FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-2328(P)(A), REVISION 0, SUPPLEMENT 1, REVISION 0

"PWR SMALL BREAK LOCA EVALUATION MODEL, S-RELAP5 BASED"

PROJECT NO. 728

1.0 INTRODUCTION

AREVA NP Inc. (AREVA) submitted EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0 (Reference 1), for U.S. Nuclear Regulatory Commission (NRC) staff review and approval for application of the S-RELAP5 thermal-hydraulic analysis computer code to the Small Break loss-of-coolant accident (SBLOCA) in Westinghouse Electric Company (Westinghouse) and Combustion Engineering (CE) pressurized water reactors (PWRs). AREVA replaced the previously NRC approved methodology using the ANF-RELAP, RODEX2, and TOODEE2 codes for the SBLOCA analysis with only two codes, S-RELAP5 and RODEX2A. NRC approval for this previous change was given in March 2001 with the basis presented in Reference 2. Modifications have been made to these methodologies and incorporated in accordance with the annual Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 reports by AREVA. As such, this supplement to EMF-2328 (P)(A), Revision 0, of Reference 3 provides additional modeling information regarding the manner in which the SBLOCA evaluation model (EM) will treat the following eight areas:

1) Spectrum of break sizes,
2) Core bypass flow paths in the reactor vessel,
3) Reactivity feedback,
4) Delayed reactor coolant pump (RCP) trip,
5) Maximum accumulator/Safety Injection Tank (SIT) temperature,
6) Loop seal clearing,
7) Break in attached piping,
8) Core nodalization.

These changes are intended to improve the rigor and completeness of the original methodology, while also addressing and resolving several staff issues raised regarding the AREVA small-break methodology over the last several years.

2.0 REGULATORY EVALUATION

This application, with the eight modifications, is submitted for review and is intended to satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The NRC staff reviewed the eight changes listed above to EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, in accordance with the requirements of 10 CFR 50.46, Appendix K, and developed Request for Additional Information (RAI) questions that were transmitted to AREVA in Reference 4.

The NRC staff review of the eight changes and responses to the RAI questions are discussed in the following sections.

3.0 RELAP5 CODE BACKGROUND

The RELAP5 computer code is a light-water reactor transient analysis code developed for the NRC for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for nuclear power plant analyses. RELAP5 is a general purpose code that, in addition to calculating the behavior of a reactor coolant system (RCS) during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable gas, and solutes. The RELAP5 code is based on a nonhomogeneous and non-equilibrium model for the two-phase system. The solution technique is by a partially implicit numerical scheme to permit economical calculation of system transients. The objective of the RELAP5 development effort was to produce a code that included important first-order effects necessary for accurate prediction of system transients that was sufficiently simple and cost effective so that parametric or sensitivity studies were possible.

The code includes many generic component models from which general systems can be simulated. These component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control and trip system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, counter-current flow limiting (CCFL), boron tracking, and non-condensable gas transport. The code also incorporates many user conveniences such as extensive input checking, free-form input, internal plot capability, restart, renodalization, and variable output edits.

4.0 TECHNICAL EVALUATION

The NRC staff reviewed each of the following changes to EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, to assure the changes meet the requirements set forth in 10 CFR 50, Appendix K. The NRC staff also utilized Standard Review Plan 15.6.5, "Loss-of-coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," as a further guide to support the review of the changes to the S-RELAP5 code. These changes include:

1) Spectrum of break sizes,
2) Core bypass flow paths in the reactor vessel,
3) Reactivity feedback,
4) Delayed reactor coolant pump (RCP) trip,
5) Maximum accumulator/Safety Injection Tank (SIT) temperature,
6) Loop seal clearing,
7) Break in attached piping,
8) Core nodalization.

Each of these changes will be addressed separately below.

4.1 Spectrum of Break Sizes

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]. The NRC staff agrees that this is conservative since **[** **]**.

[**]** up to, and including, the break that represents 10 percent of the cold leg flow area. The break spectrum will be further refined near the potential worst break size displaying the highest peak cladding temperature (PCT) and in the break range where the evolution of the mitigating systems (pumped or passive injection) would determine where the transient temperature is being turned over. **[** **]**. The NRC staff has required that the limiting break be resolved using a finer break spectrum resolution. AREVA has chosen to increment the break sizes by **[** **]** near the limiting small break in the **[** **]** break range. The limiting break in this range will occur for the largest break size that results in a pressure remaining just above the accumulator actuation pressure. To assure that a slightly smaller or larger break size is not as limiting, the **[** **]** break incremental change is assured of capturing the limiting break. The NRC staff agrees that this approach will identify the limiting small break in the **[** **]** diameter range. With this in mind, the NRC staff further requires that the largest small break that depressurizes to a pressure just above the SIT actuation pressure be included in the break spectrum evaluation. The **[** **]** diameter break increment resolution is expected to capture this particular break size, however it is mentioned and emphasized here since it is important to locate this break size since it could be the limiting small break.

[**]**. This change will result in more liquid being held up in the steam generators (SGs), increasing the degree of core uncovery for the larger small breaks, which will also increase PCTs for these breaks. The NRC staff agrees with the change to the hot leg model, as it will produce higher PCTs for the larger small breaks in the spectrum.

AREVA also will now include low-pressure safety injection (LPSI) and low-head safety injection (LHSI) boundary conditions when simulating SBLOCAs. These low pressure systems are now included since the SBLOCA spectrum includes larger small breaks that will activate these pumps. The NRC staff also notes that the LPSI/LHSI head-flow curves used to simulate the flow behavior of these pumps are to be based on surveillance test data with the uncertainty in head and flow appropriately included, as is the currently done for high-pressure safety injection (HPSI) pump modeling. The NRC staff agrees with the inclusion of the low pressure pumps in the loss-of-coolant accident (LOCA) break spectrum simulations.

4.2 Core Bypass Flow Paths in the Reactor Vessel

The S-RELAP5 vessel nodalization now includes **[**

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AREVA further notes that the **[**

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To accommodate the closure of the **[**

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It is important to mention that Westinghouse plants with **[**

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As such, these paths will be included in the analyses of SBLOCA for Westinghouse plant designs. The NRC staff finds the inclusion of such well-defined **[**

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4.3 Reactivity Feedback

AREVA notes that the current S-RELAP5 model includes reactivity feedback from the control rod insertion, only. Moderator feedback is not included since the moderator temperature coefficient (MTC) is typically negative. Excluding negative feedback is clearly conservative when the MTC is negative. However, the NRC staff has required SBLOCA analyses to also include moderator reactivity feedback when the MTC becomes positive. The MTC can become positive at beginning of life conditions, and as such, moderator density feedback should be included, since depressurization can cause positive reactor feedback that increases core power prior to reactor trip. The maximum plausible value of the MTC will be incorporated based on the technical specification maximum allowed positive MTC.

AREVA further states that modeling of fuel temperature reactivity feedback or other negative feedback such as that for void, in accordance with Section 1.A.2 of Appendix K, will be given their minimum calculated values while also including the appropriate uncertainty.

The NRC staff agrees with the modeling changes AREVA will employ when including reactivity feedback from the moderator density and fuel temperatures following all SBLOCAs simulated with S-RELAP5.

4.4 Delayed Reactor Coolant Pump Trip

SBLOCA emergency core cooling system (ECCS) licensing analyses have shown that with the availability of only one HPSI pump, PCT can exceed current licensing limits unless the RCPs are tripped. The continued operation of the RCPs following a SBLOCA has a significant detrimental effect upon core uncovery for certain break sizes. The increased core uncovery is caused by the action of the RCPs that redistribute the coolant inventory within the primary system, which affects reactor coolant vessel hydraulic response. Hot leg breaks are expected to be more limiting when the RCPs continue to operate. Following a hot leg break, the continued operation of the RCPs affects the primary coolant mass distribution in two ways: (1) by displacing liquid from the cold legs toward the reactor vessel; and (2) by pressurizing the upper downcomer region. During the early portion of the event before significant liquid has been lost from the cold legs, these two effects cause a higher two-phase level to be established in the hot side of the system that includes the inner vessel region composed of the lower plenum, core, and upper plenum, plus the hot legs and hot sides of the SGs, than would be possible if the RCPs have been tripped. The increase in liquid mass redistributed to the hot side of the system is lost through the break in the hot leg so that when the voiding in the RCPs caused the loss of the driving head, the level in the inner vessel equilibrates with the downcomer level, producing a deeper core uncovery and higher PCT than would occur had the RCPs been tripped.

Hot leg breaks in the range of 0.05 to 0.2 square feet are expected to be limiting, but is very plant specific. It is further noted that because of modeling techniques and thermal hydraulic assumptions, some thermal hydraulic codes could show cold leg breaks to be more limiting. As such, the NRC staff requires that a range of cold leg and hot leg breaks be evaluated to identify the limiting break size and location. To prevent SBLOCAs from exceeding the criteria limits, the timing for tripping the RCPs during the event must also be identified.

AREVA has agreed to evaluate a spectrum of hot and cold leg breaks to support the RCP trip procedure and determine/verify the trip timing consistent with the Emergency Operating Procedures (EOPs). This spectrum may include a sensitivity on RCP trip time if such is required to support the trip procedure or address an RAI from the NRC staff.

The NRC staff accepts the AREVA proposed evaluation procedure for supporting the plant EOP for RCP trip timing following a LOCA.

4.5 Loop Seal Biasing

The ability of all thermal hydraulic blowdown codes to accurately predict the clearing of liquid from the horizontal and vertical sections of the suction legs or loop seal has been the subject of much concern over the years. The industry thermal hydraulic codes have failed to properly predict the number of loop seals that clear, as well as the amount of residual water remaining in the horizontal section of the piping following the clearing process. Of particular concern is that for break sizes of about 4.0 inches in diameter and smaller, integral test experiments show that only one loop seal will clear and is usually the suction piping in the broken loop. AREVA also provided data demonstrating the number of loop seals that clear versus break size from the BETHSY, ROSA, SEMISCALE, LOBI, and EOS integral test facilities, confirming this approximate break threshold. Moreover, because of the inability of all thermal hydraulic codes to properly capture the correct loop seal clearing thermal hydraulic behavior, the NRC staff

requires that all break sizes less than 3.5 inches in diameter should be biased to allow only one loop seal to clear. To accomplish this objective with S-RELAP5, [

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The number of loop seals that clear following a SBLOCA affects the degree of long term core uncovery and PCT. The more loop seals that clear, the less the loop resistance to steam flow from the core passing through the loop pipe to the break in the cold leg. The lower loop resistance produces a lower upper plenum pressure with which to drive the vapor, produced in the core, through the external loop to the break. This lower upper plenum pressure allows a higher two-phase level in the core during uncovery, since the loop pressure drop controls the static head difference between the liquid level in the downcomer and the two-phase level in the core. With only a single loop seal cleared, the loop resistance during the long term core uncovery period is maximized, which produces a lower level in the core and a more limiting PCT than that for the same break with multiple loop seals cleared.

It is noted that AREVA uses a [

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AREVA also has increased the number of [

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The NRC staff agrees with the changes AREVA has incorporated into the S-RELAP5 model to assure [ ].

The NRC staff will require AREVA to identify the critical break size at and below which only one loop seal is allowed to clear in the analysis submittal. The number of loop seals cleared for all other break sizes should also be identified. Additional information concerning the way that analyses are expected to address loop seal clearing near the switch or critical break size is provided in Attachment 2 of this safety evaluation.

4.6 Maximum Accumulator/SIT and Refueling Water Storage Tank Temperature

AREVA will employ the [

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The maximum accumulator liquid temperature should also be taken to be at its **[** **]**, if one exists. In any case, the assumed accumulator liquid temperature should not be less than the **[** **]**.

AREVA has stated that a **[** **]**.

The NRC staff agrees with the changes AREVA will employ in their SBLOCA analyses to establish a **[** **]**.

4.7 Breaks in Attached Piping

The NRC staff has required the analyses of severed safety injection lines as part of the break spectrum analysis. Breaks in the safety injection lines result in spilling of one accumulator and the portion of pumped injection contributed to this severed line. Of particular importance is that the broken injection line during a SBLOCA will discharge to atmospheric containment pressure. Since the intact injection lines discharge to a much higher RCS pressure, the loss through the broken line will be much higher than that through each of the intact lines, thus starving the amount of liquid delivered to the RCS. With less liquid delivered to the RCS in the intact lines compared to the case where the break is in the loop piping, there is the potential for severed injection line breaks to be limiting. As such, AREVA has included the analysis of severed injection line breaks as part of their break spectrum evaluation. Also, it is noted that AREVA will generate the proper head versus flow curve for flow delivered to the intact and broken lines from the pumped injection from both high and low pressure injection pumps. The accumulator and pumped injection will spill to the containment directly. AREVA will employ atmospheric back pressure in the containment to generate the head versus flow curve to be used in the evaluation of the severed injection line. The injection line of least resistance should be chosen as the severed injection line.

The NRC staff agrees with the approach that AREVA will institute to evaluate severed injection lines.

4.8 Core Nodalization

AREVA has modified the S-RELAP5 core model to increase the number of **[** **]**.

The NRC staff further notes that a review of bundle uncovery and boil-off test data from two-phase level experiments shows that steam superheat does not begin at the two-phase level surface. Inspection of the bundle uncovery data from Thermal Hydraulic Test Facility at Oak Ridge and the G-2 336 rod bundle uncovery test data reveals that the vapor does not begin to super-heat until about 3 inches above the two-phase level. This is due to the unsteady surface level of the two-phase region due to bubbles bursting and splashing droplets upward just above the location of the level.

AREVA also states that **[**

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The NRC staff agrees with the new core modeling changes **[**

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5.0 RESPONSES TO STAFF RAIs

This section contains a brief discussion of some of the key RAI questions and responses submitted by AREVA. The RAI questions were issued in Reference 4, while the responses were documented in References 5 and 6.

5.1 Countercurrent Flow Limit (CCFL) Model

The NRC staff requested additional information regarding the CCFL model employed in S-RELAP5 because this model affects the rate at which liquid drains from the SGs back into the core during an SBLOCA, particularly for the larger small breaks since the steaming rate exiting the core into the hot legs is highest for these breaks. The higher steam velocities can hold-up liquid in the active tube region of the SGs causing additional core uncovery during the latter part of the transient. AREVA documented that they use the Wallis "type" flooding correlation derived from the UPTF Test 11. This correlation applied to the **[**

]. The NRC staff notes that a survey of test data in References 7 and 8 shows that the value for C varies from 0.7 to 1.0, while the slope varies from 0.8 to 1.0. It is noted that the value of C depends mainly on the pipe or tube inlet and exit geometries. From the data in the literature, the use of the intercept of **[ ]** bounds all of the steam water data in single tube tests characteristic of the full range of pressures experienced following SBLOCA conditions.

AREVA compared the correlation to TOPFLOW CCFL data, as well as the small break ROSA-IV Test SB-CL-18, where it was demonstrated that the CCFL model conservatively bounded flooding data over the full range of pressure conditions as well as liquid hold-up and the core uncovery in the ROSA-IV integral SBLOCA test. The S-RELAP5 code calculated PCT for the ROSA small break test SB-CL-18 over-predicted the clad temperatures at all elevations

in the core by upwards of 400 Kelvin (K), owing to the excessive water hold predicted in the SGs. These results justify the applicability of the CCFL modeling, the CCFL correlations and the new hot leg nodalization modeling modifications. Stratification in the hot legs was also properly predicted demonstrating the code predicted the flow regime changes from bubbly to slug flow then to stratified flow and the accompanying countercurrent flow behavior in the horizontal section of the hot legs.

5.2 Core Bypass Studies

A core bypass sensitivity study with the S-RELAP5 code using the ROSA-IV small break test data was also performed by AREVA. The studies demonstrated that the code properly captures the correct reduction in PCT of 40 K when the bypass flow rate between the core and downcomer is increased by 1.0 percent. An increase of 1.8 percent bypass flow rate reduced the calculated PCT by about 180 K. This reduction is consistent with the reduction in PCT when the bypass is increased by this amount observed in the SEMISCALE small break tests, S-LH-1 and S-LH-2. This demonstrates that the S-RELAP5 code displays the proper thermal hydraulic response and attendant sensitivity of PCT to bypass flow rate between the upper plenum and upper downcomer regions of a PWR.

5.3 S-UT-08 Simulation with S-RELAP5

The NRC staff also requested that the SEMISCALE small break test S-UT-08 be simulated since this test contains loop seal hydraulic behavior, water hold-up in the SGs, and extended core uncovery characteristic of phenomena affecting SBLOCA performance. Comparisons of the S-RELAP5 code prediction with the loop seal response and water hold-up in the generators showed that the code captured the core level depression behavior during loop seal clearing, producing an earlier core uncovery and higher clad temperature during the clearing period of the transient. S-RELAP5 also adequately predicted the drainage of liquid from the SGs. However, the long term core uncovery period following loop seal clearing, showed a poor comparison of the S-RELAP5 predicted liquid level with the data. The NRC staff expressed concern regarding the long term level behavioral portion of the simulation, but noted that the SEMISCALE long term behavior of the core liquid is due to characteristics of the system that are not typical of the current generations of PWRs. The NRC staff discussed these non-standard behaviors with AREVA and included the following areas which may improve the simulation of SEMISCALE Test S-UT-8 (this discussion is provided because it is considered instrumental in understanding small break behavior and the non-standard SEMISCALE design characteristics). The items discussed included condensation heat transfer, core rewet model, two-phase friction losses in hot leg, downcomer, simulation of a large small break ROSA-IV test IB-CL-03, simulation of LOFT test L3-6/L8-1, and Westinghouse and CE plant SBLOCA spectrum simulations.

5.3.1 Condensation Heat Transfer

The Akers, Deans, and Crosser condensation correlation may provide an improved model for primary condensation (steam and two-phase regions) that better match condensation data and increase the condensation rate if needed. Note that the SG liquid accumulation may be more dependent on liquid carry-over from the hot legs than on the improved condensation in the tubes. Condensation of bubbles in the two-phase region may also need to be included in the condensation model. Furthermore, use of a homogeneous quality in the SG tubes prior to pump

head degradation may also improve the liquid carry-over through the U-bend to the down side of the SGs, as well.

5.3.2 Core Rewet Model

During uncovery, water draining down into core from the upper plenum during uncovery periods can cause an increase in the steam production that may be needed in the simulation. When there are several cells that are exposed to steam cooling only, it is necessary to ensure the drainage of liquid from the hot legs and SGs can be properly vaporized by the exposed portions of the fuel.

5.3.3 Two-phase Friction Losses in Hot Leg

Two-phase friction losses during countercurrent flow in the hot leg need to account for an additional loss due to interfacial drag in the small diameter pipes. The correlation of Bharathan and Wallis can be used to compute a friction multiplier for the countercurrent flow conditions in the vertical section in the test. For the horizontal part of the hot leg, the Wallis's correlation for stratified countercurrent horizontal flow can be used. These correlations will increase the drain time of the SGs by upwards of several minutes and hopefully improve the prediction.

5.3.4 Downcomer

The downcomer is circular in geometry and not annular and could affect the vapor release rate modeling in this region. Slug flow is expected in the downcomer (unlike that encountered in PWRs) because of the small diameter pipe. As steam rises to the top of the downcomer, the slug flow behavior will displace more liquid into the cold legs, refills the loop seals and delays the clearing process and prolongs the uncovery. Lower liquid levels in the downcomer also result in lower core recovery after loop seal clearing. Lastly, when the path to the lower plenum from the downcomer uncovers, bubbles produced in the lower plenum due to wall heat could be quickly released to the steam region of the vessel to simulate the reverse flow and suction (pushing or forcing) of steam and two-phase into the downcomer. In the downcomer, when bulk steam enters at the bottom, the bubble release rate can be lowered to simulate the slower slug-type passage of steam in the circular cross section of the vertical piping, which carries inventory out of the downcomer.

Factors affecting the core uncovery level depression in S-UT-8 are:

- For depression in level to occur, the upper plenum pressure must be greater than the downcomer pressure during the period the pump suction leg or loop seal region contains liquid.

- The following cause the upper plenum pressure to be higher (except item 5)

  Qualitative effects (high, medium, low impacts)

  1. Higher inventory level in SG uphill side than downhill side (liquid hold-up) because:

     - Upper head draining into upper plenum supplies inventory which is carried into SG uphill side (high)

- o SG downflow side losses inventory faster than uphill side (medium)
- o CCFL in SG slows draining of uphill side (low)
- o More condensate is generated in uphill side of SG (low)

2. Frictional losses in vertical section of hot leg into SG during countercurrent flow (medium) adds to the loop pressure losses (medium)
3. Wall heat in core region (medium)
4. Core rewet steam production (medium)
5. Bypass line provides vent path for steam from upper head to downcomer (low)
6. Equilibrium fluid state in the cold legs and downcomer during HPSI flow (low)

Furthermore, SG liquid holdup is expected to affect core uncovery to lesser extent in Westinghouse and CE designed plants than in SEMISCALE Test S-UT-8 due to geometric differences and unique test conditions. For limiting break sizes, SG liquid holdup precedes boil-off, core uncovery by 400 seconds; conservative modeling of HPSI flow can over shadow the effects of liquid holdup prior to clearing of loop seals. AREVA models minimum HPSI flow for all break sizes including the limiting SBLOCA

For non-limiting break sizes, the impact of liquid hold up is greatest for larger breaks than the limiting break. In order for the larger break sizes to become limiting due to more detailed modeling of liquid holdup, the change on pre-loop seal clearing core uncovery must produce increases in the pre-loop seal clearing core temperature increase by at least 500 degrees Fahrenheit (^{o}F). This has not been the case for both Westinghouse and CE plants. As such, the NRC staff believes that the EM conservatively models core inventory at the start of accumulator actuation, which is expected to overshadow liquid holdup effects prior to loop seal clearing. The models have shown a conservative treatment of core inventory at the start of reflood for the larger small breaks (accumulator actuation), which over shadows liquid holdup effects prior to loop seal clearing.

In view of the above points, the NRC staff believes that a more detailed modeling to address the above non-standard behaviors in order to predict the long term level behavior in SEMISCALE would not be expected to change the limiting break size or limiting PCT from the plant break spectrum analyses. The NRC staff further believes that the addition of these corrections would improve the S-RELAP5 long term level prediction with S-UT-08. However, the intent of the comparison was to investigate the CCFL and flow regime/drainage behavior in the SGs and hot legs plus the ensuing initial core level depression due to loop seal clearing. Comparison with the S-UT-08 data demonstrated that the S-RELAP5 modeling captured these early key hydraulic phenomena.

5.3.5 Simulation of a Large Small Break ROSA-IV test IB-CL-03

The NRC staff requested that a large small break benchmark be simulated with S-RELAP5 to show that the code properly captures loop seal and SG water hold-up behavior, as these larger break sizes could become more limiting during future power uprates. As a result, AREVA benchmarked the S-RELAP5 code against the ROSA-IV test IB-CL-08, which is a 17 percent (of the cold leg area) intermediate size cold leg break.

Results of the comparison showed that the S-RELAP5 code over-predicted the amount of water retained in both SGs. As a consequence, the PCT was also over-predicted by about 50 K owing to the bounding nature of the predicted hold-up of liquid in the SGs. **[**

]. Furthermore, results of this comparison provides validation of the CCFL modeling, CCFL correlations and hot leg modeling techniques discussed in the previous sections.

5.3.6 Simulation of LOFT Test L3-6/L8-1

Since assessment of the impact of RCPs operation on SBLOCA ECCS performance is required, the NRC staff requested confirmation of the S-RELAP5 code ability to simulate a small break with the RCPs running. Comparison of the S-RELAP5 code with LOFT L3-6 and L8-1 (L3-6 includes the early portion of the event with the RCPs running, while L8-1 includes tripping of the RCPs at 2371.4 sec). The importance of this test is to show the code properly captures the more limiting nature of small breaks in the cold leg due to operation of the RCPs, which results in more fluid lost through the break and increased core uncovery when compared to the case with the RCPs tripped at reactor trip after the initiation of the break.

Comparison of the S-RELAP5 prediction with the data showed that the code predicted a PCT of 683 K compared to the test data PCT of 637 K, when the RCPs were tripped and core uncovery ensued. Although the location of the PCT in the S-RELAP5 simulation occurred at a slightly higher elevation, the code still produced a bounding or conservative temperature.

The comparison with the data showed that the code predicted the primary system mass inventory which remained within the upper and lower bounds of the data. Primary pressure was predicted well, along with the fluid densities in the cold legs and vessel. The good agreement with the data verified the ability of the S-RELAP5 code to capture the key behavior governing SBLOCA behavior for breaks with the RCPs operating, including the resulting uncovery when the RCPs are tripped.

5.3.7 Westinghouse and CE Plant Small Break LOCA Spectrum Simulations

AREVA simulated SBLOCA spectrum analyses for Westinghouse 3 and 4 loop plants and a CE design 2x4 loop plant. The analyses determined the PCT for each plant to an accuracy of [ ] in diameter, investigating break sizes from about [ ] down to and including a one-inch diameter break size. The results of the analyses are summarized below:

Plant	Limiting Break Size	PCT	Break size below which only one loop seal clears
W-3 loop	7.60 inch	1735 ^{o}F	3.396 inch
W-4 Loop	8.10 inch	1429 ^{o}F	3.92 inch
CE 2x4 loop	3.50 inch	1831 ^{o}F	3.79 inch

A severed injection line break was also simulated for each of the plant types; however, they were not more limiting than the cold leg breaks identified above.

The differences in the limiting break sizes for the Westinghouse versus the CE design is due to the lower SIT pressure for CE plants compared to those for Westinghouse, as well as differences in the capacity of the HPSI pumped injection systems.

Results of the spectrum evaluations demonstrated the needed level of break size resolution (**[ ]** diameter increments) to properly identify the limiting break size. It is also important to note that break size below which only one loop seal clears is approximately less than 4 inches in diameter, consistent with the behavior of the scaled integral test experimental data findings summarized by AREVA in Reference 1.

6.0 CONCLUSIONS

AREVA has modified the S-RELAP5 based methodology for the purpose of analyzing SBLOCA of the size 10 percent of the cold leg area or less in Westinghouse and CE designed nuclear steam supply systems. Modifications to the methodology included the following eight areas:

1) Spectrum of break sizes,
2) Core bypass flow paths in the reactor vessel,
3) Reactivity feedback,
4) Delayed reactor coolant pump (RCP) trip,
5) Maximum accumulator/SIT temperature,
6) Loop seal clearing,
7) Break in attached piping,
8) Core nodalization.

Each of these modifications above was supported by validation against separate effects tests, as well as several integral system experiments. The validation also demonstrated that the S-RELAP5 conservatively bounded the thermal hydraulic response for the models listed above. Most importantly, PCT was over-predicted owing to the conservative nature of each of the changes in the above eight areas.

The NRC staff mentions that it is necessary for all SBLOCA submittals utilizing the Reference 1 methodology identify the critical break size, at and below which, only one loop seal clears of liquid. Additional information concerning the way that analyses are expected to address loop seal clearing near the switch or critical break size is provided in Attachment 2 of this SE. The NRC staff further requires that the largest small break that depressurizes to a pressure just above the SIT actuation pressure be included in the break spectrum evaluation. The **[ ]** diameter break increment resolution is expected to capture this particular break size, however, it is mentioned and emphasized here since it is important to locate this break size as it could be the limiting small break.

The NRC staff notes that AREVA has addressed and successfully resolved the NRC staff issues with SBLOCA modeling raised over the past several years and accepts the changes to the S-RELAP5 based methodology described in the eight areas listed above. The NRC staff accepts the new modifications to the S-RELAP5 based methodology governing SBLOCA spectrum evaluations for Westinghouse 3-loop, Westinghouse 4-loop, and CE designed nuclear steam supply systems as submitted in Reference 1. These analyses apply to breaks sizes equal to and less than 10 percent of the cold leg piping flow area.

7.0 REFERENCES

1) EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” March 2012. (Non-publicly available; a non-proprietary version is in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML12065A391)

2) NRC letter dated March 15, 2001, “Acceptance for Referencing of Licensing Topical Report EMF-2328 (P), Revision 0, ‘PWR Small Break LOCA Evaluation Model, and S-RELAP5 Based.’” (Non-publicly available; a non-proprietary version is in ADAMS under Accession No. ML010800365)

3) EMF-2328(P)(A) Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” March 2001. (Non-publicly available; a non-proprietary version is in ADAMS under Accession No. ML011410383)

4) Request for Additional Information Regarding AREVA NP Inc. (AREVA) Topical Report (TR) EMF-2328 (P)(A),Revision 0, Supplement 1, Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” letter from J. G. Rowley (USNRC) to P. Salas (AREVA NP, Inc.), March 6, 2014. (Non-publicly available; a non-proprietary version is in ADAMS under Accession Nos. ML14041A437 and ML14042A056)

5) Response to Request for Additional Information Regarding EMF-2328 (P)(A), Revisions 0, Supplement 1, Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” June 6, 2014. (Non-publicly available; a non-proprietary version is in ADAMS under Accession No. ML14161A034)

6) Revised Response to Request for Additional Information Regarding EMF-2328(P)(A), Revisions 0, Supplement 1, Revision 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” October 3, 2014. (Non-publicly available; a non-proprietary version is in ADAMS under Accession No. ML14280A376)

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