion of the fuel cladding temperature. Supplement 5 states that the assumptions of LOOP and RIP trip coincident with the LOCA are conservative. However, sensitivity analyses or justification supporting this assumption are not provided. In the event the RIPs are not immediately tripped, their operation would keep fluid in much of the reactor coolant system homogeneously distributed following a LOCA. This would enhance decay heat removal, but the turbulent mixing of the fluid may result in low quality fluid at the break location for a longer period of time, which may result in lower system inventory. Thereafter, inopportune tripping of the RIPs at this lower system inventory may result in more severe consequences (RAI-10).

- The chopped cosine power distribution is used for the ABWR LOCA analyses. However, since the PCT for ABWR LOCA results from the early power to flow mismatch, it is not unreasonable to expect that a top peaked power distribution may provide a higher PCT (RAI-19).

Westinghouse performed various sensitivity studies in response to the above concerns. Three sensitivity calculations consider feedwater coastdown times of 0.01, 10, and 20 seconds, respectively. The results demonstrate that there is no change in PCT in either case. Therefore, the assumed coastdown time of 1 second for feedwater flow following the initiation of a LOCA is conservative and acceptable.

To justify the initial core flow assumption, the Westinghouse response supplies a power flow operating map for ABWR showing that 102 percent power is designed to be sustainable only in the range of 90 percent to 111 percent core flow. Operation outside of this flow range could be sustained by adjustment of control rods. This response justifies the selected initial core flow values and is acceptable.

Westinghouse performed LOCA analyses for the MSLB, FWLB, and HPCF line break in which RIP operation was continued for 5 seconds and 20 seconds following the initiation of the LOCA, after which the RIPs were tripped. The results of these sensitivity studies showed that, assuming a trip of all RIPs coincident with the LOCA, LOOP coincidence provides conservative results. Any decrease in total system inventory because of continued operation of RIPs is more than compensated for by makeup flow (e.g., feedwater, ECCS). Therefore, continued operation of the RIPs is conservatively bounded by assuming that the RIPs are tripped at the same time as the initiation of the break.

Sensitivity calculations also were performed by Westinghouse for a bottom-skewed and top-skewed power distribution for a typical fuel cycle. The results indicated that the chopped cosine resulted in the highest PCT. The assumption of a chopped cosine axial power distribution is acceptable.

3.2.1.6 Reactor Internal Pump (RIP) Model

The GOBLIN pump model uses the angular momentum conservation equation. The coolant momentum equations and the pump angular momentum equation are coupled in GOBLIN through the four-quadrant pump homologous curves. This pump model is the same as the one previously approved by the NRC. Supplement 5 provides the method for calculating the hydraulic and frictional torque for the RIP model based on the pump design parameters. The frictional torque used in the pump model is a function of the pump speed and is set to a constant value below a specific speed threshold. The pump inertia is adjusted to match the minimum safety analysis limit for the pump coastdown time constant. In Supplement 5, the behavior of the RIP model was compared against the Okiluoto 1 plant startup test data, and it demonstrated acceptable agreement.

In response to a staff question on the effect of the coastdown time on the PCT (RAI-6), Westinghouse performed a sensitivity study using higher pump inertia for the FWLB. The results show that the PCT is slightly lower for the sensitivity case because of the longer coastdown time. The result of the sensitivity is used to show that the model is conservative because of the conservative selection of the minimum coastdown time. Moreover, in the base and sensitivity case, the PCT is predicted to occur before pump speed reaching the value below which a constant frictional torque is applied. Therefore, the value of the constant frictional torque is not expected to affect the PCT. In response to the staff's concerns regarding the pump model behavior for reverse flow conditions and the loss coefficient used for reverse flow through a locked rotor (RAI-6), Westinghouse clarified that the RIP homologous curves that are implemented in USA7 (i.e., GOBLIN) include conditions representing head versus flow characteristics corresponding to a locked rotor (i.e., zero rotor speed). Westinghouse also presented information that indicated that the pump homologous curves were developed based on pump data that included forward and reverse flow through a locked rotor (zero speed). It was also shown that at low flows (less than 25 percent of rated), the absolute value of the pump head was virtually the same for flow in the positive and negative directions, indicating that there would be little resistance for flow reversal through the pumps.

The staff finds that the RIP model in USA7 is acceptable for use in ECCS analysis.

3.2.1.7 Boiling Transition Model

The onset of the boiling transition in GOBLIN is calculated using a boiling length based critical power ratio (CPR) correlation that was developed for the SVEA-96 Optima2 fuel used in the ABWR design. The CPR correlation is developed from steady-state test data collected from the FRIGG loop and includes corrections because of the sub-bundle to full-bundle effect, the double-peaked axial power profile correction, and the R-factor correction. These corrections are documented in WCAP-16081-P-A (Ref. 22).

To demonstrate the ability of GOBLIN to conservatively predict the occurrence of boiling transition, GOBLIN simulations are compared against test data from the FRIGG loop. Eighty five of the 253 FRIGG transient tests are used to validate the GOBLIN dryout prediction capability. The selected test cases include a variety of axial power shapes and power and flow transients, including flow coastdowns as are expected to occur during a LOCA event in the ABWR.

Comparisons of the measured versus predicted dryout times show that GOBLIN predictions are typically conservative as compared to the measurements. All of the flow transient tests were predicted conservatively by the boiling transition model in USA7.

The NRC SER for the previous CPR correlation, which forms the basis of the current one, recommended the use of different uncertainties based on the system pressure (i.e., 3.15 percent below 45 bar and 2.32 percent above 45 bar). However, a single value of the uncertainty is used for the current CPR correlation in GOBLIN. Westinghouse justifies this based on the occurrence of PCT very early in the transient before significant changes in pressure. Moreover, based on the requirements in Section I.C.4.c of 10 CFR Part 50,

Appendix K (Ref. 10), it is not required to use multiple uncertainties since the CPR correlation is based on steady-state experimental data (RAI-20).

Based on the review of the boiling transition model in USA7 and Westinghouse's responses to the staff's concerns, the staff finds the EM to be conservative and acceptable for prediction of boiling transition.

3.2.2 Conformance of USA7 to 10 CFR Part 50, Appendix K

Section I.A. Sources of Heat during the LOCA

Core thermal power assumed for the ABWR LOCA analysis presented in Supplement 5 is 102 percent of the rated thermal power (see Table B-1 of Supplement 5). This assumption is in compliance with the requirement of Appendix K. However, in Section 6.1 of Supplement 5, the applicant also indicated that lower power level may be used in future analyses if it is demonstrated that the uncertainty in power measurement instrumentation is less than 2 percent. The applicant assured that, as required by Appendix K, the power level less than the demonstrated uncertainty would not be used.

The chopped cosine power distribution is used for the ABWR LOCA analyses in Supplement 5. The applicant performed sensitivity studies to determine the effect of power distributions on the PCT. Calculations were performed for a bottom-skewed power distribution and a top-skewed power distribution based on typical fuel cycle burnup. The results indicated that the chopped cosine resulted in the highest PCT (see RAI-19). This satisfies the requirement of Appendix K.

Section I.A.1. The Initial Stored Energy in Fuel

The fuel thermal conductivity model in USA7 accounts for the effects of burnup and temperature and is consistent with the NRC approved methodology (i.e., STAV7.2 code thermo-mechanical fuel rod design code). The fuel data inputs for USA7 describing the initial fuel conditions (e.g., gap size, gap gas compositions, and gap volume) are generated by the STAV7.2 code in a bounding manner, which minimizes gap conductance for the entire burnup range. These considerations result in compliance with Appendix K requirement.

Section I.A.2. Fission Heat

The point kinetic model is used for the calculation of fission power in USA7. The calculations account for the feedback effect from voiding, Doppler broadening, moderator temperature, and control rod. The point kinetics model input parameters (e.g., delayed neutron fraction, void and Doppler feedback coefficient) are conservatively estimated from the NRC-approved methodology (PHOENIX code fuel bundle designs, 2-D transport code).

Section I.A.3. Decay of Actinides

The actinide decay power in USA7 is determined from the decay rate equations described in the American Nuclear Society (ANS) Standard 5.1. The energy release from isotopes uranium-239 and neptunium-239 is accounted for. As required by Appendix K, the actinide production rate was chosen to yield the highest actinide decay power throughout the fuel life.

Section I.A.4. Fission Product Decay

The decay heat in USA7 is calculated by accounting for the decay of 11 groups of fission products. The calculated decay heat is in agreement with the Appendix K proposed 1971 ANS standard. Furthermore, as prescribed by Appendix K, the decay heat multiplier 1.2 is used. The fraction of the locally generated gamma energy that is assumed to be deposited fuel and cladding is justified.

Section I.A.5. Metal-Water Reaction Rate

As proposed in Appendix K, the Baker-Just equation is used to calculate the rates of energy release, hydrogen generation, and zirconium oxidation.

Section I.A.6. Reactor Internal Heat Transfer

USA7 accounts for heat transfer from the piping, vessel walls, and non-fuel internal hardware, which is in compliance with the requirement of Appendix K (see RAI-23).

Section I.B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

The previously approved mechanistic model for calculation of cladding swelling and rupture is used in USA7 [5]. This model has been compared against the applicable data in Reference [4].

Section I.C.1.a. Break Characteristics and Flow

Supplement 5 presented the results of break spectrum sensitivity studies. These sensitivity studies considered double-ended breaks in feedwater line, main steam line, RHR suction and injection lines, HPCF line, and drain line. However, USA7 cannot simulate the characteristic effects of entrainment or vapor-pull-through in a longitudinal split break.

Modeling of a longitudinal break is particularly important if the two-phase flow is expected at the break location (e.g., FWLB and RHR line break). However, as discussed in response to RAI-7, in most break cases the break flow would be choked at the feedwater sparger nozzles. Therefore, the characteristic effects of entrainment or vapor-pull-through in a longitudinal split break would not be expected to have a significant effect on the analysis.

However, additional analysis by the applicant showed that if the break area is smaller than 50 percent of the area of a feedwater line, choking would occur at the break rather than at the feedwater sparger nozzles. Consequently, the reviewers indicated that the SER will impose a restriction on modeling of longitudinal FWLB in GOBLIN, unless it can be demonstrated that the break area is greater than 50 percent of the feedwater line flow area.

Section I.C.1.b. Discharge Model

As required by Appendix K, the Moody model is used in USA7 to calculate the break flow. The break sizes considered in Supplement 5 cover a range from 50 percent to 100 percent. A sufficient range of break sizes is equivalent to use of different break discharge coefficients. The range of break size considered in Supplement 5 more than covers the required range of discharge coefficients from 0.6 to 1.0 (See RAI-8).

Section I.C.1.d. Noding near the Break and the ECCS Injection Points

As discussed in Section 3.2.1.1, the applicant performed several break location and ECCS injection point noding sensitivity studies. The results of these sensitivity studies are discussed in detail in response to RAI-24. Consequently, the noding in the vicinity of break and ECCS injection points as used in USA7 is acceptable.

Section I.C.2. Frictional Pressure Drops

As required by Appendix K, the frictional pressure losses are calculated using models that include a realistic variation of the friction factor with Reynolds number and realistic two-phase friction multipliers that are based on acceptable open literature correlations and test data.

Section I.C.3. Momentum Equation

The momentum equation used in USA7 accounts for all terms specified in Appendix K requirement.

Section I.C.4. Critical Heat Flux

The onset of the boiling transition in USA7 is calculated using a boiling length based critical power ratio (CPR) correlation that was developed for the SVEA-96 Optima2 fuel. This correlation, called D4.1.2, was developed from steady-state test data collected from the FRIGG loop and documented in Reference 1. The correlation has been reviewed and approved by the NRC. See the response to RAI-20 for additional information on this correlation. As discussed in Section 3.2.1.7, that correlation has been benchmarked against the experimental data.

In response RAI-26, the applicant also confirmed that the USA7 has an input option that prevents the reestablishment of nucleate boiling after the first boiling transition occurs. The applicant confirmed that, as required by Appendix K, this option is exercised for the analysis presented in Supplement 5.

Section I.C.5. Post-CHF Heat Transfer Correlations

Post-CHF heat transfer correlations used in USA7 include the Groeneveld flow film boiling correlation specified in Appendix K, the NRC approved Westinghouse upper-head injection correlation, and the Bromley film boiling correlation. 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 to a wide range of tests (Ref. 1).

Section I.C.6. Pump Modeling

The pump behavior in USA7 is modeled using the angular momentum conservation equation. The coolant momentum equations and the pump angular momentum equation are coupled in through the four-quadrant pump homologous curves. Further discussion on the pump model is provided in Section 3.2.1.6. In response to RAI-6, the applicant confirmed that the pump model is the same model that was used in earlier approved submittals, and in earlier applications, such as RELAP4. The two-phase Semiscale test homologous pump data was used as input.

Section I.D.1. Single Failure Criterion

As required by Appendix K, the analyses in Supplement 5 have been performed assuming the single active component failure that results in the most severe consequences. Since no single active failure of ECCS equipment results in an extended uncovery of the ABWR core, the limiting single failure is determined as the one that results in the least transient system inventory.

Section I.D.2. Containment Pressure

USA7 analysis conservatively assumes atmospheric pressure in containment throughout the LOCA. This assumption is in compliance with Appendix K requirements.

Section II. Required Documentation

The reviewers confirmed that Supplement 5, responses to RAIs and questions during the audit and the previously approved model documentation provided sufficient details to permit the technical review of the analytical approach used in USA7. The documentation also provided appropriate sensitivity studies comparisons against the experimental data.

3.3 Accident Scenario Identification Process

A postulated LOCA may be initiated by a break in connecting piping of a wide range of sizes and locations. According to 10 CFR Part 50, Appendix K (Ref. 10), conservative LOCA analysis requires that the worst possible single failure of the ECCS be assumed when demonstrating ECCS performance. A spectrum of pipe breaks sizes, locations, and single failures is necessary in the evaluation of an ECCS performance.

The staff finds that the accident scenarios considered in the ECCS analysis follow the requirements in 10 CFR Part 50, Appendix K (Ref. 10) and are acceptable.

3.4 Code Assessment

The staff performed a detailed review of the ABWR LOCA results that are presented in Supplement 5. The staff also requested additional information on the event sequence timings and the transient plots for several important parameters for the scenarios in Supplement 5 (RAI-32). The review of the information in Supplement 5 and that was provided by Westinghouse in response to staff requests resulted in several important findings, which are summarized here.

High Amplitude Flow Oscillations in the Downcomer

The staff noted unrealistically high amplitude oscillations in the downcomer flow near the feedwater injection node for one of the LOCA scenarios (MSLB6a). Westinghouse explained that when the downcomer wide range (WR) level decreases to the low water level-1 (LWL-1) setpoint, LPFL flow is delivered to the upper downcomer at the feedwater inlet elevation. Injection of LPFL flow into a volume containing vapor or two-phase mixture results in the immediate condensation of vapor and saturation of the LPFL fluid. Under some conditions, injection of the LPFL fluid into a downcomer node may drive the node to a fully saturated liquid or even a subcooled liquid condition. When this occurs, it is possible to trigger high amplitude flow oscillations.

During the audit, Westinghouse stated that this behavior was an artifact of the thermal equilibrium formulation of GOBLIN code. Westinghouse presented additional plots for MSLB6a scenario showing that the flow conditions in the nodes surrounding the LPFL injection node do not propagate to the downcomer exit flow paths (i.e., flow from downcomer to lower plenum do not show these oscillations). Therefore, the staff concludes that the observed condensation induced flow oscillations do not impact the ECCS analysis.

Compensation Factor for Mixture Level Calculation

In examining the mixture level in the downcomer for the drain line break (DLB) analysis, the staff notes that the calculated mixture level is lower than the range available on the WR-level plant instrumentation. Similar behavior also is observed in the FWLB, the MSLB, and the RHR suction and injection line break simulations.

Since the WR-level represents the collapsed liquid level between the reference leg elevation and the WR-level lower tap elevation, the reason for the mixture level to be lower than the WR-level is expected to be because of a compensation factor on the WR-level that was not explained in Supplement 5.

Westinghouse acknowledged that such a compensation factor is indeed used to account for the effect of pressure on measurement of the WR-level. However, the compensation factor is not properly used in the Supplement 5 analyses. As a result of this, the WR-level is calculated to be lower than the mixture level under some low pressure conditions. Westinghouse stated that the ECCS activation is affected by this code input error, but the overall effect on PCT and minimum inventory is not significant. Westinghouse agrees to use the correct compensation factor in future analyses.

The staff finds the explanation provided by Westinghouse acceptable. The use of the correct compensation factor will be verified by the staff in future analyses using USA7.

Void Fraction "Sandwiching" in Plena

In the ABWR design, HPCF flow is delivered to the upper plenum. In USA7, the upper and lower plenums are represented by a series of one-dimensional nodes. The two-phase level tracking model is not activated in these volumes as noted in Section 3.2.1.2 of this SER. The review of responses to RAI-32 revealed that in virtually all of the LOCA cases in Supplement 5, the lowest void fraction node is "sandwiched" between two higher void fraction nodes in the upper and lower plenums.

Responding to the staff's question about the void fraction "sandwiching" behavior in the upper plenum noted in the LOCA analysis results, Westinghouse performed additional calculations (Ref. 21). The calculations demonstrated that the void distribution "sandwiching" in the upper plenum was because of the different flow path areas for the different regions in the upper plenum. The lower void fraction in the middle node of the upper plenum was because of a larger flow area for this node. The larger flow area results in relatively lower superficial vapor velocities and lower void fraction.

Responding to the staff's question about the void fraction "sandwiching" observed in the lower plenum, Westinghouse explained that this was caused by the negative flow through the bypass region that was deposited in the "sandwiched" node.

The staff considers the explanation of the void fraction "sandwiching" in the upper and lower plenums to be acceptable.

3.5 Uncertainty Analyses

Requirements in 10 CFR Part 50, Appendix K (Ref. 10) provide for a conservative analysis that prevents the need for an uncertainty analysis of the type discussed in the SRP (Ref. 8). Westinghouse has chosen to follow the approach outlined in 10 CFR Part 50, Appendix K (Ref. 10).

The staff has confirmed that the methodology used by Westinghouse includes appropriate conservatism in accordance with the requirements in 10 CFR Part 50, Appendix K (Ref. 10). Therefore, Westinghouse is not required to perform an uncertainty analysis.

3.6 Confirmatory Analyses

In support of the staff review of Topical Report WCAP-17116-P, "Westinghouse BWR ECCS Evaluation Model" (Ref. 1), the staff requested the Office of Nuclear Regulatory Research to perform the specific confirmatory analyses using TRAC/RELAP Advanced Computational Engine (TRACE) Simulations of Toshiba Advanced Boiling Water Reactor (ABWR) Loss of Coolant Accident (LOCA) analyses. The specific LOCA events considered were 1) feedwater line break, and 2) high pressure core flooder injection line break. The results of the TRACE calculations (Ref. 24) indicated that the ABWR meets all pertinent regulatory requirements. In all cases, the analyses provided in the ABWR topical report (Ref. 1) bound the TRACE calculation results.

3.7 Quality Assurance Plan

The staff audited the quality assurance (QA) plan in February 2011 (Ref. 25). Subsequent to this audit, the staff concluded that the QA procedures used by the applicant were acceptable.

The staff also has concluded that any changes to GOBLIN are restricted by the methodology (Ref. 1) and that changes to the models in Reference 1 may not be made without NRC review and approval. Changes in numerical methods to improve code convergence or code enhancements or error corrections must be tested, and auditable records must be kept in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

4.0 CONDITIONS AND LIMITATIONS

The staff finds that Westinghouse has adequately demonstrated the compliance of USA7 to the requirements of 10 CFR Part 50, Appendix K. However, the approval of USA7 is subject to the following limitations:

- The modeling of longitudinal feedwater line breaks using USA7 shall be restricted to break areas that are demonstrated to be greater than 50 percent of the cross-sectional flow area of a feedwater line.

- The DRAGON option shall not be used for hot channel analysis using USA7 without prior review and approval by the NRC. LOCA analysis using USA7 will use multiple parallel channels in GOBLIN to represent the average core and hot channel.

- The compensation factor, which accounts for the effect of pressure on measurement of WR-level, should be used in all future analyses using USA7.

- In future analyses using USA7, if downward flow from the upper plenum into core region is predicted during the HPCF injection period, the assumptions one-dimensional nodalization and no two-phase level tracking and injection flow–fluid Interaction models in the upper plenum should be reconsidered. This SER restricts the use of one-dimensional nodalization of upper plenum without the activation two-phase level tracking and injection flow–fluid interaction code option, unless it is demonstrated that the flow at the core exit is in upward direction during the HPCF injection period.

- Changes in numerical methods (see Section 3.7 of this SER) to improve code convergence or code enhancements or error corrections must be tested, and auditable records must be kept in accordance with Appendix B to 10 CFR Part 50.

5.0 CONCLUSION

The applicant submitted Supplement 5 to the Westinghouse BWR ECCS EM (USA7) requesting approval to use the GOBLIN and CHACHA-3D codes for ABWR ECCS analysis. Supplement 5 and the information provided by the applicant in response to RAIs and during the audit were reviewed to determine the compliance of USA7 to the requirements of 10 CFR Part 50, Appendix K.

Since the basic thermal-hydraulic phenomena in ABWR and BWR/2 through BWR/6 are the same, many elements of USA7 which were previously reviewed by the NRC and were found to be applicable for BWR/2 through BWR/6, also are applicable to the ABWR. Therefore, only those aspects of USA7 that are unique for the ABWR or found to be inadequately addressed in Supplement 5 and the previously reviewed model documentation for BWRs were reviewed in this SER. This SER applies to the Toshiba ABWR only.

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Date: January 2013

6.0 REFERENCES

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10. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities, Appendix K, “ECCS Evaluation Models.”

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