AREVA

PWR Small Break LOCA Evaluation Model, S-RELAP5 Based

EMF-2328(NP)(A)
Revision 0
Supplement 1(NP)(A)
Revision 0

Topical Report

September 2015

AREVA Inc.

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EMF-2328(NP)(A)
Revision 0
Supplement 1(NP)(A)
Revision 0

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

September 1, 2015

Mr. Pedro Salas, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR TOPICAL REPORT EMF-2328(P)(A), REVISION 0, SUPPLEMENT 1, REVISION 0, "PWR [PRESSURIZED WATER REACTOR] SMALL BREAK LOCA [LOSS-OF-COOLANT ACCIDENT] EVALUATION MODEL, S-RELAP5 BASED" (TAC NO. ME8227)

Dear Mr. Salas:

By letter dated March 2, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12065A390), AREVA NP Inc. (AREVA) submitted Topical Report (TR) EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. By letter dated April 22, 2015 (ADAMS Accession No. ML15071A351), an NRC draft safety evaluation (SE) regarding our approval of EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, was provided for your review and comment. By letter dated May 11, 2015 (ADAMS Accession No. ML15160A617), AREVA provided comments on the draft SE. The NRC staff's disposition of the AREVA comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that AREVA publish approved proprietary and non-proprietary versions of TR EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, AREVA and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Mirela Gavrilas, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure:
Final Safety Evaluation (Non-Proprietary)

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 on 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

[

]. 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 LOCA break spectrum simulations.

4.2 Core Bypass Flow Paths in the Reactor Vessel

The S-RELAP5 vessel nodalization now includes [

].

AREVA further notes that the [

].

To accommodate the closure of the [

].

It is important to mention that Westinghouse plants with [

].

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 [

]. The NRC staff further notes that

[

].

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, [

].

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 [

].

AREVA also has increased the number of [

].

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.

4.6 Maximum Accumulator/SIT and Refueling Water Storage Tank (RWST) Temperature

AREVA will employ the [

].

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 (THTF) 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 [

].

The NRC staff agrees with the new core modeling changes [

].

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 CFFL 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)
     - SG downflow side losses inventory faster than uphill side (medium)
     - CCFL in SG slows draining of uphill side (low)
     - 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 (°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 10 inches 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 °F	3.396 inch
W-4 Loop	8.10 inch	1429 °F	3.92 inch
CE 2x4 loop	3.50 inch	1831 °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/Safety Injection Tank (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. 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)

7) Hsieh, C. et al, "Countercurrent Air/Water and Steam/Water Flow Above a Perforated Plate," NUREG/CR-1808, November 1980. (ADAMS Accession No. ML091320404)

8) Wallis, G. B., et al, "Counter-current Annular Flow Regimes for Steam and Subcooled Water in a Vertical Tube," Dartmouth College, EPRI NP-1336, January 1980.

Principle Contributor: L. Ward

Date:

AREVA

March 2, 2012
NRC:12:012

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Request for Review and Approval of EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"

Ref. 1: Letter, Pedro Salas (AREVA NP Inc.) to Document Control Desk (NRC), "Pressurized Water Reactor Safety Analysis Licensing Topical Reports," NRC:11:063, June 24, 2011.

AREVA NP Inc. (AREVA) requests the NRC's review and approval of the enclosure, EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based."

During recent licensing reviews, the NRC staff identified various issues with EMF-2328(P)(A), Revision 0. In Reference 1, AREVA informed the NRC staff of our intention to develop a generic resolution to this issue via a supplement to the topical report. The submittal of the enclosed supplement to the referenced topical report fulfills this commitment.

Proprietary and non-proprietary versions of the topical report supplement are enclosed.

AREVA considers some of the material contained in the enclosed document to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure.

In support of the Office of Nuclear Reactor Regulation's prioritization efforts, the prioritization scheme matrix is attached.

If you have any questions related to this submittal, please contact Ms. Gayle F. Elliott, Product Licensing Manager at 434-832-3347 or by e-mail at gayle.elliott@areva.com.

Sincerely,

Pedro Salas, Manager
Corporate Regulatory Affairs
AREVA NP Inc.

cc: H. D. Cruz
Project 728

A002
1007
NRR

T4.12.1
Contract No. NA
WBS: NA

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

July 27, 2012

RECEIVED
AUG 7 2012
By [signature]

NRC-IC-12-025

Mr. Pedro Salas, Manager
Site Operations and Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: ACCEPTANCE FOR REVIEW OF AREVA NP, INC (AREVA) TOPICAL REPORT (TR) EMF-2328(P)(A), REVISION 0, SUPPLEMENT 1, REVISION 0, "PWR [PRESSURIZED WATER REACTOR] SMALL BREAK LOCA [LOSS-OF-COOLANT ACCIDENT] EVALUATION MODEL, S-RELAP5 BASED" (TAC NO. ME8227)

Dear Mr. Salas:

By letter dated March 2, 2012, AREVA submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." The NRC staff has performed an acceptance review of this TR. We have found that the material presented is sufficient to begin our comprehensive review. The NRC staff expects to issue its request for additional information by December 20, 2012, and issue its draft safety evaluation (SE) by November 20, 2013. The NRC staff estimates that the review will require approximately 300 staff hours including project management time. The review schedule milestones and estimated review hours were discussed and agreed upon in a telephone conference between Gayle Elliott, AREVA Product Licensing Manager and the NRC staff on May 18, 2012.

Section 170.21 of Title 10 of the *Code of Federal Regulations* requires that TRs are subject to fees based on the full cost of the review. You did not request a fee waiver; therefore, NRC staff hours will be billed accordingly.

As with all TRs, the SE will be reviewed by the NRC's Office of the General Counsel (OGC) to determine whether it falls within the scope of the Congressional Review Act (CRA). During the course of this review, OGC considers whether any endorsement or acceptance of a TR by the NRC amounts to a rule as defined in the CRA. If this initial review concludes that the SE, with its accompanying TR, may be a rule, the NRC will forward the package to the Office of Management and Budget (OMB) for further review and consideration. Any review by OMB would impact the schedule for the final issuance of the SE.

If you have questions regarding this matter, please contact Holly Cruz at (301) 415-1053.

Sincerely,

John R. Jolicoeur, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

March 6, 2014

Mr. Pedro Salas, Manager
Corporate Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road
P.O. Box 10395
Lynchburg, VA 24506-0935

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: AREVA NP INC. (AREVA) TOPICAL REPORT (TR) EMF-2328(P)(A), REVISION 0, SUPPLEMENT 1, REVISION 0, "PWR [PRESSURIZED WATER REACTOR] SMALL BREAK LOCA [LOSS-OF-COOLANT ACCIDENT] EVALUATION MODEL, S-RELAP5 BASED" (TAC NO. ME8227)

Dear Mr. Salas:

By letter dated March 2, 2012 (Agencywide Documents Access and Management System Accession No. ML12065A390), AREVA submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On February 7, 2014, Gayle Elliott, AREVA Product Licensing Manager, and I agreed that the NRC staff will receive the response to the enclosed Request for Additional Information (RAI) questions on or before June 6, 2014. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1002.

Sincerely,

/RA/

Jonathan G. Rowley, Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosures:
1. RAI questions (Non-Proprietary)
2. RAI questions (Proprietary)

NOTICE: Enclosure 2 transmitted herewith contains Proprietary Information. When separated from Enclosure 2, this transmittal document is decontrolled.

REQUEST FOR ADDITIONAL INFORMATION QUESTIONS

AREVA NP INC. TOPICAL REPORT

EMF-2328(P)(A), REVISION 0, SUPPLEMENT 1, REVISION 0, "PWR SMALL BREAK LOCA EVALUATION MODEL, S-RELAP5 BASED"

The following request for additional information (RAI) questions are based on the U. S. Nuclear Regulatory Commission (NRC) staff review of AREVA NP Inc. (AREVA) Topical Report (TR) EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR [Pressurized Water Reactor] Small Break LOCA [Loss-of-Coolant Accident] Evaluation Model, S-RELAP5 Based." Please note that a general observation is that there were no discussions or presentations of the validation of the proposed changes against integral and separate effects test data. For this reason the RAI questions below, for the given changes, request validation of the new code against appropriate experimental data.

RAI # 1:

1.1 Section 2.3 of the TR describes changes to the hot leg model in S-RELAP5.

a). Please describe the counter current flow limit (CCFL) correlation and coefficient employed in the hot leg and identify the junctions in the hot legs to which the CCFL correlation is applied.

b). Please provide comparison of the new S-RELAP5 model predictions to data (such as ROSA SB-CL-18) and demonstrate that the code correctly captures stratified counter current flow conditions in the hot leg as well as carry over into the steam generators when steam flow conditions are sufficient to reduce or preclude liquid downflow.

c). Identify the steam mass flow rate at which complete liquid carry over is predicted and show that this condition is supported by the UPTF Test 11 conditions and CCFL correlation. Please note that there are other tests in UPTF test series to validate this model and other facilities such as the Transient Two-Phase Flow experiments.

1.2 Justify that **[ ]** volumes are sufficient to capture the full range of conditions in the hot leg, when NUREG/IA-0116 states that **[ ]** volumes are required to properly simulate counter current flow conditions in the hot leg.

a). Present nodal sensitivity studies to show that **[ ]** volumes are adequate.

b). Please show the sensitivity of the code results to inclination angle.

c). Please show the sensitivity of the CCFL correlation coefficients to the prediction of counter current flow (liquid downflow versus steam upflow) and carry over in the hot leg.

1.3 No validation of these or the other method changes in EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, were identified or referenced.

a). Please show the validation of the model and CCFL correlation against the SEMISCALE Test S-UT-8 experiment, which treats water hold-up in the steam generators.
b). Show plots of the fluid levels in the steam generator uphill volumes and hot legs.
c). Show liquid levels in the loop seal regions.
d). Also, present the clad temperature versus time at the peak cladding temperature (PCT) location and core two-phase and liquid levels, and core pressure responses with the test data.
e). Also, please show a plot of the flow regimes in the hot legs for this test.
f). Please demonstrate that loop seal behavior is also captured for this test.

1.4 Please compare the S-RELAP5 predictions to other integral test data such as the larger small breaks in the ROSA IV large scale test facility test series (see SB-CL-14 10 percent break, for example) to further demonstrate and validate the changes to the hot leg model against integral data.

a). Please show that the model correctly simulates hot leg counter current flow behavior for the largest break sizes included in the small break LOCA (SBLOCA) spectrum (i.e., severance of the largest diameter safety injection line at the discharge leg connection).

1.5 Please also show plant calculations for the largest SBLOCA in the cold leg (0.5 - 1.0 square feet (ft^2)).

a). Show with the changes to the hot leg [ ] and CCFL correlation.
b). Show without the changes to the hot leg [ ] and CCFL correlation.
c). Show the clad temperature versus time and hot leg liquid levels as well as the core two-phase steam generator and liquid levels.
d). Show that carry over and stratification are correctly simulated.

1.6 Please describe the CCFL correlation [ ].

a). Show comparisons to appropriate test data to validate these models.

b). Show that the correlation properly limits the counter current flow behavior in these regions.

RAI # 2:

Section 3 of the TR describes the core bypass modeling. Since the S-RELPA5 code will be applied to plants with upper head spray nozzles.

2.1 Please demonstrate through a comparison to test data that S-RELAP5 can properly capture the effect of core bypass on system response following a SBLOCA.

2.2 Show that loop seal hydraulic behavior is properly captured as well as the core two-phase and liquid level responses in the core during uncovery.

2.3 Present data comparison to the clad temperature at the PCT location. Please see NUREG/CR-4438 for a description of these tests and data.

RAI # 3:

3.1 Please demonstrate that the downcomer modeling for the case when the reactor coolant pumps (RCPs) are operating capture the proper hydraulic phenomena.

a). Should the downncomer levels decrease toward the bottom of the downcomer (cross-over to the lower plenum), demonstrate that the model simulates the two-phase flow communication properly between the downcomer and lower plenum as the fluid transitions from a low quality two- phase mixture to vapor.

b). Also, as vapor is pumped into the down comer from the cold legs describe how the code models the penetration of vapor into the downcomer liquid region at the surface and show that entrainment of downcomer liquid as vapor passes from the intact loops is properly accounted for in the model.

3.2 Please justify the applicability of the S-RELAP5 physical models and modeling techniques to SBLOCAs with the RCP running.

a) Please present validation of the model against integral and separate effects test data.

RAI # 4:

Section 7 discusses the approach to loop seal clearing following an SBLOCA. There are concerns when with the limiting break analysis when multiple loop seal clearing behavior results in the suction legs. Typically, PCT is maximized when only the broken loop seal clears due to the increased resistance of vapor flow through only one loop versus multiple venting loops.

[

]

4.1 Please provide an analysis an appropriate sample plant with and without the proposed modification. Show the impact on PCT for these two cases.

4.2 Also, describe any changes to the downcomer model that impacts entrainment of liquid out the break.

4.3 To validate the suggested approach, please show that the scaling approach to set the maximum break size for which only one loop seal clears, predicts test data for loop seal clearing in different facilities (ROSA test facility, as well as SEMISCALE Tests S-07-10 and S-07-10D, for example). Please see EGG-SEMI-5201 for these test data. These tests included loop seal effects and the impact on long term core uncovery.

a). Please provide plots of the loop seal levels as well as the core two-phase and liquid level responses.
b). Also provide the relevant transient plots to demonstrate that the key phenomena, including loop seals completely or partially cleared, are properly simulated.

(The Bethsy facility also provides additional integral data to validate the changes in EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0.)

RAI # 5:

Please provide a plant sample break spectrum analysis with all of the EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, changes included. This could be accomplished by comparing the results of a sample plant SBLOCA spectrum with and without the EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, changes. This should include the severance of the safety injection line, as well.

<u>RAI # 6</u>:

NUREG/IA-0116 documented S-RELAP5 code failures for the UPTF Test 11 for runs 36 – 45, please verify that the coding error has been corrected in the AREVA version and that these tests were properly simulated.

T4.12.2
Contract No: N/A
WBS: N/A

AREVA

June 6, 2014
NRC:14:032

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Response to Request for Additional Information Regarding EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"

Ref. 1: Letter, Pedro Salas (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Review and Approval of EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR Small Break LOCA Evaluation Model, S-RELAP5 Based'," NRC:12:012, March 2, 2012.

Ref. 2: Letter, Jonathan G. Rowley (NRC) to Pedro Salas (AREVA NP, Inc.), "Request for Additional Information Re: AREVA NP Inc. (AREVA) Topical Report (TR) EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR (Pressurized Water Reactor) Small Break LOCA (Loss of Coolant Accident) Evaluation Model, S-RELAP5 Based' (TAC No. ME8227)," March 6, 2014.

In Reference 1, AREVA Inc. (AREVA) requested that the NRC review and approve the topical report EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." In Reference 2, the NRC provided a Request for Additional Information (RAI) regarding this topical report. The response to this RAI is enclosed with this letter.

AREVA considers some of the information contained in the enclosed documents to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the RAI responses are enclosed.

If you have any questions related to this submittal, please contact Gayle Elliott by telephone at (434) 832-3347, or by e-mail at Gayle.Elliott@areva.com.

Sincerely,

David K. White

for Pedro Salas, Director
Regulatory Affairs
AREVA Inc.

cc: J. G. Rowley
Project 728

Enclosures

1. Proprietary Version - NRC:14:032, Attachment A, Response to Request for Additional Information (RAI) Related to EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"
2. Non-proprietary Version - NRC:14:032, Attachment A, Response to Request for Additional Information (RAI) Related to EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"
3. Notarized Affidavit

AREVA

October 3, 2014
NRC:14:055

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Revised Response to Request for Additional Information Regarding EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"

Ref. 1: Letter Pedro Salas (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Review and Approval of EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR Small Break LOCA Evaluation Model, S-RELAP5 Based'," NRC:12:012, March 2, 2012.

Ref. 2: Letter, Jonathan G. Rowley (NRC) to Pedro Salas (AREVA NP, Inc.), "Request for Additional Information Re: AREVA NP Inc. (AREVA) Topical Report (TR) EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR (Pressurized Water Reactor) Small Break LOCA (Loss of Coolant Accident) Evaluation Model, S-RELAP5 Based' (TAC No. ME8227)," March 6, 2014.

Ref. 3: Letter Pedro Salas (AREVA Inc.) to Document Control Desk (NRC), "Response to Request for Additional Information Regarding EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR Small Break LOCA Evaluation Model, S-RELAP5 Based'," NRC:14:032, June 6, 2014

AREVA Inc. (AREVA) requested the NRC's review and approval of the topical report EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based" in Reference 1. The NRC provided a Request for Additional Information (RAI) in Reference 2. AREVA provided a response to this RAI in Reference 3. A revised response to this RAI is enclosed with this letter.

The revised RAI response consists of the following changes:

1. Response to RAI 1 (pages A-10 and A-11): The use of the CCFL correlations has been modified relative to that utilized in the EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0 topical report. This was revised in the Reference 3 response but not called out.
2. Response to RAI 1.3a (page A-36 and A-37): The discussion of the comparison of S-RELAP5 predictions to the measurements from the test S-UT-8 has been modified.
3. Response to RAI 4 (pages A-84 through A-86): The response to RAI 4 has been revised to reflect a change in the loop seal modeling relative to that described in the EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0 topical report and in the Reference 3 response.
4. Response to RAI 5 (pages A-107 through A-192): The response to RAI 5 was revised to reflect the results for the three sample problems using the revised loop seal modeling approach described in the response to RAI 4.

T007
NRR

AREVA INC.

3315 Old Forest Road, Lynchburg, VA 24501
Tel.: 434 832 3000 - www.areva.com

AREVA considers some of the information contained in the enclosed documents to be proprietary. As required by 10 CFR 2.390(b) an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the RAI responses are enclosed.

If you have any questions related to this submittal, please contact Gayle F. Elliott by telephone at 434-832-3347, or by e-mail at Gayle.Elliott@areva.com.

Sincerely,

Pedro Salas, Director
Regulatory Affairs
AREVA Inc.

cc: J. G. Rowley
Project 728

Enclosures:

1. A Proprietary version of Attachment A: "Revised Response to NRC Request for Additional Information (RAI) Related to EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR Small Break LOCA Evaluation Mode, S-RELAP5 Based'."
2. A Non-proprietary version of Attachment A: "Revised Response to NRC Request for Additional Information (RAI) Related to EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, 'PWR Small Break LOCA Evaluation Mode, S-RELAP5 Based'."
3. Notarized Affidavit

ATTACHMENT A

REVISED RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATED TO EMF-2328(P)(A), REVISION 0, SUPPLEMENT 1, REVISION 0, "PWR SMALL BREAK LOCA EVALUATION MODEL, S-RELAP5 BASED"

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RAI 1:

1.1 Section 2.3 of the TR describes changes to the hot leg model in S-RELAP5.

a) Please describe the counter current flow limit (CCFL) correlation and coefficient employed in the hot leg and identify the junction in the hot legs to which the CCFL correlation is applied.

b) Please provide comparison of the new S-RELAP5 model predictions to data (such as ROSA SB-CL-18) and demonstrate that the code correctly captures stratified counter current flow conditions in the hot leg as well as carry over into the steam generators when steam flow conditions are sufficient to reduce or preclude liquid downflow.

c) Identify the steam mass flow rate at which complete liquid carry over is predicted and show that this condition is supported by the UPTF Test 11 conditions and CCFL correlation. Please note that there are other tests in UPTF test series to validate this model and other facilities such as the Transient Two-Phase Flow experiments.

1.2 Justify that [ ] volumes are sufficient to capture the full range of conditions in the hot leg, when NUREG/IA-0116 states that [ ] volumes are required to properly simulate counter current flow conditions in the hot leg.

a) Present nodal sensitivity studies to show that [ ] volumes are adequate.

b) Please show the sensitivity of the code results to inclination angle.

c) Please show the sensitivity of the CCFL correlation coefficients to the prediction of counter current flow (liquid downflow versus steam upflow) and carry over in the hot leg.

1.3 No validation of these or the other method changes in EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, were identified or referenced.

a) Please show the validation of the model and CCFL correlation against the SEMISCALE Test S-UT-8 experiment, which treats water hold-up in the steam generators.

b) Show plots of the fluid levels in the steam generator uphill volumes and hot legs.

c) Show liquid levels in the loop seal regions.

d) Also, present the clad temperature versus time at the peak cladding temperature (PCT) location and core two-phase and liquid levels, and core pressure responses with the test data.

e) Also, please show a plot of the flow regimes in the hot legs for this test.

f) Please demonstrate that loop seal behavior is also capture for this test.

1.4 Please compare the S-RELAP5 predictions to other integral test data such as the larger small breaks in the ROSA IV large scale test facility test series (see SB-CL-14 10 percent break, for example) to further demonstrate and validate the changes to the hot leg model against integral data.

a) Please show that the model correctly simulates hot leg counter current flow behavior for the largest break sizes included in the small break LOCA (SBLOCA) spectrum (i.e., severance of the largest diameter safety injection line at the discharge leg connection).

1.5 Please also show plant calculations for the largest SBLCOA in the cold leg (0.5 – 1.0 square feet (ft^2)).

a) Show with the changes to the hot leg horizontal flow regime map and CCFL correlation.

b) Show without the changes to the hot leg horizontal flow regime map and CCFL correlation.

c) Show the clad temperature versus time and hot leg liquid levels as well as the core two-phase steam generator and liquid levels.

d) Show that carry over and stratification are correctly simulated.

1.6 Please describe the CCFL correlation applied to the inlet of the steam generator inlet plenum and the inlet to the tubes.

a) Show comparisons to appropriate test data to validate these models.

b) Show that the correlation properly limits the counter current flow behavior in these regions.

Response to RAI 1:

RAI 1 is related to the counter current flow limit (CCFL) correlation and the overall behavior in the hot leg. Benchmarks of the code to applicable experiments have been performed to validate the S-RELAP5 model. These benchmarks include ROSA (5 percent cold leg break and 17 percent cold leg break) as well as the SEMISCALE test S-UT-8.

The use of CCFL correlations has been modified relative to that utilized in the EMF-2328(P)(A) Revision 0, Supplement 1, Revision 0 evaluation model. The revised model will add the Wallis CCFL correlation [

]

The CCFL model will also continue to be applied in [

] as was described in Supplement 1. The CCFL model that is applied to [

]

The benchmarks to the ROSA (5 percent cold leg break and 17 percent cold leg break), LOFT, and SEMISCALE test S-UT-8 use this revised CCFL model. The sample problems provided in response to RAI-5 also use this revised CCFL model. The approved version of the topical report, when issued, will be modified to include a Section 2.4 as shown below.

2.4 CCFL

[

]

Response to RAI 1.1:

Response to RAI 1.1a:

The CCFL correlation used in the hot leg is a Wallis-type CCFL model that has been derived from UPTF Test 11. The Wallis-type CCFL model is applied [

] Further discussion of the CCFL modeling incorporated in the Supplement 1 methodology is described in the response to RAI 1.1c.

Response to RAI 1.1b:

A benchmark of ROSA-IV Test SB-CL-18 was performed with the S-RELAP5 code to validate the model. The model was created, utilizing guidance consistent with AREVA's Small Break LOCA topical report, EMF-2328(P)(A) Revision 0, and recommendations proposed in Supplement 1 to this topical report.

Several of the RAIs are related to understanding the application of the CCFL model [

] and the SBLOCA phenomena associated with flow stratification that affects the level of liquid hold-up in the steam generator tubes and drainage to the reactor vessel.

A ROSA-IV S-RELAP5 (SR5) model representing the 5 percent cold leg break test, Test SB-CL-18, was prepared in accordance with the EMF-2328(P)(A), Revision 0 (Reference 2) and Supplement 1 (Reference 1) recommendations. This model is used as the baseline for ROSA-IV facility and modified as necessary to represent benchmarking of other tests. In particular, the Test SB-CL-18 benchmarking observations comparing the test data with S-RELAP5 predictions are presented here to demonstrate acceptable and conservative implementation of the CCFL correlation.

Test SB-CL-18 was one of the SBLOCA experiments conducted in the ROSA-IV large scale test facility (LSTF) in 1988. This test was selected for International Standard Problem 26 (ISP-26) for benchmarking various system computer codes including RELAP5/MOD2 by participating organizations. AREVA has selected this test for benchmarking S-RELAP5 because it experiences the various modes of a small break LOCA transient, from an initial system depressurization followed by pump coastdown and loss of two-phase circulation, reflux boiling, loop seal clearing and core level depression, core boil-off, and finally to accumulator injection and core recovery. This test simulates a break area equivalent to 5 percent of the cross-sectional area of the pump discharge pipe with no pumped emergency core cooling (ECC) injection.

The RELAP5 model for the ISP-26 (Reference 3) program was verified against the system design data provided in Reference 4, and subsequently modified to implement the AREVA SBLOCA methodology features. The S-RELAP5 code is benchmarked using the modeling approach in accordance with the approved EMF-2328(P)(A), Revision 0 and the submitted Supplement 1 (Reference 1) recommendations to demonstrate the analytical capability of the code in predicting the various modes of a SBLOCA transient. The following sections present a description of the ROSA-IV LSTF test facility, test conditions and calculational model, along with a summary of results of the benchmark.

ROSA-IV Test Facility

ROSA-IV LSTF, as shown in Figure 1-1, is a scaled facility representation of a Westinghouse 4-loop PWR plant, with a fluid volume scaling ratio of 1 to 48. The 1:1 elevation scaling of the system is preserved because it has a first-order effect on SBLOCA transients. The core is simulated by 1064 electrically heated rods. The ROSA-IV facility consists of a pressure vessel with two symmetrical loops, each representing two loops of the PWR plant. The pressurizer is connected to the intact loop.

Model Description

The ROSA RELAP5 base model was originally developed for the ISP-26 program. [

]

For the present benchmark, the ROSA RELAP5 model was revised to implement the provisions of the AREVA SBLOCA calculation model consistent with the approved SBLOCA methodology presented in EMF-2328(P)(A), Revision 0 (Reference 2) and the recommendations proposed in Supplement 1 to this methodology (Reference 1). Alterations are primarily in the primary system nodalization in selected regions, as summarized below.

[ ]

An earlier study of the SB-CL-18 experiment performed by AREVA modeled the steam generator tubes with [

]

Results of the Benchmark

The steady-state initial conditions for the study are presented along with the test conditions in Table 1-1, indicating that the key target parameters are met by S-RELAP5. The calculated sequence of major events is presented along with the test data in Table 1-2.

The transient was initiated at time zero by opening the break. The break is modeled in S-RELAP5 by a trip valve that fully opens at time zero. This break causes a flow of subcooled fluid out of the break, resulting in a rapid system depressurization, followed by pump coastdown and the loss of the ability to force two-phase circulation around the system. The system then enters a period of reflux cooling until the pump suction seals clear. Reflux cooling is lost after the pressure in the primary drops below the steam generator pressure, soon after loop seal clearing.

Figure 1-2 shows that S-RELAP5 initially predicts a higher break flow than measured, and then under-predicts the flow until 150 seconds. After about 200 seconds, the correlated break flow, when the flow is high quality steam, approximates the measured flow. The comparison of the correlated primary and secondary pressure with the test data is presented in Figure 1-3. S-RELAP5 estimates the primary system pressure reasonably well until primary pressure drops below secondary pressure shortly after 200 seconds. The code then calculates a faster primary depressurization rate during the 200-350 second period.

Figure 1-4 and Figure 1-5 compare the calculated and measured differential pressure across the upflow tubes of the steam generators in the intact (Loop A) and broken loop (Loop B), respectively. As seen in these figures, when the measured pressure difference drops at approximately 70 seconds, the pumps lose their ability to force flow through the system because of two-phase degradation. S-RELAP5 over-predicts the fluid retained in the tubes until approximately 200 seconds, approximately coincident with the time when the break flow becomes high quality steam.

Figure 1-6 compares the measured and calculated pressure difference across the pump suction upflow in the broken loop (B). This figure indicates that the code does not predict that the loop seal will fully clear during the time that core heatup occurs, although the measurement indicates that the loop seal clears before 150 seconds. Figure 1-7 compares the pump suction upflow differential pressure in the intact loop (A), and indicates that the code does predict that the loop seal will clear, although it is predicted to occur at about 220 seconds compared to the measured value of 140 seconds.

Following the cessation of pump forced flow through the RCS, for both the test and the calculation, the system flow slows, liquid collects in traps around the system and the loop draining phase of small break LOCA is entered. Water suspended within the steam generator (SG) up-flow tubes and reactor coolant pumps (RCP) loop seals will limit the liquid content of the core region.

Figure 1-4 and Figure 1-5 show that from 70 seconds to around 200 seconds the S-RELAP5 drag or CCFL models hold substantially more liquid in the SG up-flow tubes than occurred in the test. That in turn leads to more voided core flows during loop draining and an early cladding heat up starting around 70 seconds, as shown in Figure 1-8. Between 70 and approximately 140 seconds, the core liquid fraction in the test remains high enough to keep the cladding near saturation. During this time period, the code predicts there is significantly more liquid in the loops, principally in the SG tubes and the broken loop pump seal, than was indicated in the test. This observation is supported by the calculated heatup shown in Figure 1-9 through Figure 1-12. At approximately 110 seconds, the liquid in the SG tubes starts to drain back to the core initiating additional cooling. This cooling reduces the cladding temperature at the core exit but only stabilizes or slows the temperature excursion in the middle core regions. This is supported by Figure 1-13 and Figure 1-14, which compare the measured and calculated pressure difference across the core and the downcomer, respectively.

At approximately 200 seconds, the loops have drained sufficiently that the loop seal clearing process initiates. In the test, both loops clear and sufficient liquid is transferred to the core to provide cooling until approximately 420 seconds where a second temperature excursion initiates at the top of the core and progresses downward until accumulator injection provides sufficient liquid to refill the core. In the calculation, however, only the intact loop clears and a substantial amount of water is retained in the broken loop. This leads to a core uncovery and cladding temperature excursion starting at around 220 seconds. Accumulator injection begins much earlier in the calculation, at 340 seconds, and initiates a bottom up cooling of the core at around 360 seconds.

Conclusion

This benchmark of the ROSA-IV SB-CL-18 test shows that a combination of the S-RELAP5 drag model and application of the CCFL model [

] along with the selected coefficients, causes the code to correctly predict the phenomenon of liquid being retained in the steam generators. S-RELAP5 significantly over-predicts this liquid hold-up in the steam generator tubes; drainage to the core is conservatively limited. The significant difference in the cladding temperature excursion between code and test data can be attributed to the code conservatively predicting the retention of water in the loops and outside the vessel. This is a direct result of this conservatism emanating from the underlying approach for the EMF-2328(P)(A), Revision 0 SBLOCA methodology; therefore, validating the deterministic model.

References

1. EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0 "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2012.
2. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.
3. Y. Kukita, et al. "ISP-26 OECD/NEA/CSNI International Standard Problem No.26 ROSA-IV LSTF Cold-Leg Small-Break LOCA Experiment Comparison Report," NEA/CSNI/R(91)13, February 1992.
4. ROSA-IV Group "ROSA-IV Large Scale Test Facility (LSTF) System Description," JAERI-M 84-237, 1985.

Table 1-1: Initial Conditions for ROSA-IV Run SB-CL-18

Parameter	Specified	Measured	Calculated (S-RELAP5)
Pressurizer pressure (MPa)	15.5	15.5	15.6
Hot leg fluid temperature (loop A/B) (K)	598/598	599/599	599.9/599.9
Cold leg fluid temperature (A/B) (K)	562/562	563/564	564.4/564.4
Core power (MW)	10	10	10
Core inlet flow rate (kg/s)	48.6	48.7	48.4
Pressurizer water level (m)	2.7	2.7	2.67
Primary coolant pump speed (A/B) (rpm)	800/800	769/796	788/788
SG secondary pressure (A/B) (MPa)	7.3/7.3	7.3/7.4	7.3/7.3
SG secondary liquid level (A/B) (m)	10.3/10.3	10.8/10.6	9.52/9.52

Table 1-2: Chronology of Events – ROSA-IV Test SB-CL-18

Event	Time (s)	Time (s)
	Test	S-RELAP5
Break	0	0
Scram	9	8.145
SG Feedwater stop	16	15.2
Pressurizer empty	25	30.5
Initial Temperature Excursion	120	70
Loop Seal Clearing	~140	~220*
Primary/Secondary Pressure Reversal	180	~200
Accumulator Injection	455	340
* The broken loop did not clear		

Figure 1-1: General View of LSTF

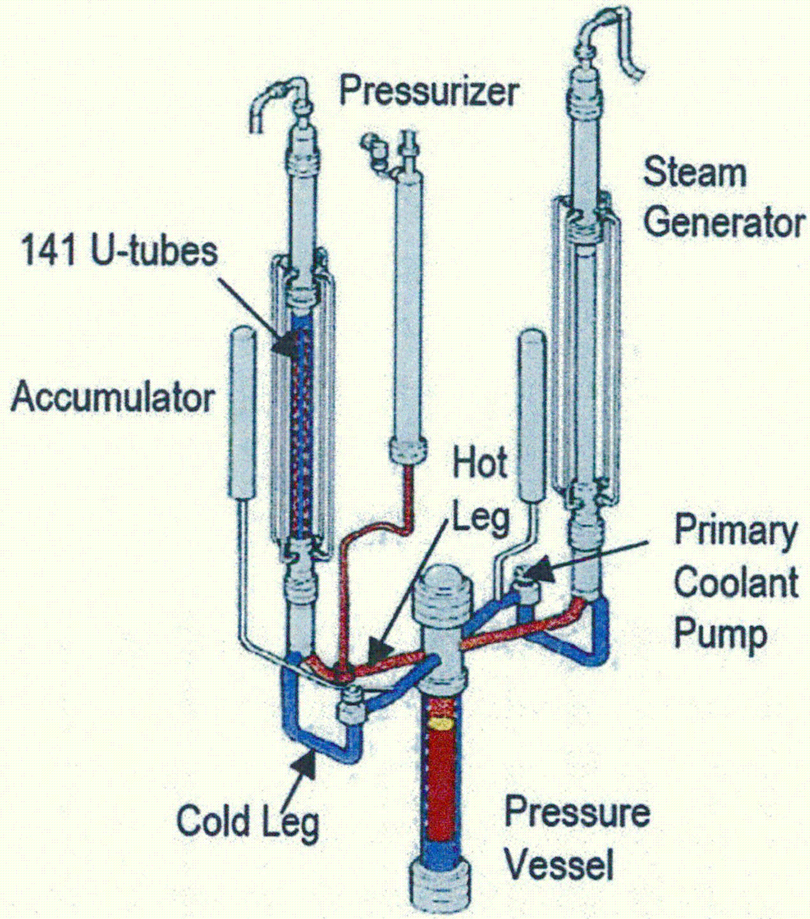


Figure 1-2: Flow Rate through the Break

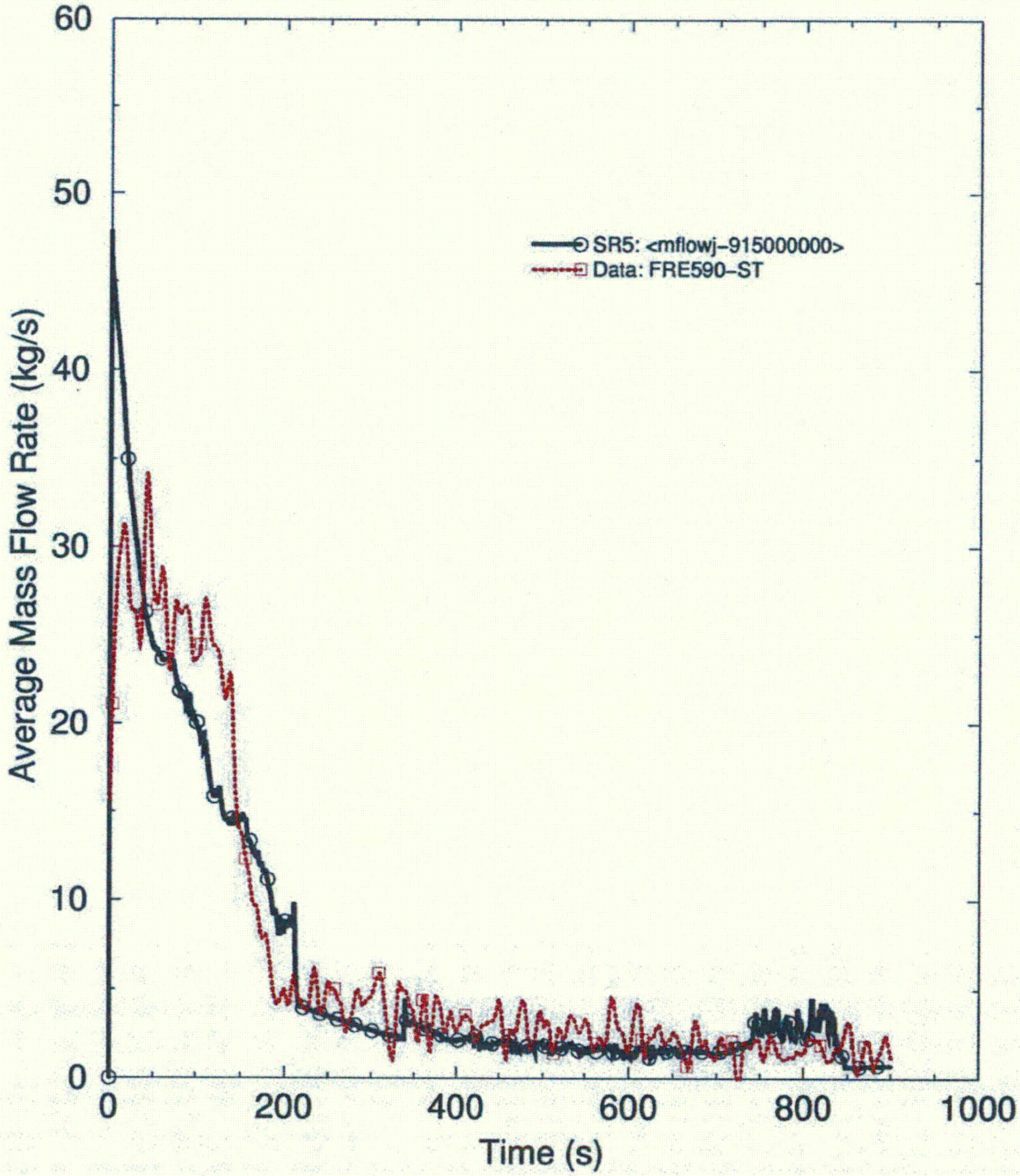


Figure 1-3: System Pressure Response

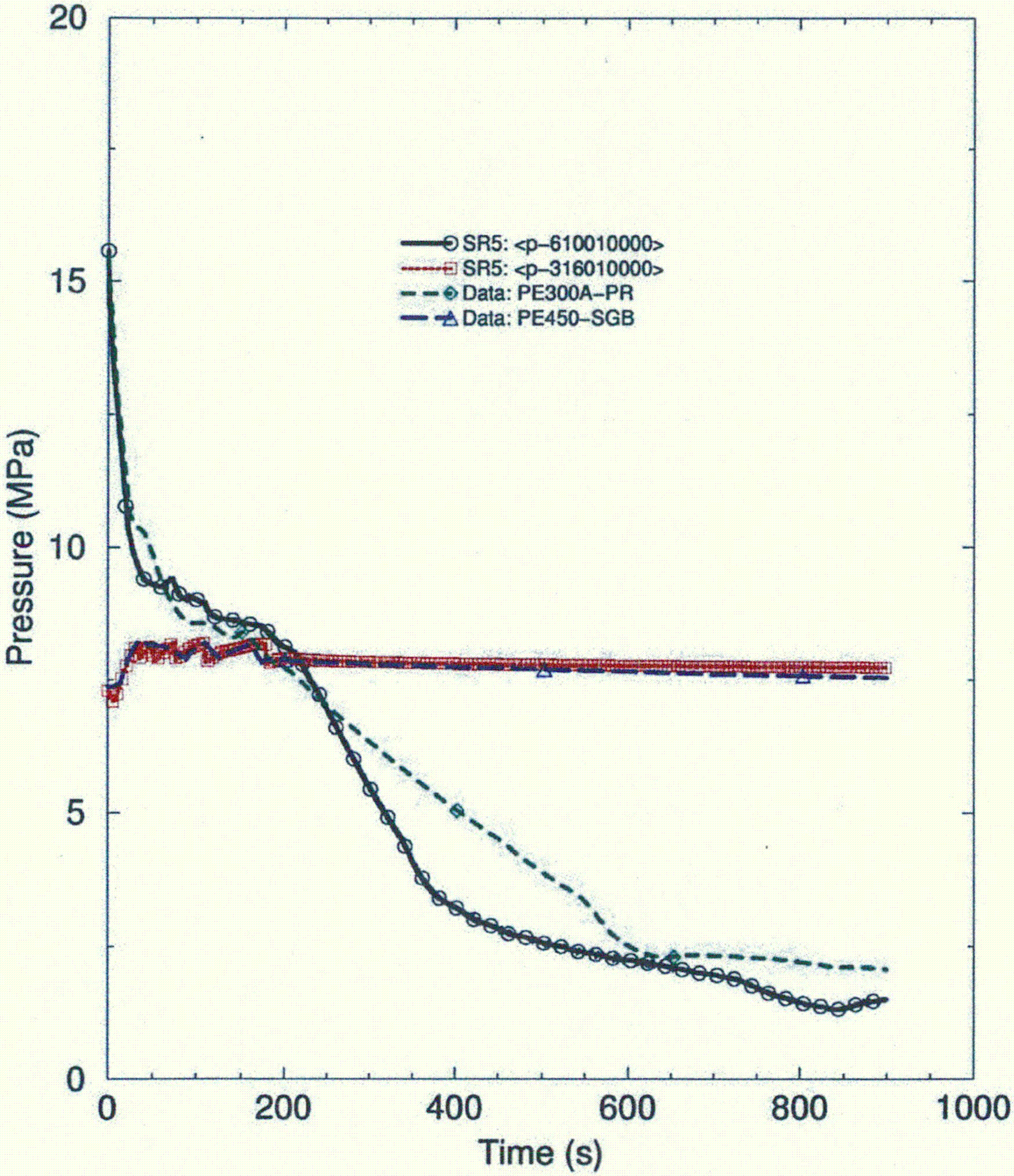


Figure 1-4: Intact Loop Steam Generator-A Tubes Upflow Differential Pressure

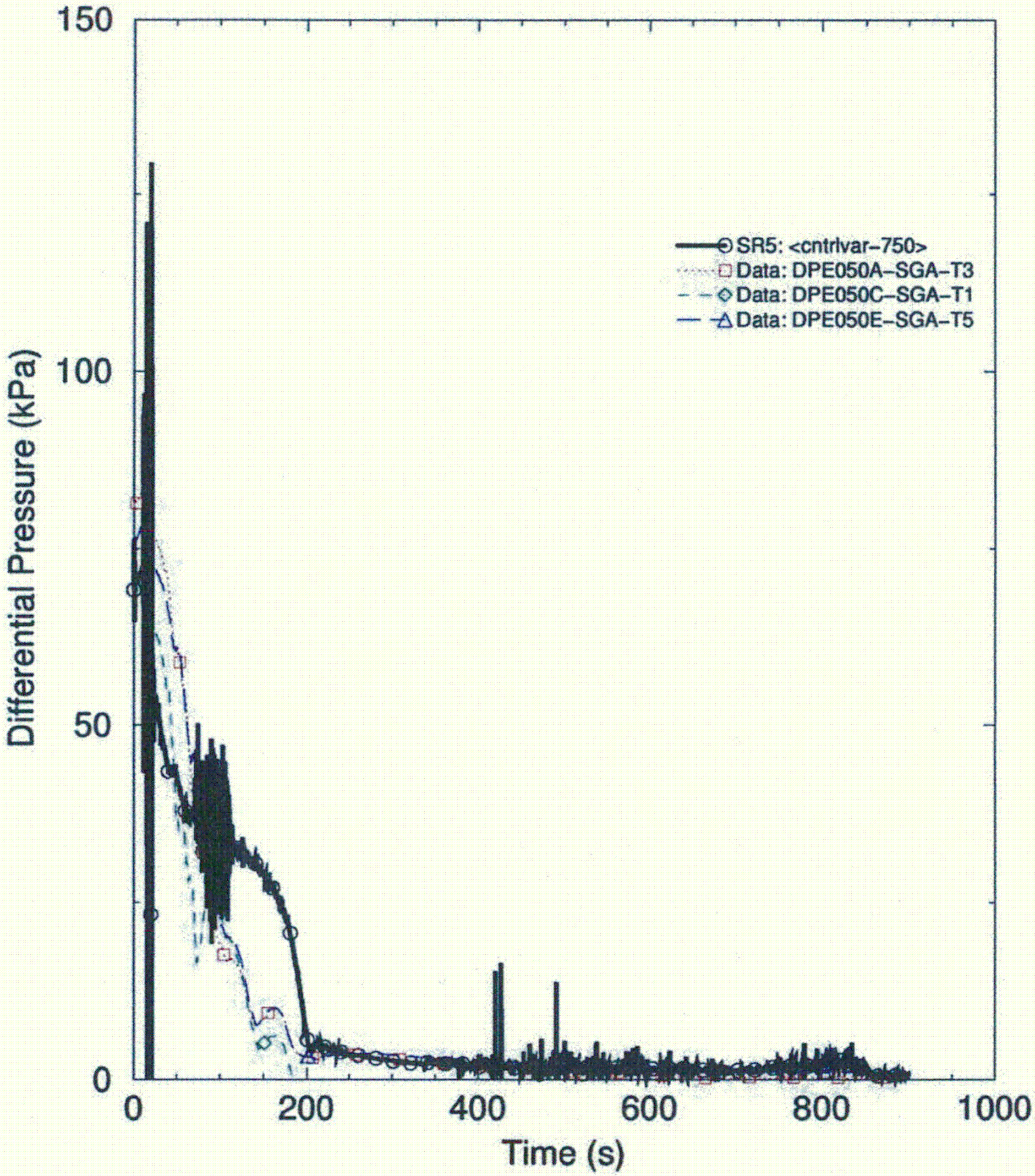


Figure 1-5: Broken Loop Steam Generator-B Tubes Upflow Differential Pressure

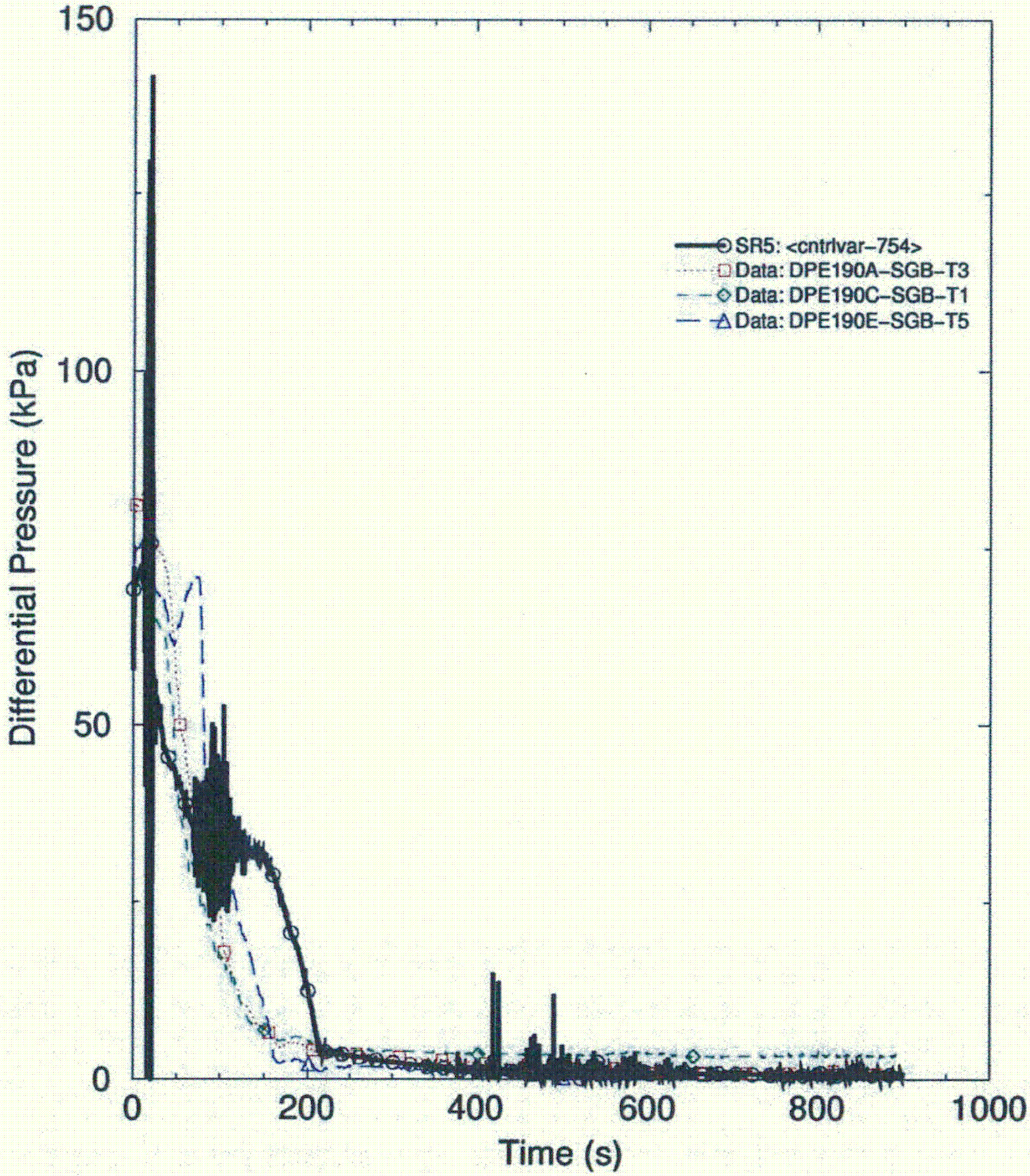


Figure 1-6: Broken Loop Pump Suction Seal Upflow DP

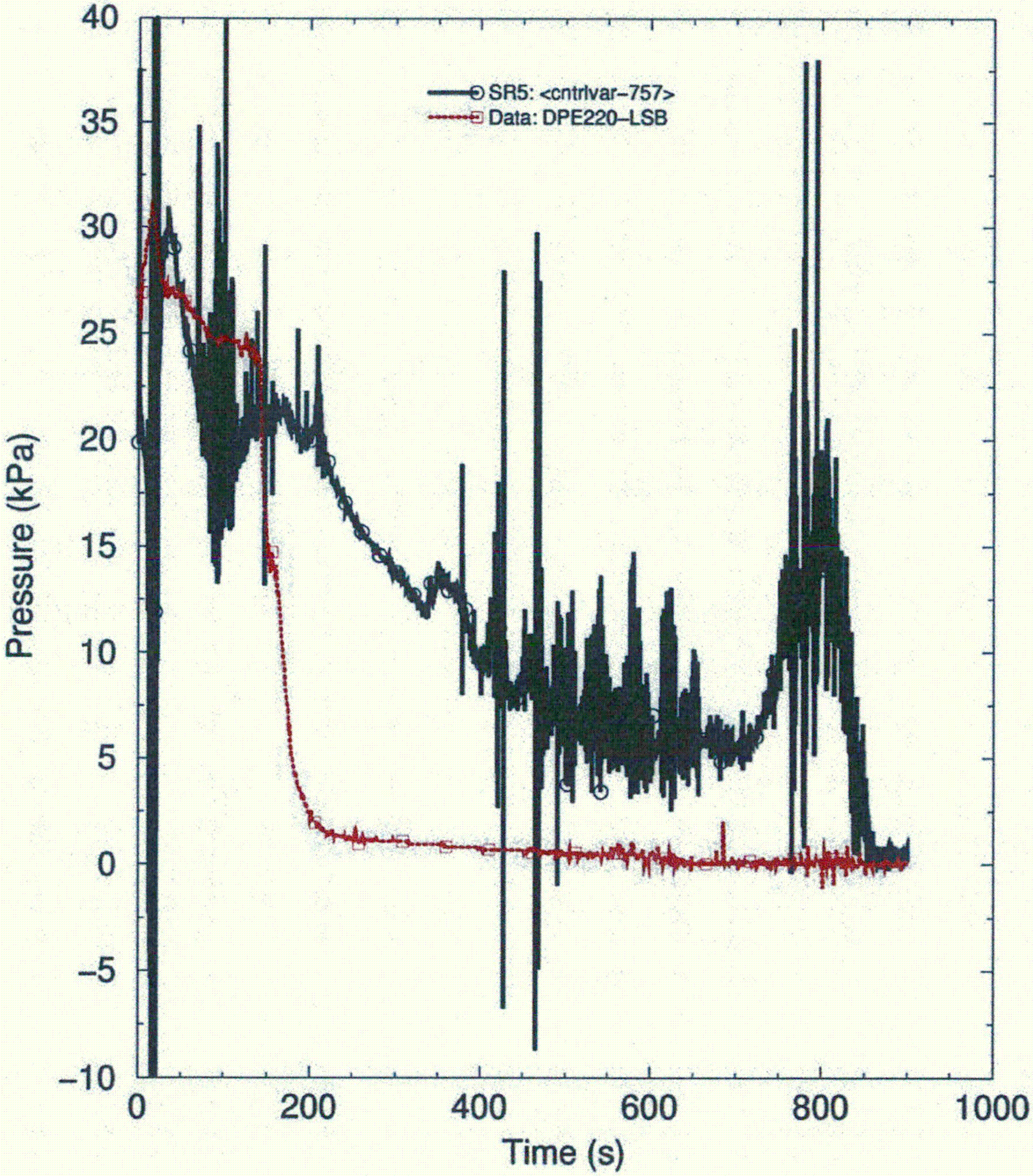


Figure 1-7: Intact Loop Pump Suction Seal Upflow DP

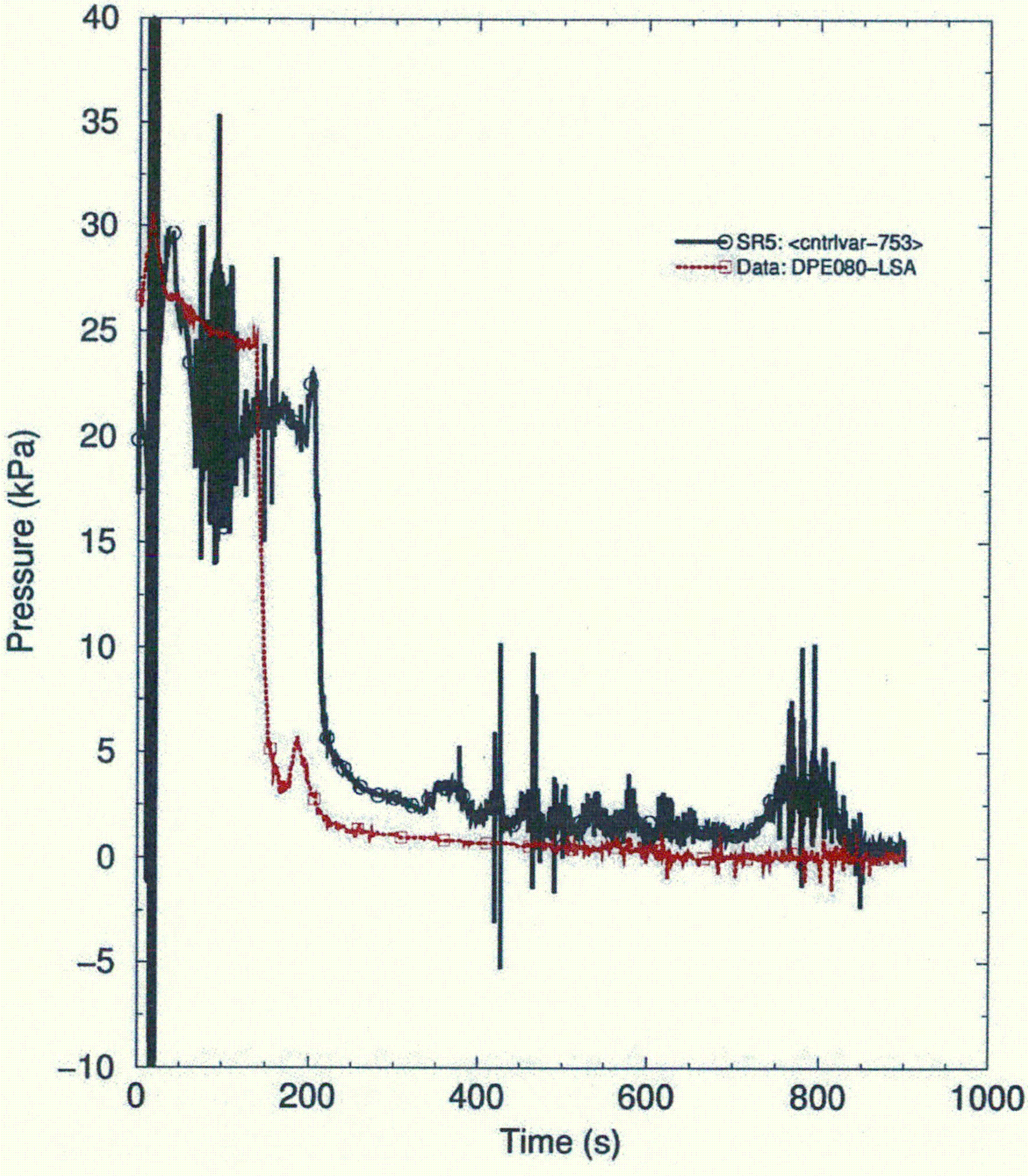


Figure 1-8: Clad Temperature at Elevation = 1.018 m

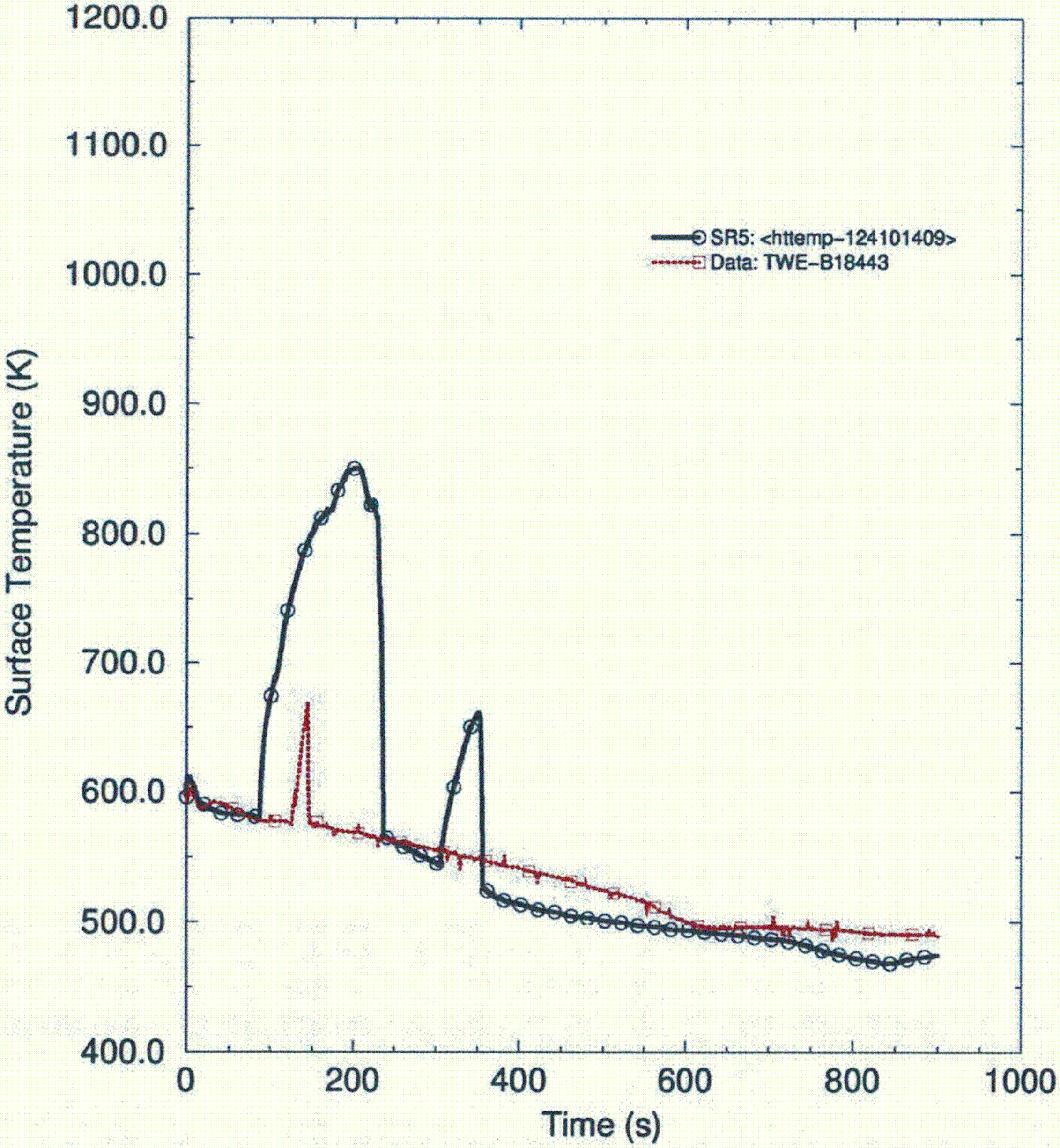


Figure 1-9: Clad Temperature at Elevation = 1.83 m

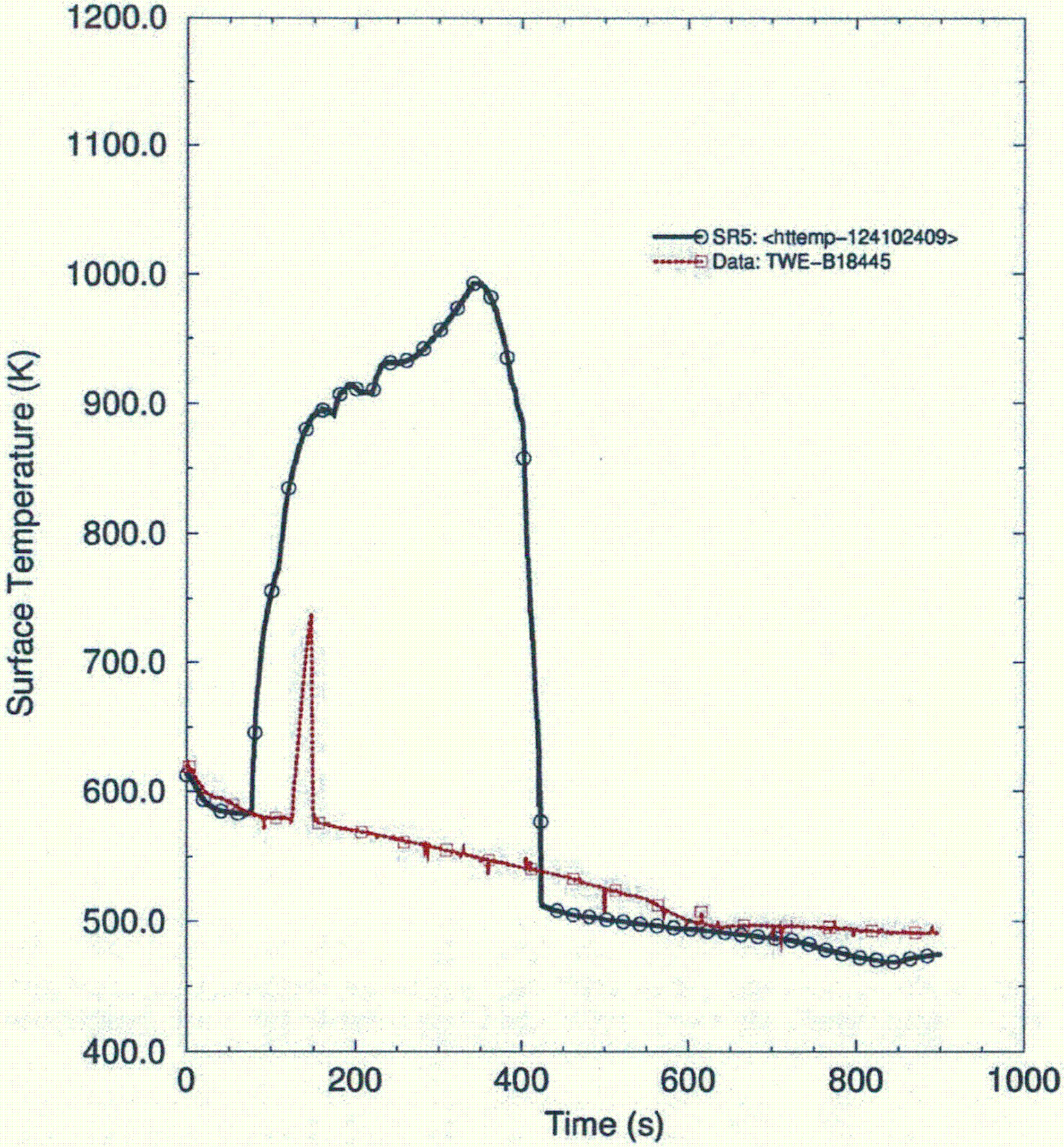


Figure 1-10: Clad Temperature at Elevation = 2.236 m

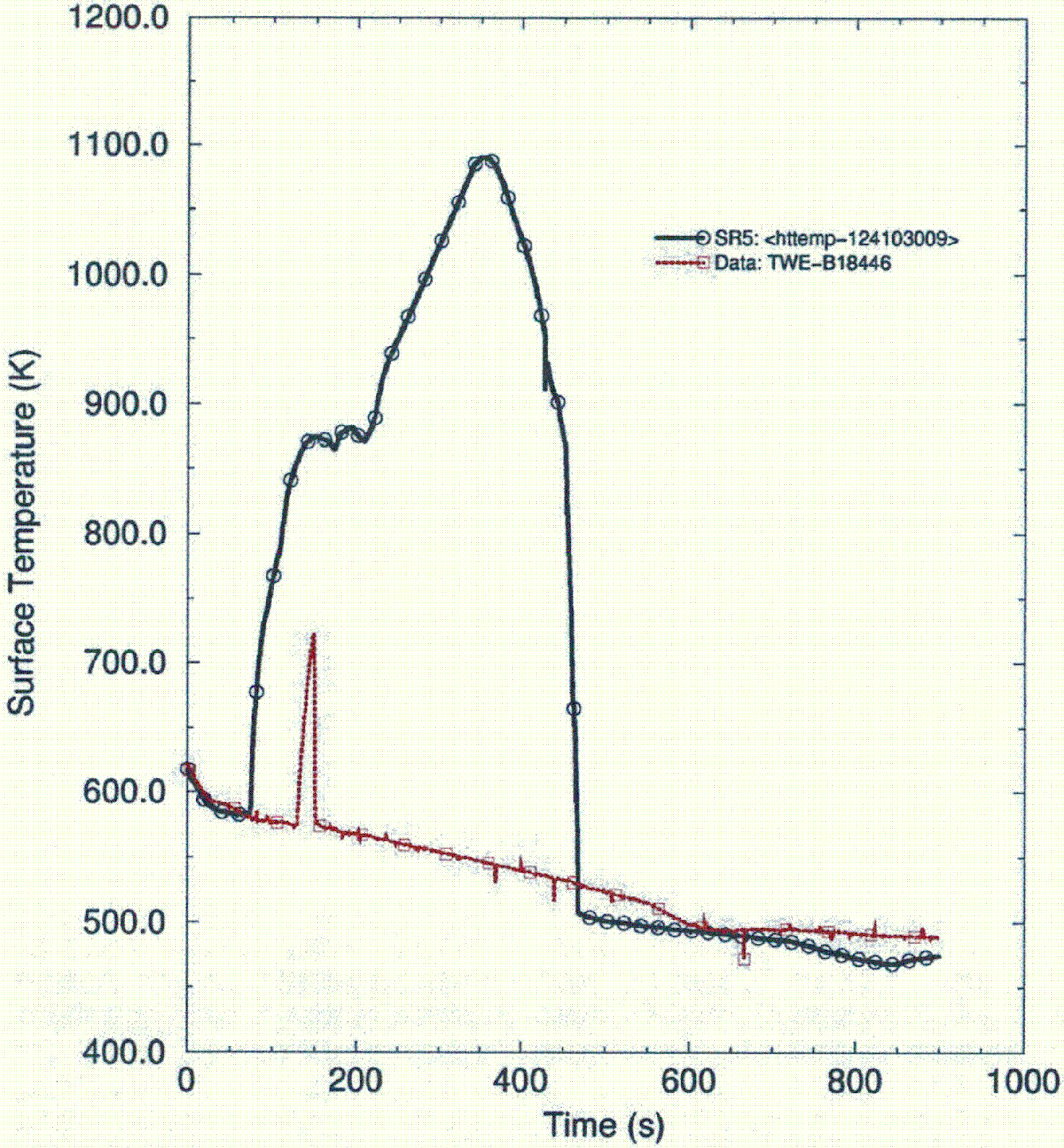


Figure 1-11: Clad Temperature at Elevation = 3.048 m

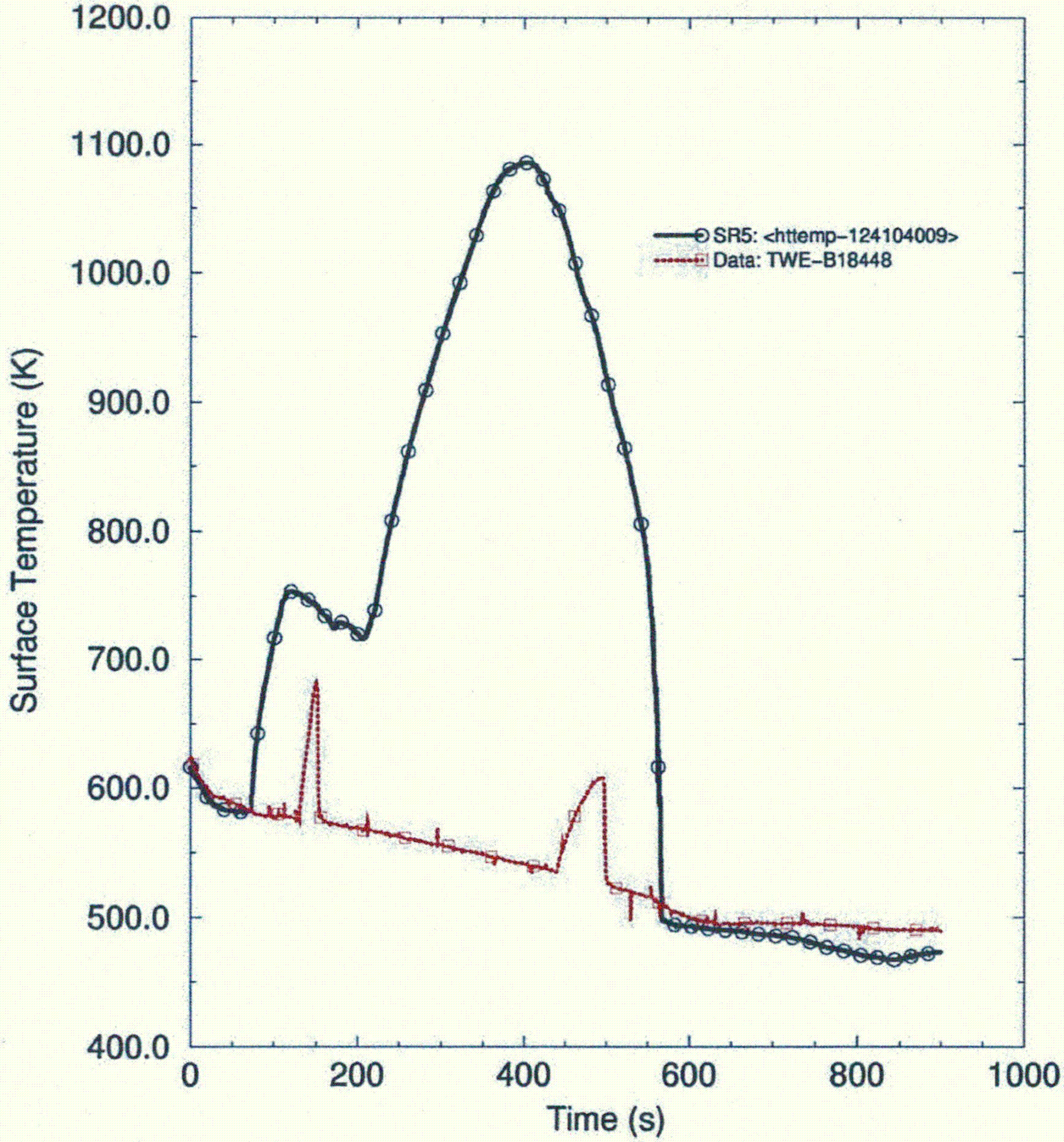


Figure 1-12: Clad Temperature at Elevation = 3.61 m

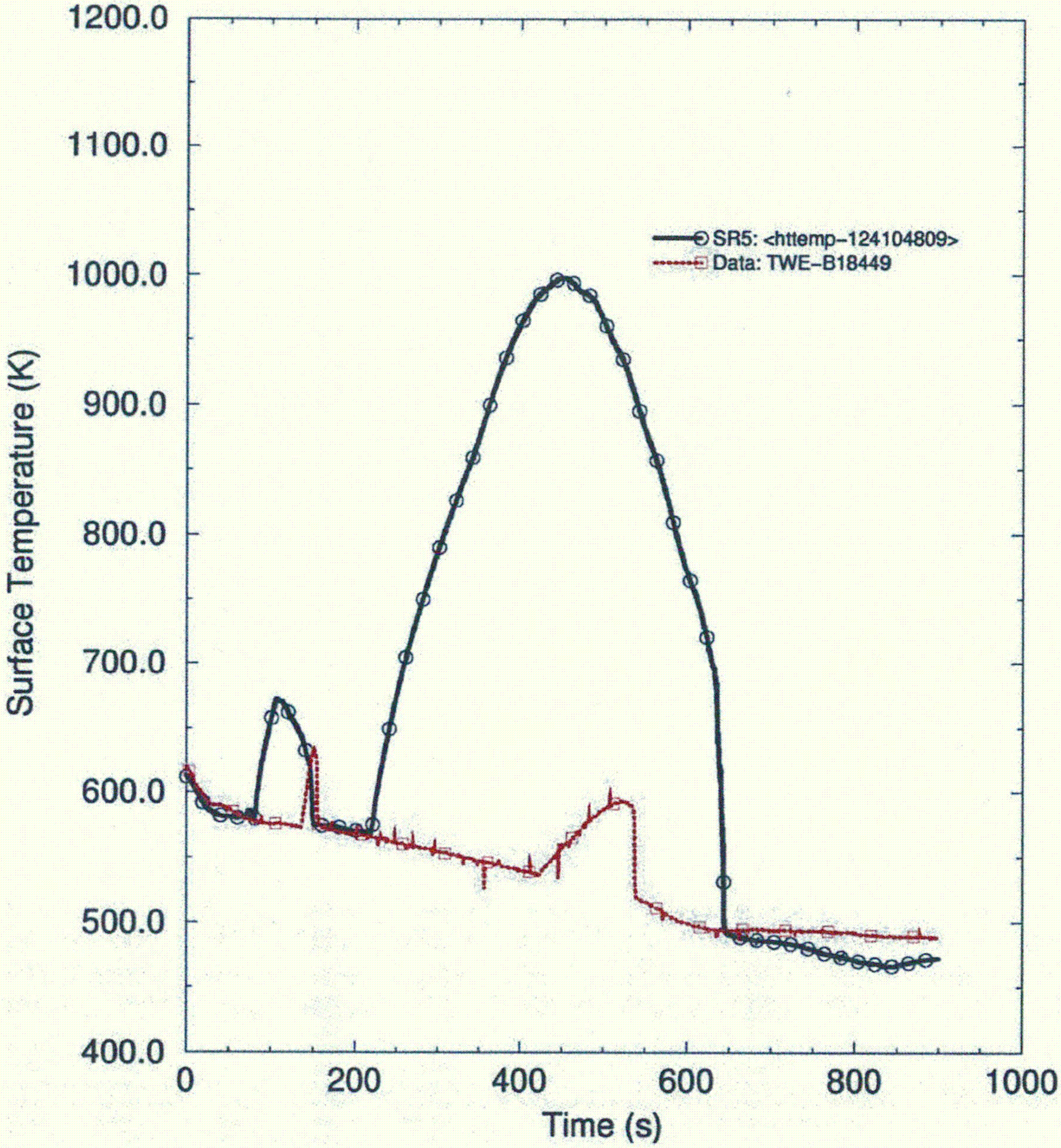


Figure 1-13: Core Differential Pressure

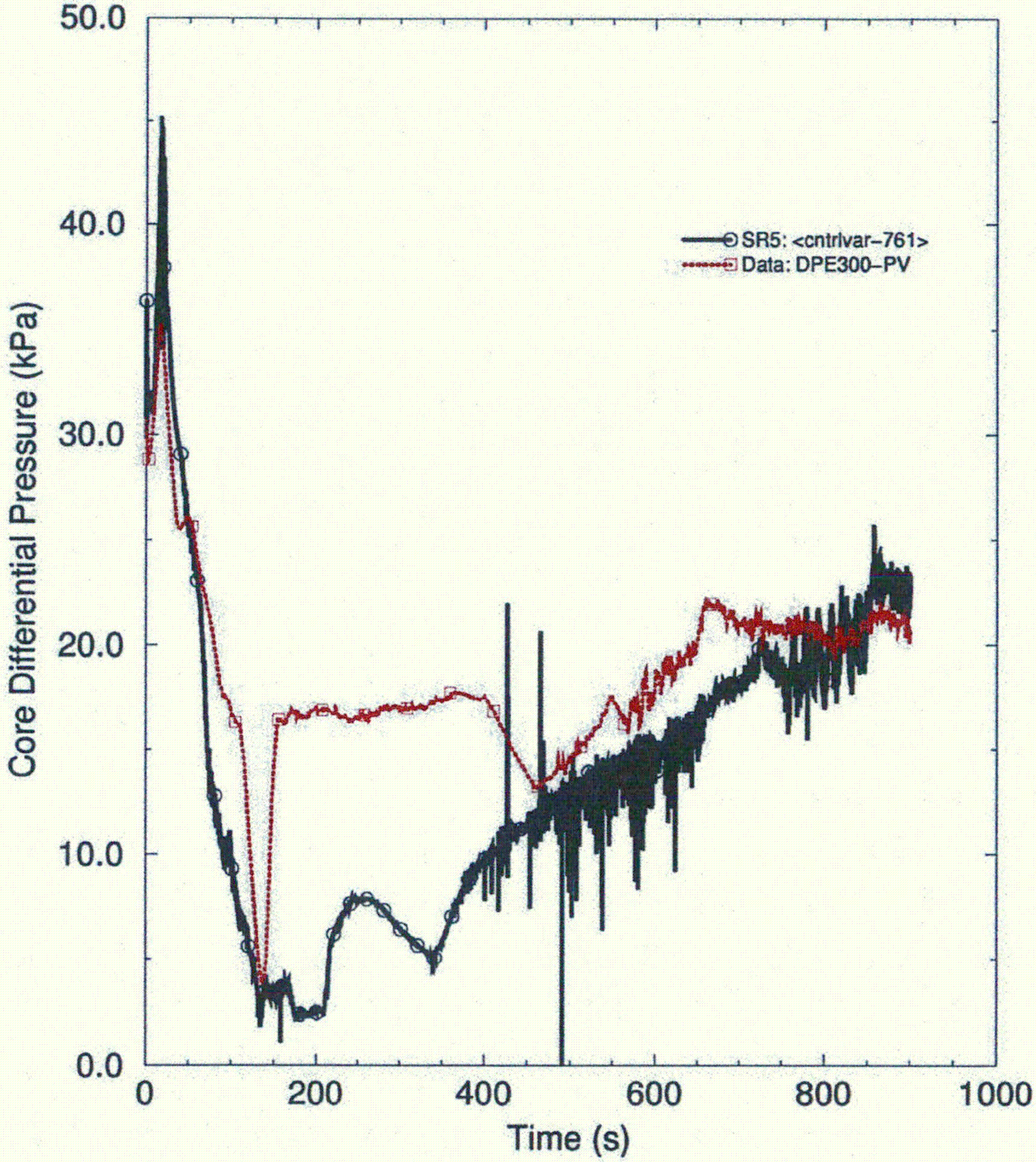
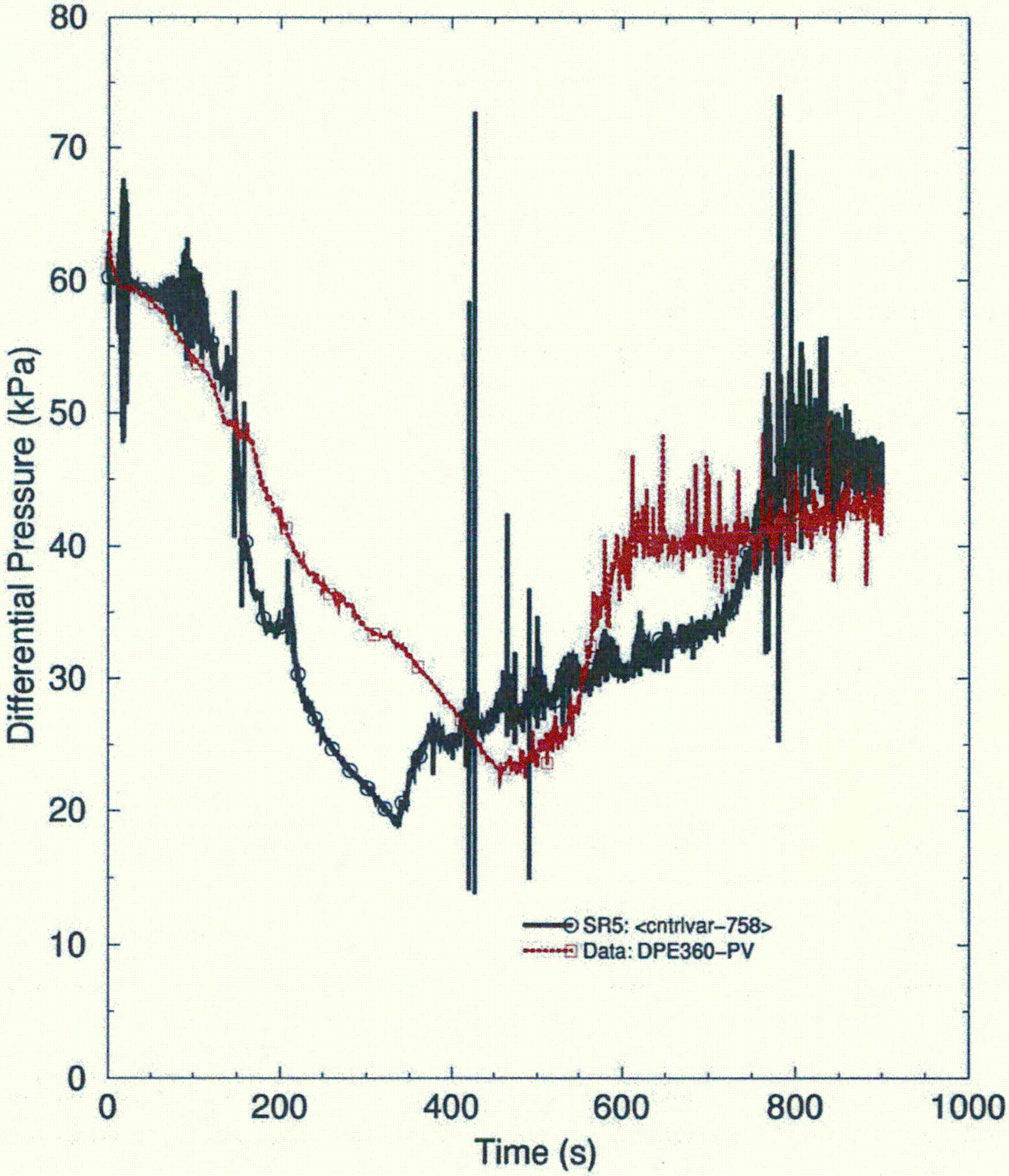


Figure 1-14: Downcomer Differential Pressure



Response to RAI 1.1c:

In S-RELAP5, AREVA has implemented a general from of CCFL model (Bankoff-type) in order to calculate correct counter-current flow at selected locations [ ] The Bankoff-form reduces to Wallis-form of CCFL correlation for β = 0. The implementation of the model is described in Section 5.5 of Reference 1. [

]

In the EMF-2328(P)(A) methodology, AREVA applies the Wallis-form of the CCFL correlation [ ] As described in the response to RAI 1.1a, the correlation coefficients, m and c were developed from the UPTF Test 11 data and the values for [ ] respectively.

$$\sqrt{j_g^*} + m\sqrt{j_f^*} = c$$

[ ]

An assessment of the UPTF Test 11 to validate the CCFL correlation in S-RELAP5 has been performed in the past. The assessment of UPTF Test 11 to validate the Wallis form of the CCFL correlation in S-RELAP5 has been performed by AREVA and can be found in Section 4.3.1.11.5 of Reference 3. The benchmark confirmed that applying the Wallis form of the CCFL model, with [ ] produces liquid downflow rates in reasonably good agreement with data.

[ ]

AREVA has demonstrated that the Wallis form of CCFL correlation with coefficients developed using the UPTF Test 11 data [ ] will calculate conservative CCFL for the Transient Two-Phase Flow (TOPFLOW) experiments to ensure applicability across a range of pressure conditions. (Figure 1-15). The provided figure indicates conservative trend of flooding characteristics for steam/water application. The original paper (Reference 5) can be referred to for discussion on the air/water response depicted on the same figure, which is not relevant to the subject. It can be seen from this figure that S-RELAP5 will conservatively calculates the steam flow rate at which there will not be any liquid down flow.

In summary, S-RELAP5 with the use of Wallis-form of CCFL correlation with [ ] will calculate conservatively the hot leg counter-current flow as well as the steam flow at which no liquid down flow through the hot leg will occur.

In addition to the application of CCFL [

] A detailed discussion of the S-RELAP5 CCFL model and the S-RELAP5 verification and validation (V&V) effort to benchmark it can be found in the supporting documents (Section 5.5 of Reference 1 and Section 3.7.6 of Reference 2) and the AREVA RLBLOCA and SBLOCA methodology framework can be reviewed in the topical reports (References 3 and 4). Historically, a comprehensive test facility benchmarking as part of the code V&V has been performed for S-RELAP5 and the predecessor codes (see Section 5.0 of Reference 1 and Sections 3 & 4 of Reference 2).

The CCFL phenomenon at fuel assembly exit in S-RELAP5 is represented by the Kutateladze CCFL correlation to limit downflow at the core upper tie plate, as discussed in Section 4.3.1.11.3 of Reference 3. A comparison of S-RELAP5 predicted Kutateladze parameters relative to the correlation derived from the UPTF Test 29 data demonstrates that the S-RELAP5 calculation is conservative. In addition to the benchmarking, several sample problems representing the applicable plant designs are executed to demonstrate the code's ability to predict the expected phenomena and behavior, including CCFL. Typical sample problems studied are the 3- and 4-loop Westinghouse and the CE plant designs.

References:

1. EMF-2100(P), Revision 16, "S-RELAP5 Model and Correlation Code Manual," December 2011.
2. EMF-2102(P), Revision 1, "S-RELAP5 Code Verification and Validation," November 2010.
3. EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.
4. EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.
5. C. Vallee, et al., "Counter-Current Flow Limitation Experiment in a Model of the Hot Leg of a Pressurised Water Reactor – Comparison between Low Pressure Air/Water Experiments and High Pressure Steam/Water Experiments," NURETH-13, Kanazawa City, Ishikawa Prefecture, Japan, September 27 – October 2, 2009.

Figure 1-15: SBLOCA Hot Leg to SG Inlet CCFL Compared to TOPFLOW CCFL Data

[ ]

Response to RAI 1.2:

a) As described in response to RAI 1.1a and 1.1c, the Wallis CCFL correlation developed from UPTF Test 11 [ ] The assessment of UPTF Test 11 has been performed using S-RELAP5 and is documented in Section 4.3.1.11.5 of Reference 1. [ ] The benchmark results contained within the response to RAI 1 supports the conclusion that the 5-volume hot leg model with the selection of hot leg CCFL correlation coefficients developed using UPTF Test 11 is adequate.

b) The methodology used in Supplement 1 to EMF-2328(P)(A) is such that the code [ ] Therefore, a sensitivity study on the effect of angle would not impact the evaluation model (EM) results.

c) AREVA has performed a benchmark to ROSA-IV Test SB-CL-18 in support of the response to RAI 1.1 using prescribed Wallis constants [

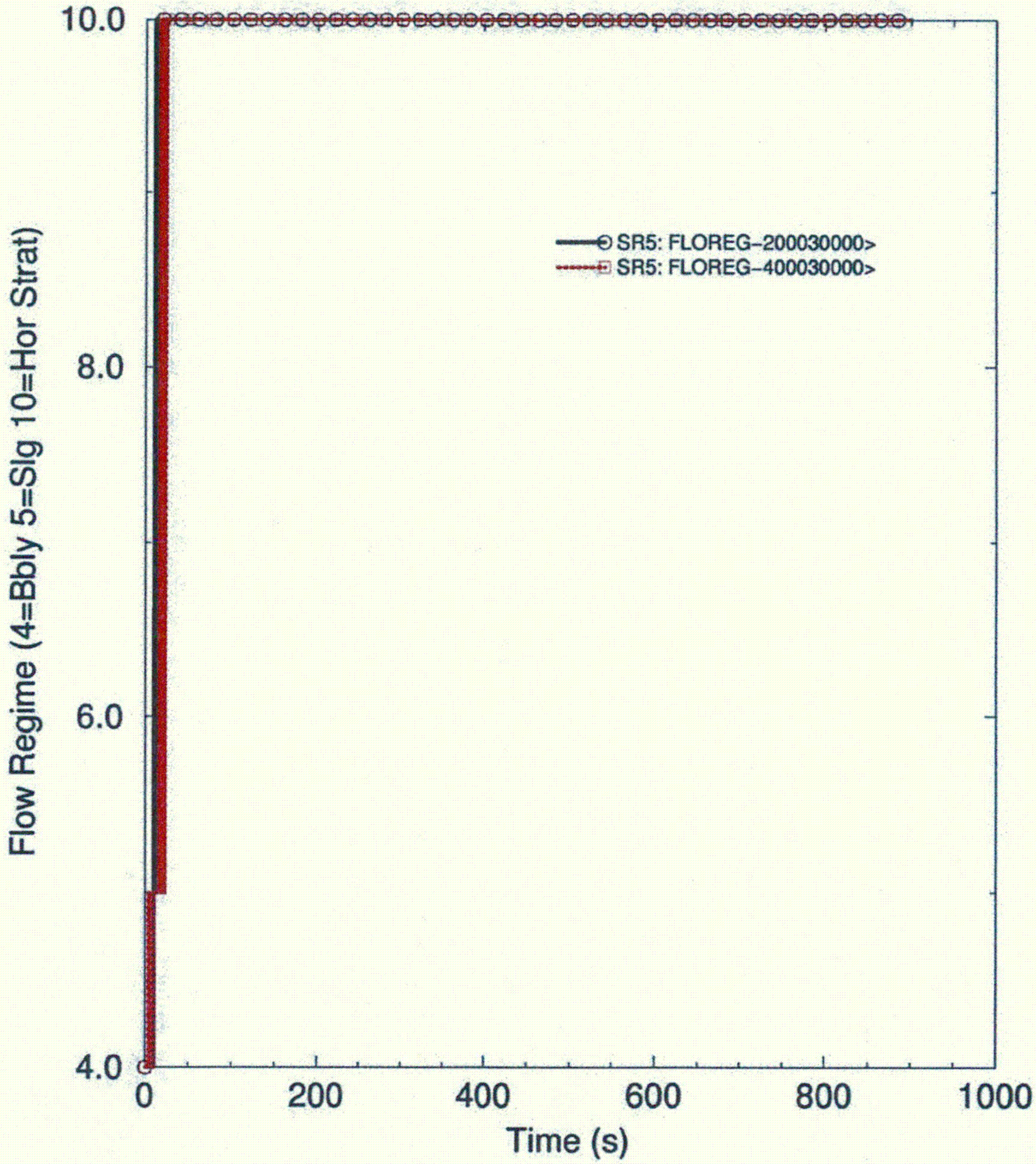
]

As expected, the hydraulic conditions in the hot leg transitions from bubbly and slug flow to horizontally stratified in less than 50 seconds. Flow stratification in the hot leg pipe section is an indicator that the code correctly captures stratified counter-current flow condition.

References:

1. EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

Figure 1-16: S-RELAP5 Predicted Flow Regime in Hot Leg (ROSA-IV Test SB-CL-18)



Response to RAI 1.3:

A benchmark of ROSA-IV Test SB-CL-18 and a benchmark of SEMISCALE S-UT-8 were performed with the S-RELAP5 code. The results of the ROSA-IV Test SB-CL-18 benchmark will be used to respond to the majority of requests for RAI 1.3 (1.3 b - 1.3 f). SEMISCALE is a small scale facility with an atypical hot leg and, as a result, is not as representative of PWR geometry.

Response to RAI 1.3a:

The S-UT-8 benchmark has been rerun in accordance with the EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0 methodology. The following major changes have been applied to the original model (EMF-2328(P)(A), Revision 0):

[ ]

The final noding diagram (including modeling changes) for the S-RELAP5 model used by this analysis is shown in Figure 1-17.

The transient was run for 750 seconds, slightly longer than the test duration of 735 seconds. The initial depressurization of the system is slightly under-predicted by the code in the first 250 seconds (Figure 1-18); however, the mass flow rate from the break matches well during the same time frame (Figure 1-20). The higher primary side pressure is likely due to the higher secondary side pressure. The collapsed liquid level in the intact SG upflow tubes matches the measured data well, emptying at about 200 seconds, as opposed to the 220 second empty time of the experiment (Figure 1-21). The downflow side is similar, with the calculated level emptying about 20 seconds earlier than the test (Figure 1-22), indicating a similar timing of loop seal clearing. This difference in the SG upflow level is the cause of the difference seen in the core liquid level at the time of uncovery. While the code calculated data and measured data show similar times of uncovering (Figure 1-23), the code calculated level is slightly higher than the measured data because of the benefit of earlier liquid draining from the SG tubes into the core. With more liquid in the SG U-tubes, the experimental core collapsed level is lower because of the manometric effect. However, this difference in SG tube liquid level is small, and, given the scale of the test, this indicates that the chosen CCFL parameters are reasonable. This is confirmed again by the PCT prediction shown in Figure 1-24, in which the code calculated initial heatup bounds the data. Thus, during the pre-loop seal clearing period with the system hydrodynamics primarily controlled by the break flow and the manometric balance on the up-side and the down-side of the loop S-RELAP5 properly calculated the system behavior including the liquid holdup in the SG tubes and the deep core uncovery during this period.

Following core quench, the S-UT-8 test undergoes another heatup during the core boil-off period, which is not predicted by S-RELAP5. The lower primary side pressure during this time period (Figure 1-18) and lower break mass flow rate in the S-RELAP5 calculation shows that the code injects more emergency core cooling (ECC) into the system, and bypasses less ECCS out the break than the test. During this boil-off period, the system response is a result of the net balance between various aspects, including the core boiling, condensation on ECC water in the cold legs, ambient heat loss, ECC bypass and the break flow.

Because these aspects of such a small scale facility are difficult for codes to predict, it is not surprising that the results do not match during this period. In particular, the downcomer is simulated in the test using a small diameter pipe as opposed to the large annulus of a PWR. The small diameter pipe can affect a number of aspects of the prediction, particularly the ECC core bypass/displacement of liquid into the cold legs. It may be possible to improve the core boil-off prediction by using different correlations and modeling approaches. For example, different condensation models, a small-pipe interphase drag correlation for countercurrent flow in vertical sections, or a different treatment of the downcomer-cold leg connection could be utilized. However, these changes would be outside of their range of validity if applied to plant applications. As the fundamental purpose for benchmarks is to demonstrate the S-RELAP5 capability to properly predict the system response as well as the cladding thermal response in an actual PWR following an SBLOCA, no attempt was made to modify the code in order to improve S-UT-8 test prediction. For benchmarking the system behavior and core thermal response during the core boil-off period, large scale facilities, such as ROSA and LOFT, are more appropriate benchmarks. The benchmarks of ROSA Tests SB-CL-18 (Response to RAI 1.1b), and IB-CL-03 (Response to RAI 1.4) and LOFT Tests L3-6 and L8-1 (Response to RAI 3) demonstrate that S-RELAP5 calculates conservative to best estimate system behavior as well as core thermal response during the entire transient period, including the core boil-off period.

As shown, during the pre-loop seal clearing period, where the comparison to a SEMISCALE test is appropriate, S-RELAP5 properly calculates the system behavior including the liquid holdup in the SG tubes and core uncovery. For the post-loop seal clearing period, where the SEMISCALE facility does not scale well, the LOFT and ROSA benchmarks demonstrate that S-RELAP5 properly simulates the transient.

Figure 1-17: S-UT-8 Noding Diagram

Figure 1-18: Primary System Pressure (Upper Plenum)

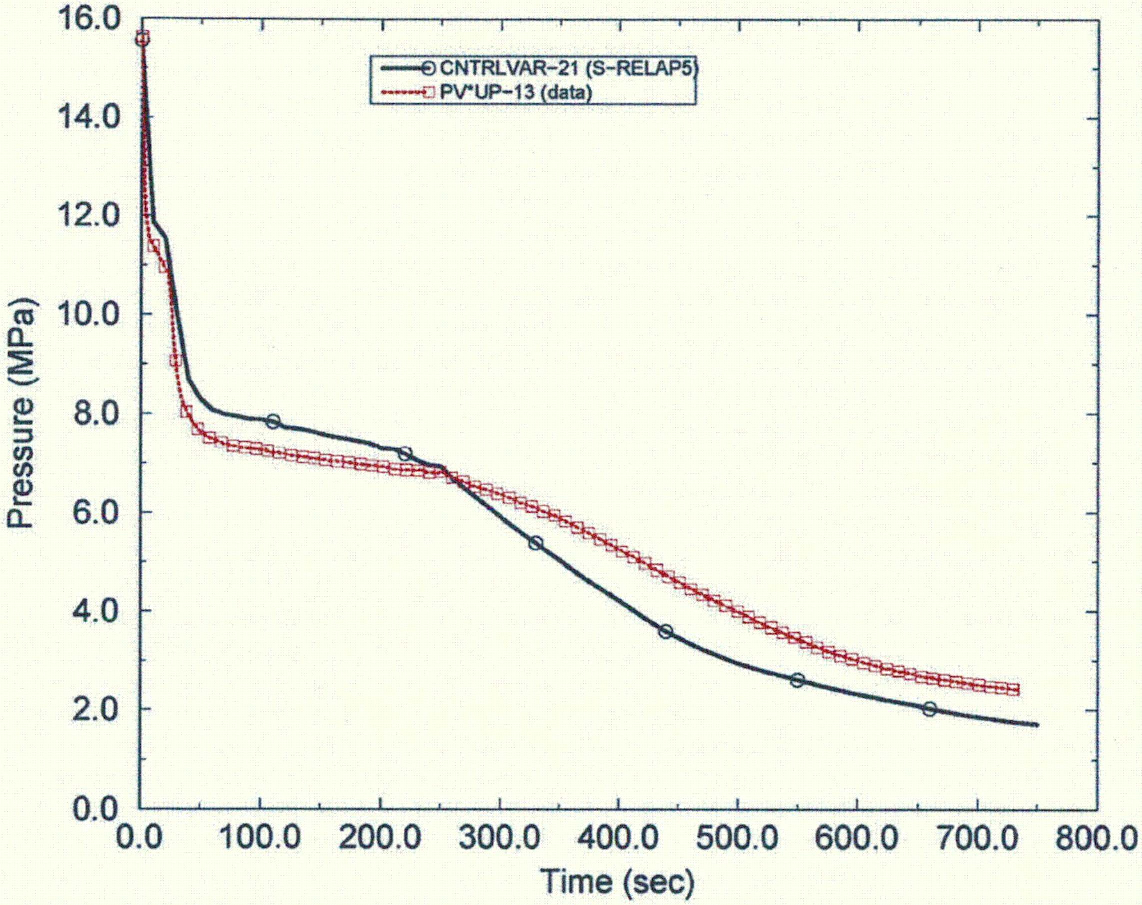


Figure 1-19: Secondary Side Pressures

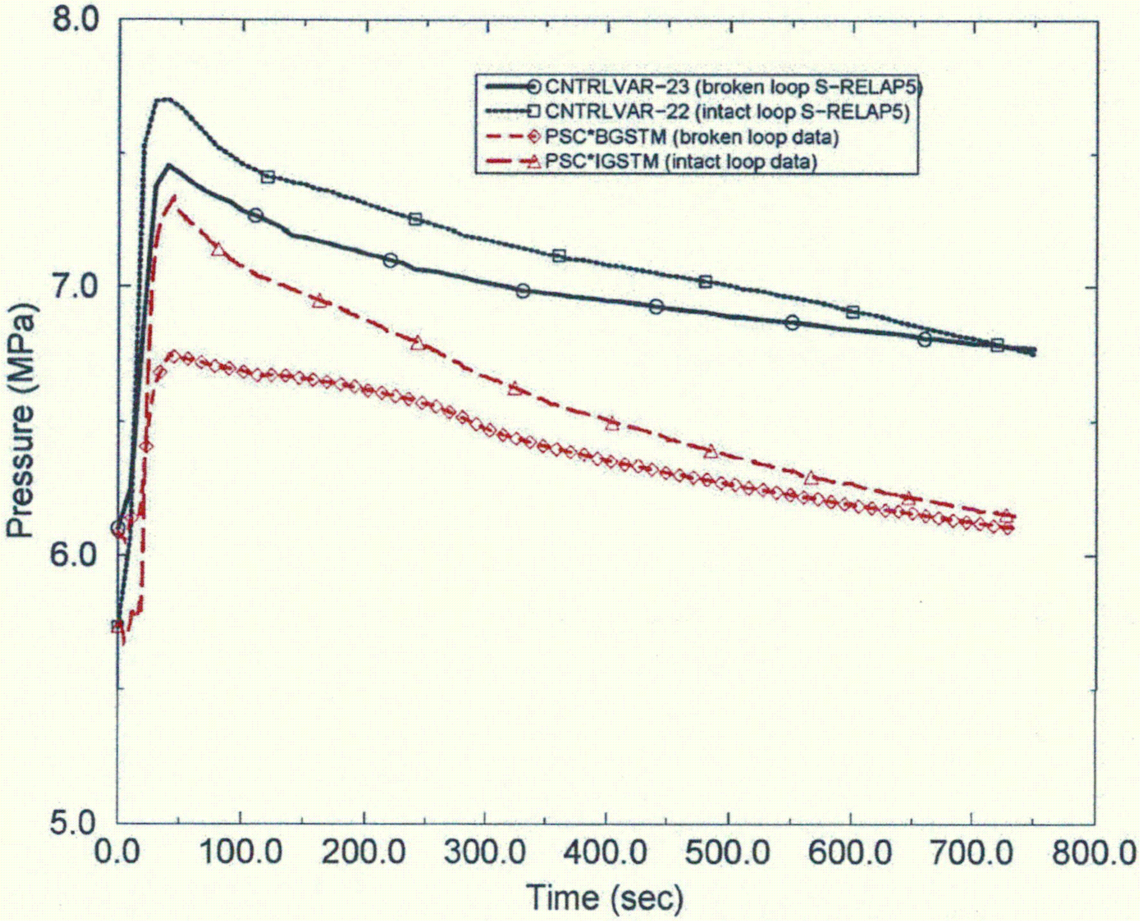


Figure 1-20: Integrated Mass Lost out the Break

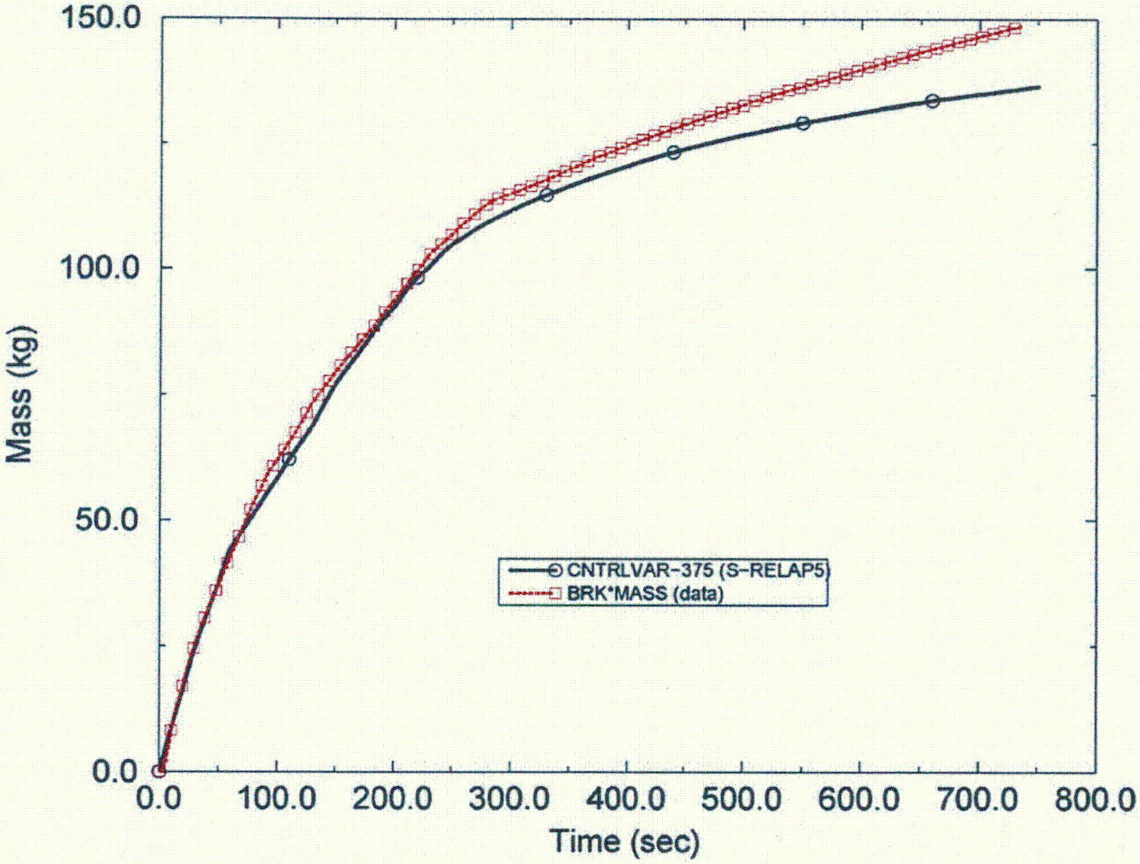


Figure 1-21: Collapsed Level in the Intact SG Upflow Tubes

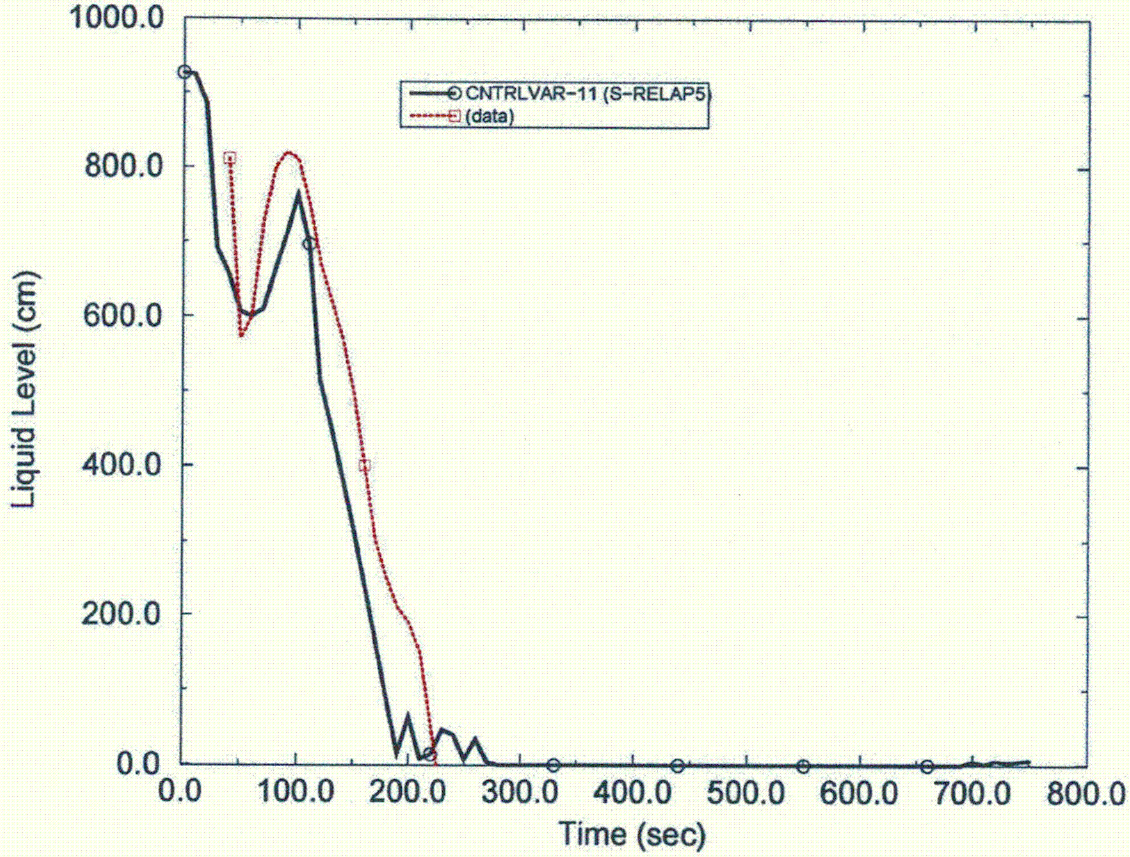


Figure 1-22: Collapsed Level in the Intact SG Downflow Tubes

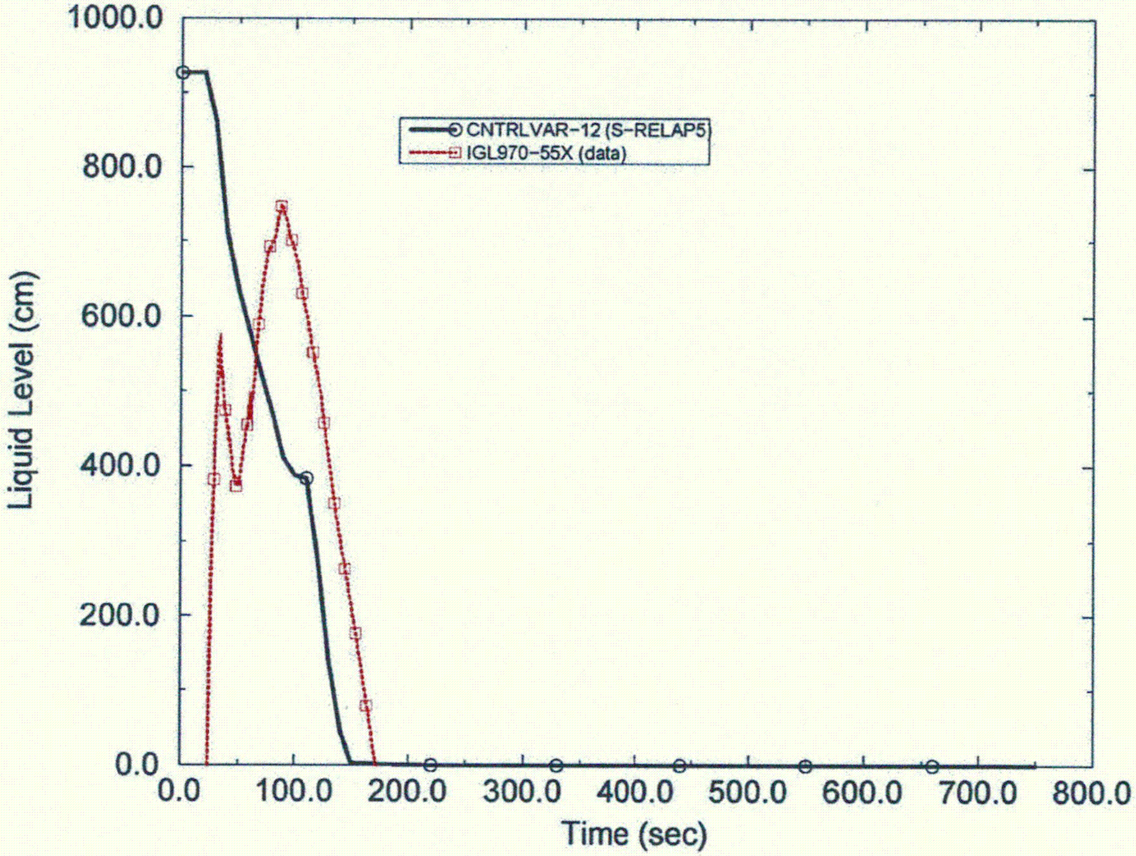


Figure 1-23: Collapsed Level in the Core

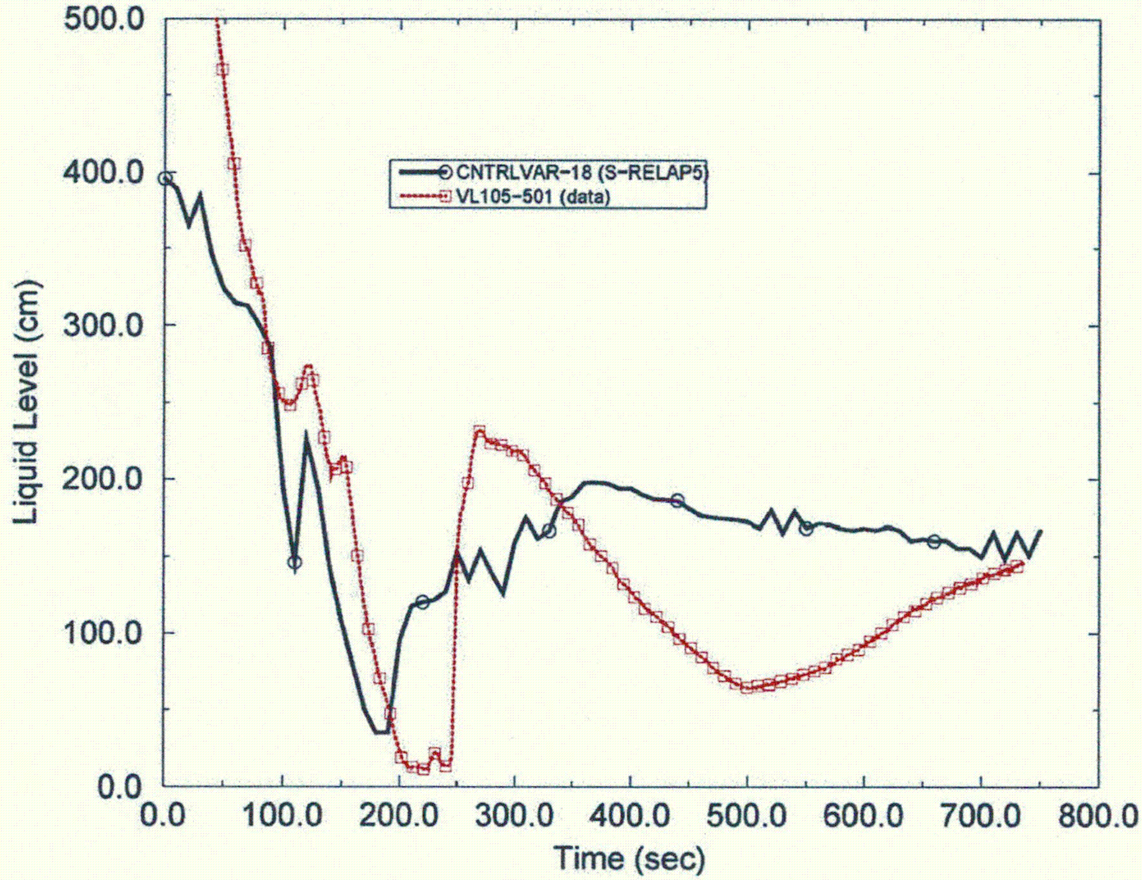
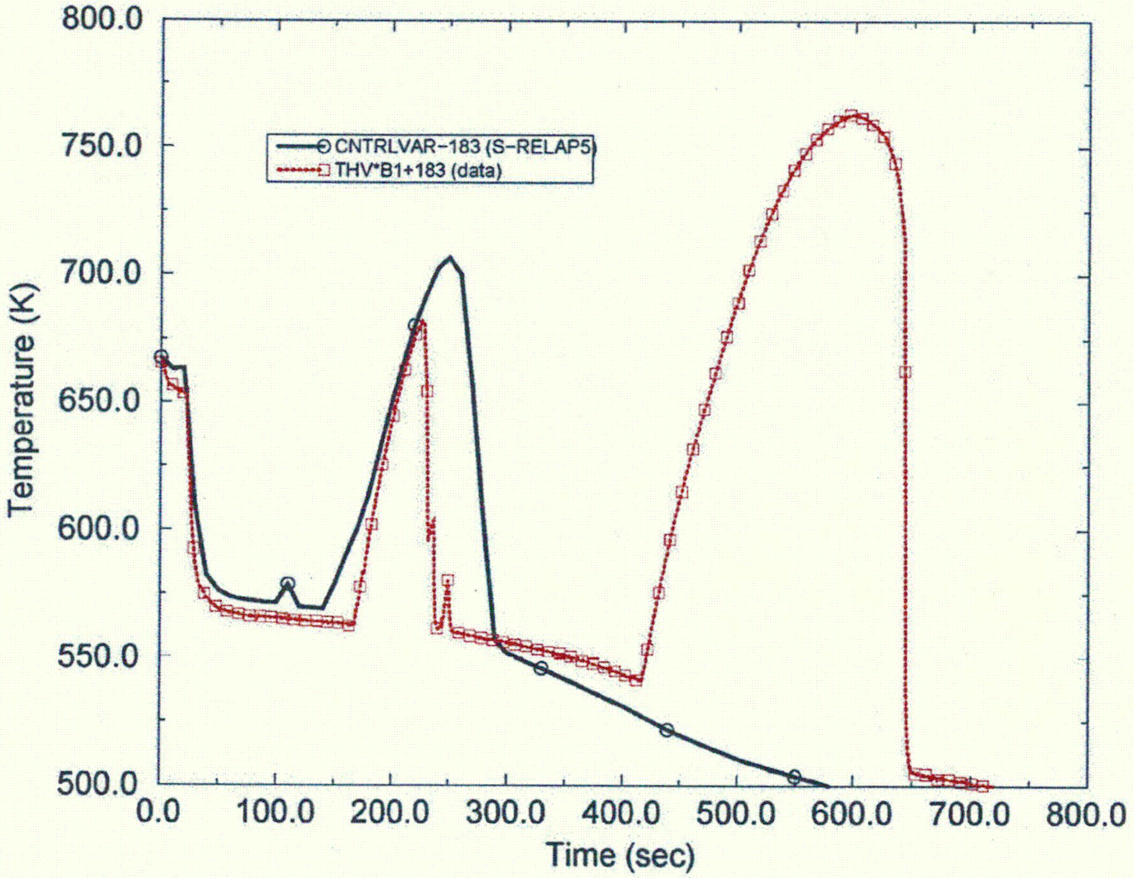


Figure 1-24: Core Mid-Plane Cladding Temperature



Response to RAI 1.3b:

Figure 1-4 and Figure 1-5 of the response to RAI 1.1 provide test data to prediction comparisons of the liquid level in the up-side of the steam generator tubes for ROSA-IV loop A (intact loop) and loop B (broken loop), respectively.

Response to RAI 1.3c:

Figure 1-6 and Figure 1-7 of the response to RAI 1.1 provide test data to prediction comparisons of the liquid level in the up-side of the pump suction piping for the broken and intact loops, respectively.

Response to RAI 1.3d:

Figure 1-11 of the response to RAI 1.1 provides a test data to prediction comparison of the cladding temperature at the peak clad temperature elevation. Core level and system pressure responses are shown in Figure 1-13 and Figure 1-3, respectively.

Response to RAI 1.3e:

S-RELAP5 hot leg flow regimes, for the benchmark against ROSA-IV Test SB-CL-18, are provided in Figure 1-16 as part of the response to RAI 1.2.

Response to RAI 1.3f:

Loop seal behavior in the ROSA-IV benchmark is discussed in the response to RAI 1.1 (Figure 1-6 and Figure 1-7). Both loop seals clear in the test, but the S-RELAP5 run preferentially clears the intact loop seal, somewhat later than the test, resulting in greater core level depression (Figure 1-13) and an over-prediction of cladding heatup (Figure 1-8 through Figure 1-12).

Response to RAI 1.4:

AREVA performed a benchmark calculation of the ROSA-V experiment IB-CL-03 with S-RELAP5 to demonstrate the S-RELAP5 code behavior for larger “small breaks.” This test simulated a 17 percent cold leg intermediate break loss of coolant accident (IBLOCA) to represent a double-ended guillotine break of an emergency core cooling system nozzle, with an assumed single-failure of diesel generators related to the high pressure injection (HPI), low pressure injection (LPI) systems and total failure of auxiliary feedwater.

Test IB-CL-03 simulated an IBLOCA with a double-ended guillotine break (DEGB) of an emergency core cooling system (ECCS) pipe connected to a cold leg. The break was sized to simulate a safety injection tank (SIT) line break (17 percent of the cold leg break area) with a single-failure of both HPI and LPI systems and total failure of auxiliary feedwater. This benchmark extends beyond the range of where the SBLOCA methodology in EMF-2328(P)(A) is approved (17 percent cold leg break area vs. 10 percent cold leg break area). As such, it is expected that some aspects of the methodology may not perform well for this intermediate-sized break. [

] Because of the intermediate size break, this modification was implemented in this study.

The experimental procedure for this test was to open the break at time 0.0, and then begin the power decay at approximately 10.0 seconds. Figure 1-25 compares the calculated primary and secondary pressure with that measured during the test. As shown in Figure 1-25, the relatively large break caused a rapid depressurization beginning at time 0.0, with an increase in the secondary side pressure after closure of the main steam isolation valves to about 8 MPa. After approximately 55 seconds, the primary dropped below the secondary pressure so that the secondary side no longer served as a heat sink. Until that time, S-RELAP5 did a reasonable simulation, but the calculated primary pressure decreased more rapidly than measured.

Figure 1-26 compares the calculated and measured break flow, and shows that S-RELAP5 simulates both the early liquid and the subsequent vapor flow reasonably well. The technique used in the ROSA-IV benchmark (RAI 1.1) of representing the break valve as being directly connected to the cold leg was not effective for this simulation. The break flow simulation improved when the S-RELAP5 model used [

]

Figure 1-27 compares the peak cladding temperature (PCT) calculated by S-RELAP5 with that measured during test IB-CL-03. As shown in this figure, the calculation begins to heatup slightly early, and reaches its peak about 25 seconds before the test. The location of the PCT is also slightly lower, more closely matching the axial power profile. The high pressure injection (HPI) system was started at about 35 seconds, almost simultaneously with the core dryout, but was ineffective in both the experiment and calculation because the injection flow rate was much smaller than break flow rate. Accumulator injection during the test started at approximately 110 seconds, but was also ineffective in terminating the temperature excursion. As the S-RELAP5 predicted primary pressure during this time was slightly low, the code predicted the start of accumulator injection by 90 seconds, contributing to the slightly early turnaround of the calculated cladding temperature excursion. The test was terminated at approximately 150 seconds to protect the experimental facility when core temperatures approached 1000 K. At this time, the protection system for the facility decreased power and initiated full ECCS flow.

One of the purposes of the ROSA-V IB-CL-03 benchmark was to evaluate the effects of CCFL in the steam generator tubes as well as above the upper core plate. The IB-CL-03 data report states that coolant remained in the upflow-side of the steam generators until about 100 seconds, contributing to the core uncovery. Figure 1-28 compares the measured and calculated pressure differential across the steam generator upside as well as the downside, for the loop with the pressurizer. As seen in this figure, both test data and the code calculation showed that there is coolant retained in the upside, with S-RELAP5 predicting more liquid hold up in the steam generator from approximately 25 to 50 seconds. The code then shows that the fluid is released back to the hot leg. The test indicated less fluid, but in the calculation, CCFL holds it in the steam generator up-side until approximately 100 seconds.

Figure 1-29 shows a similar trend for the differential pressure across the steam generator in the loop without the pressurizer. Once again, S-RELAP5 predicts that CCFL will retain more coolant than measured in the steam generator tubes, with the liquid draining back into the hot leg after approximately 50 seconds.

The calculated and measured pressure differential across the upper plenum is compared in Figure 1-30. As shown in this figure, the test showed a pressure differential prior to 40 seconds, indicating a mass of water is being held above the upper core plate by CCFL. The measured data indicates this water is swept to the hot legs. The S-RELAP5 model also predicts that the liquid is held above the core until about 40 seconds, but then predicts that the liquid in the upper plenum drops into the core region.

The test demonstrated coolant being accumulated above the core, in the steam generator U-tube upflow-side, and within the steam generator inlet plenum due to CCFL by high velocity vapor flow. The liquid hold up in both the steam generator tubes and the vessel upper plenum was predicted early in the transient by S-RELAP5, although the code predicted the liquid would drop into the core region rather than being swept to the hot legs.

Figure 1-25: Comparison of Primary and Secondary Pressures

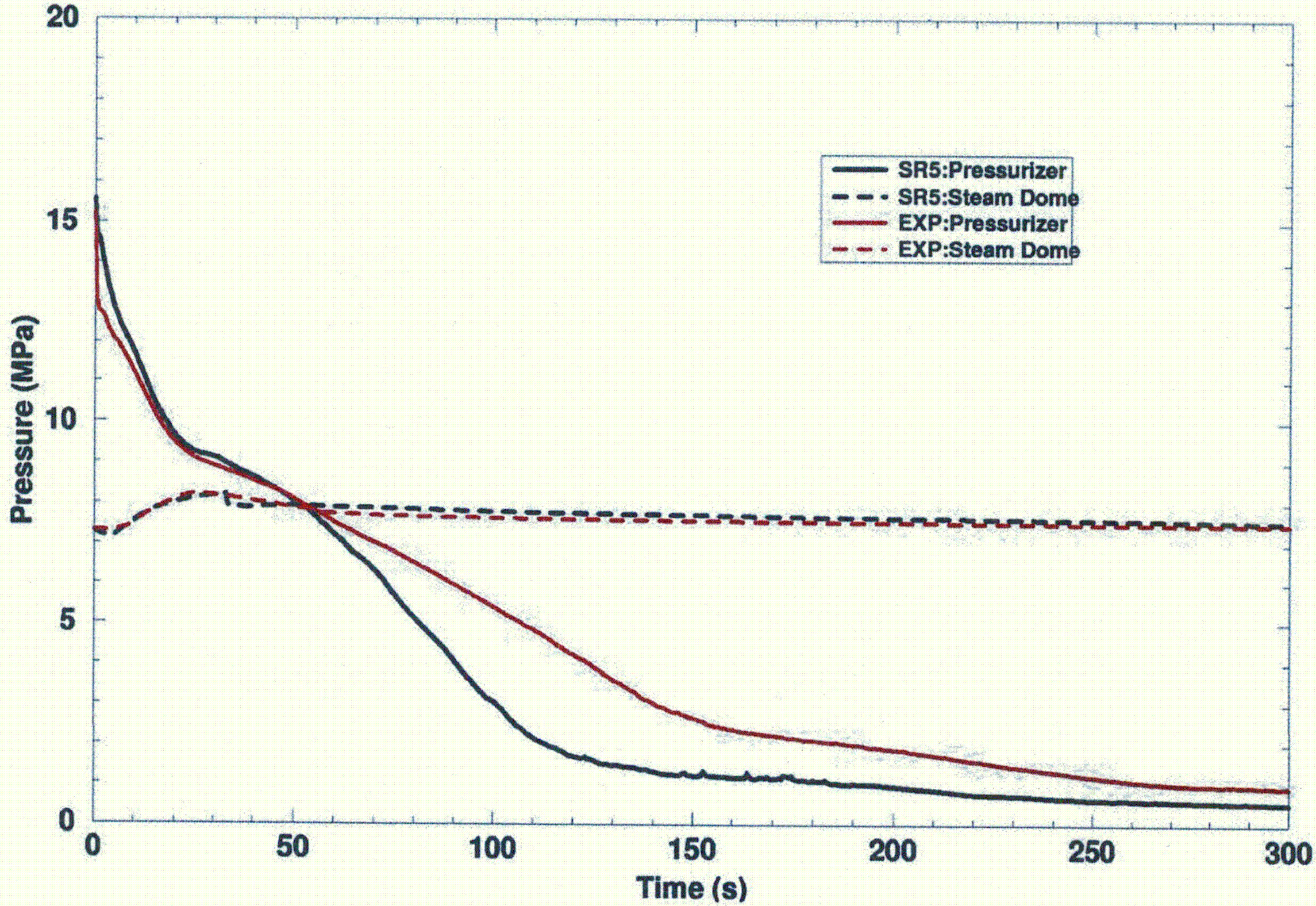


Figure 1-26: Comparison of Calculated and Measured Break Flow

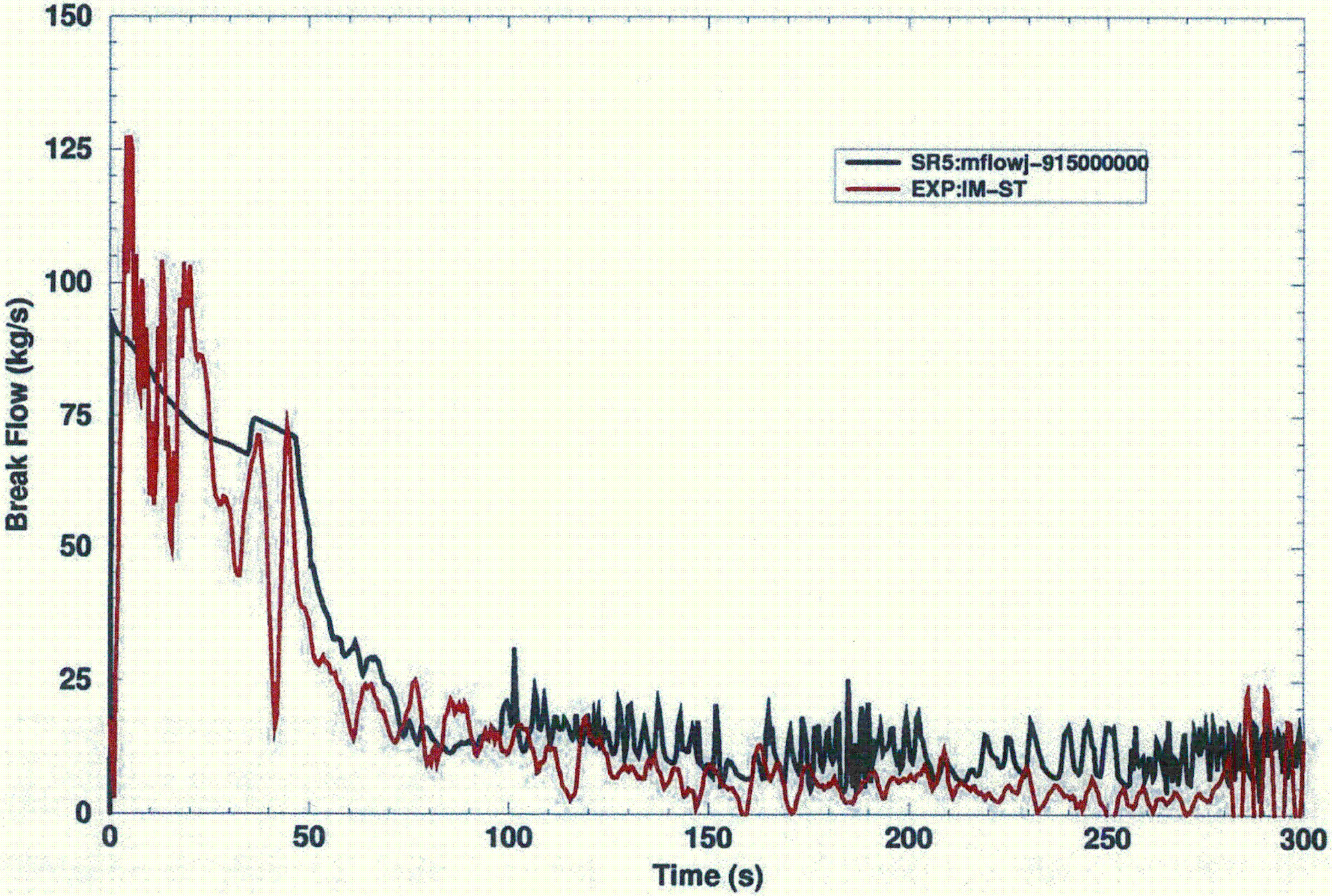


Figure 1-27: Comparison of Measured and Calculated Peak Cladding Temperature

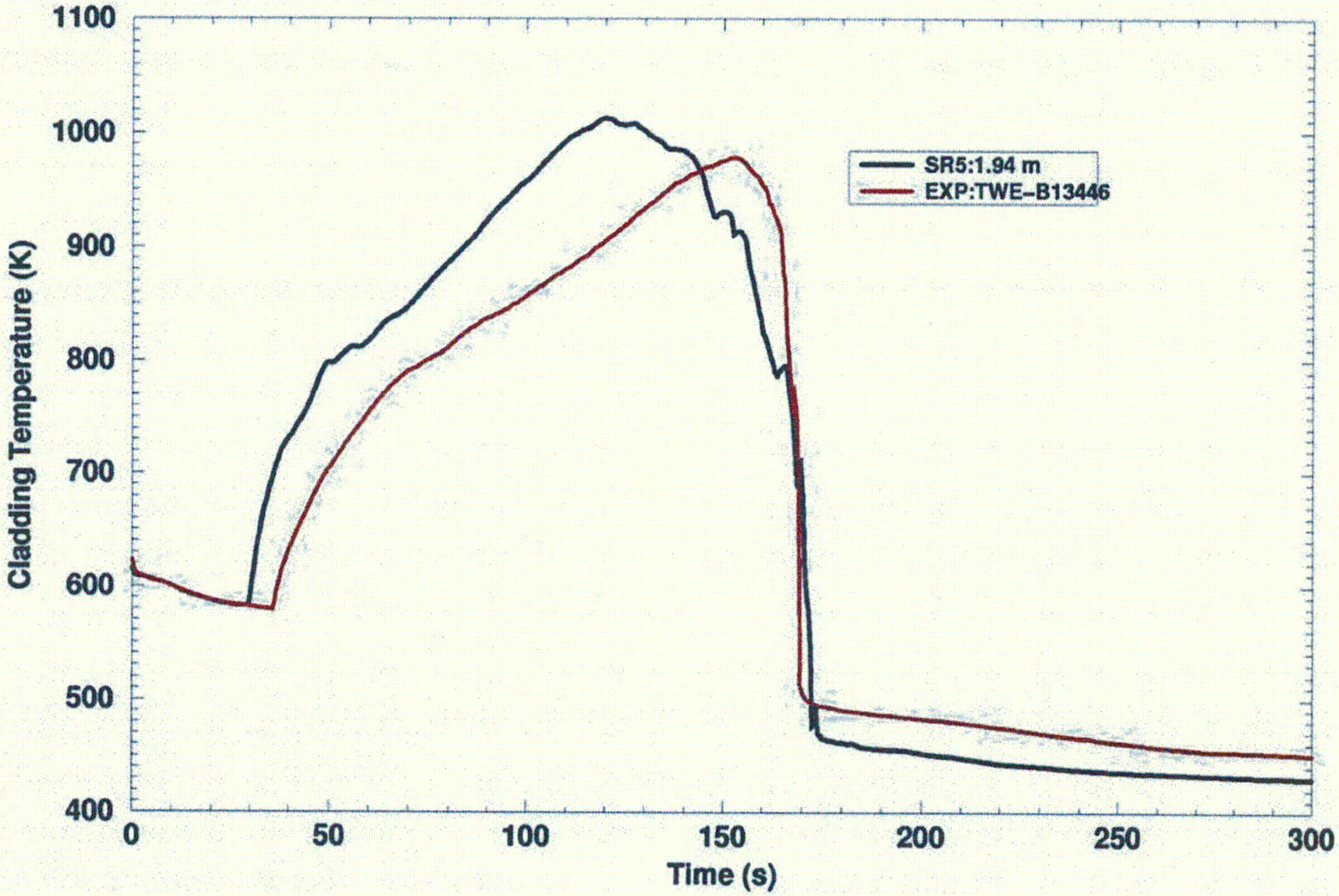


Figure 1-28: Steam Generator Differential Pressure in Pressurizer Loop

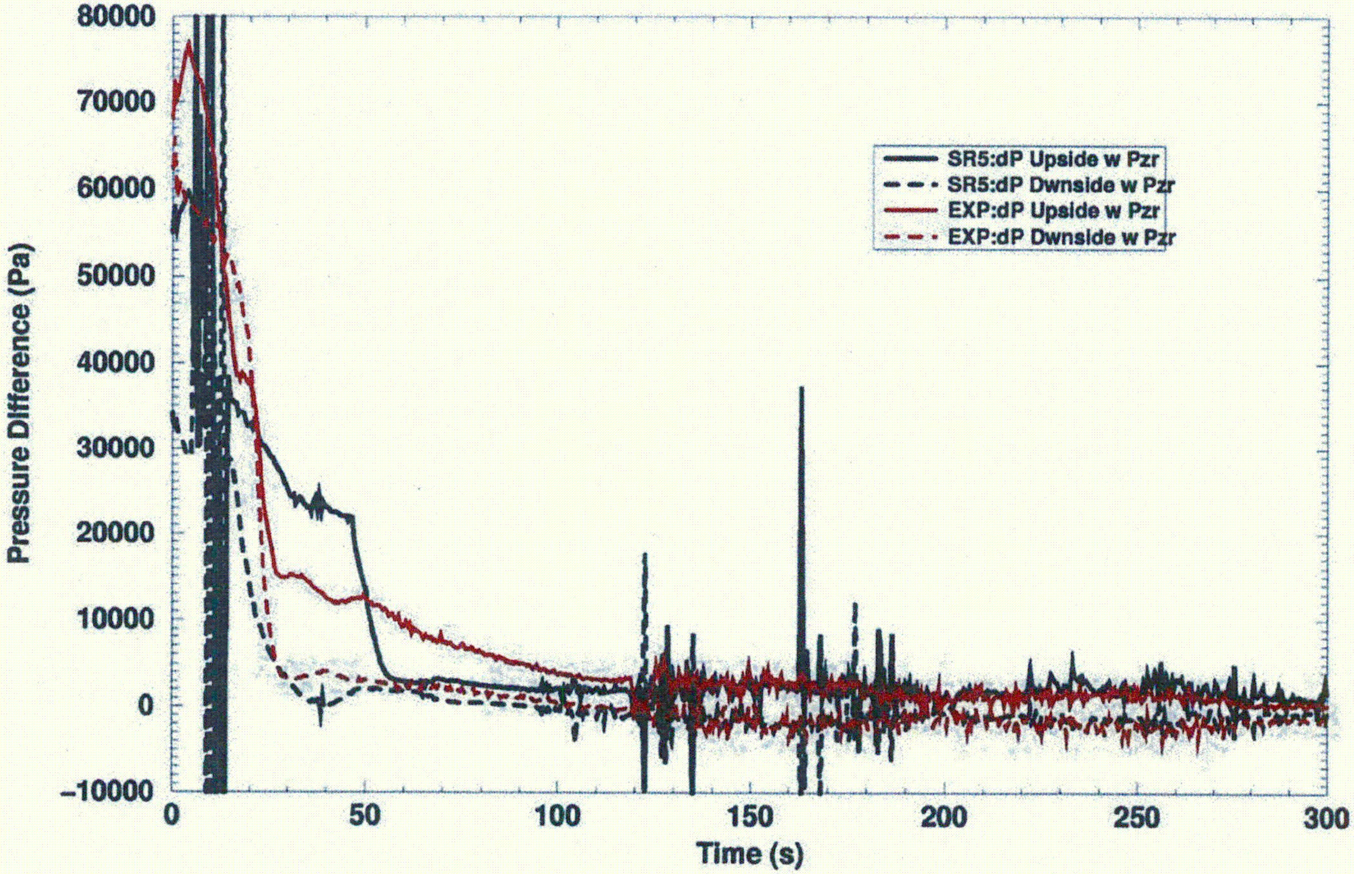


Figure 1-29: Steam Generator Differential Pressure in non-Pressurizer Loop

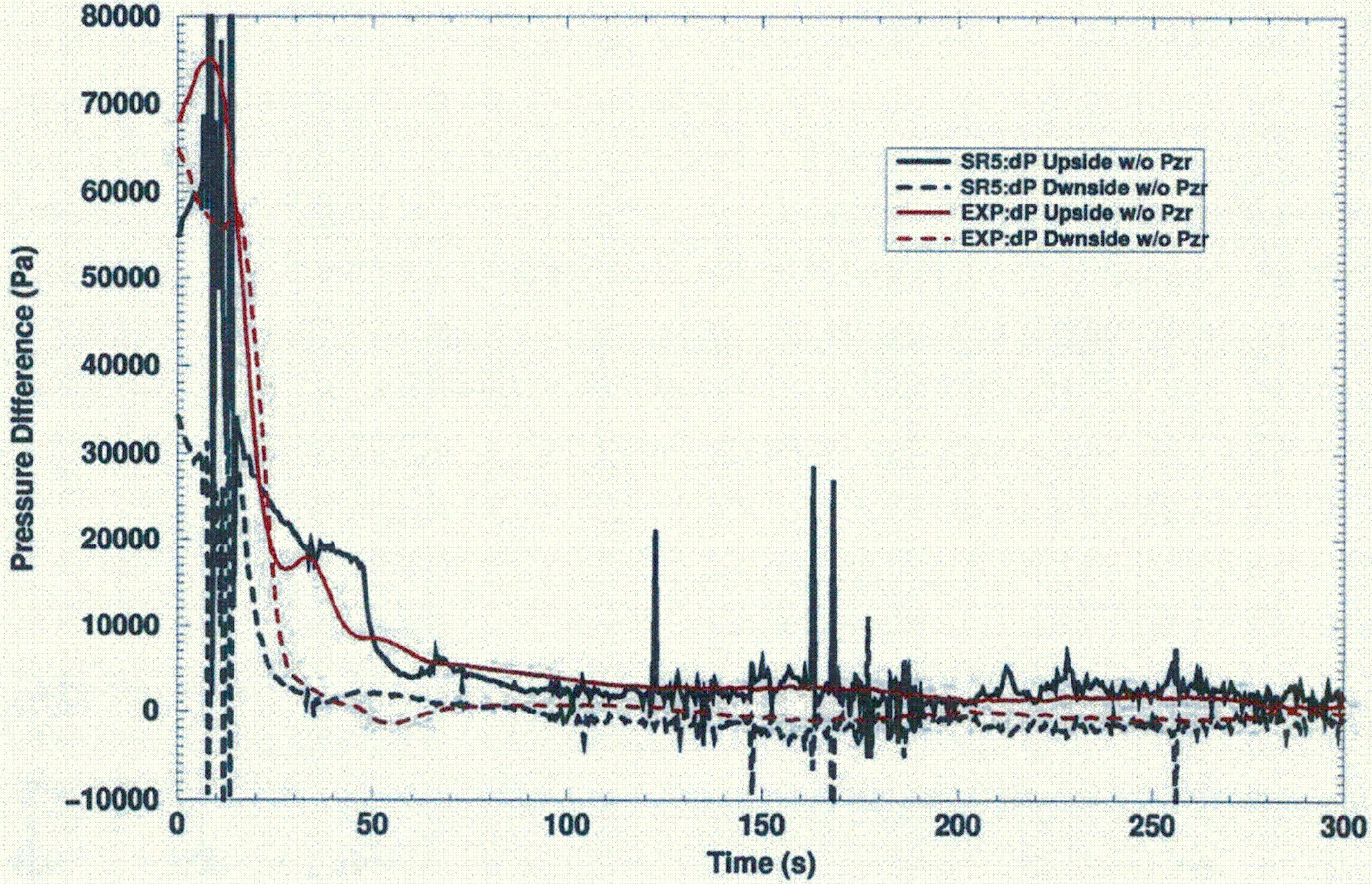
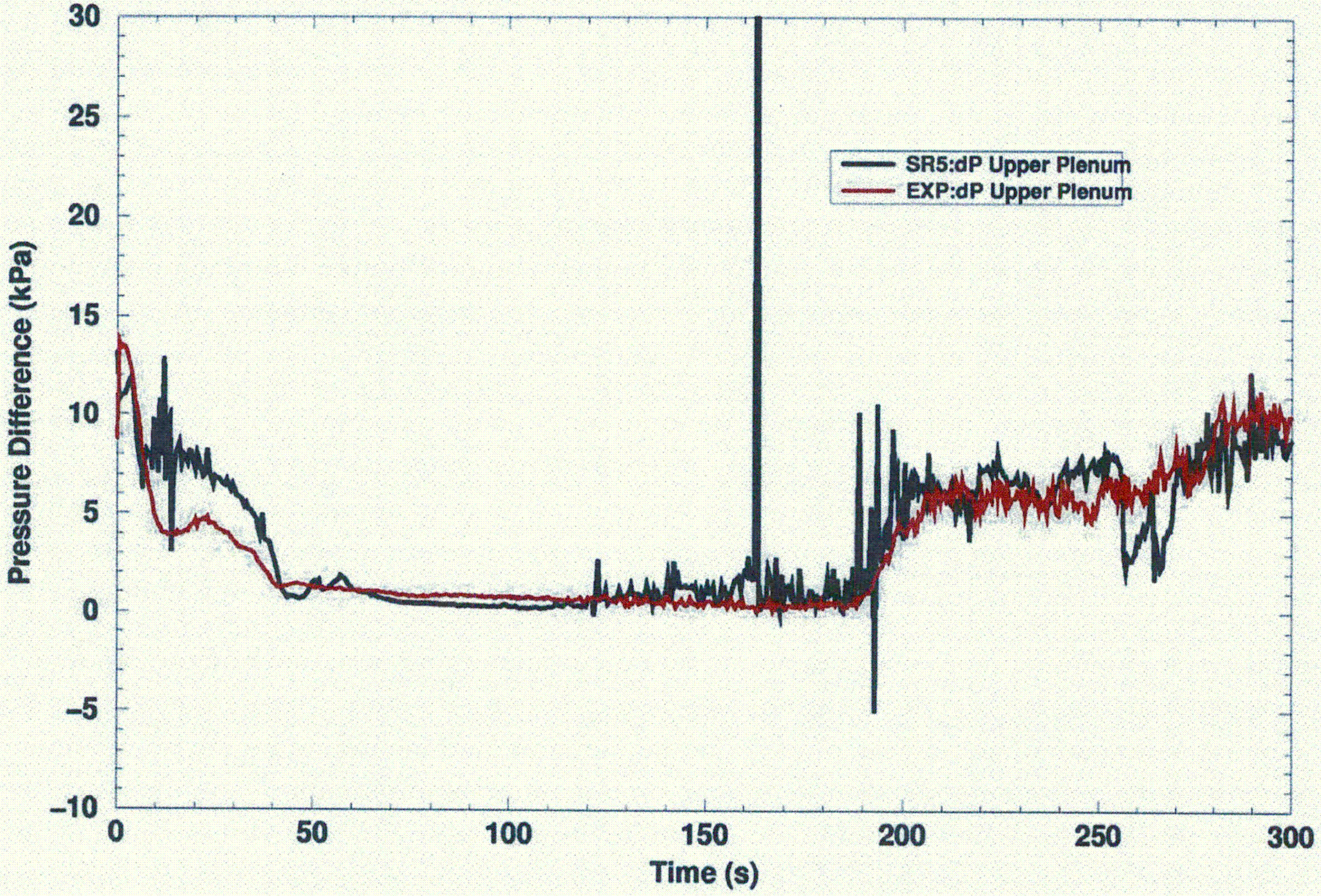


Figure 1-30: Pressure Differential across the upper plenum



Response to RAI 1.5:

Plant calculations for the largest SBLOCA in the cold leg with the methodology in Supplement 1 to EMF-2328(P)(A) are included as part of a set of sample problems in the response to RAI 5. The break spectrum included for the sample problems in the response to RAI 5 covers the break sizes for which the methodology is applicable. In addition, the response to RAI 1.3 contains a benchmark comparison against a 17 percent break in the ROSA-V test. While this is outside of the range to which the methodology will be applied, there is still a good agreement overall.

Response to RAI 1.6:

The S-RELAP5 SBLOCA model uses a form of the Wallis CCFL correlation [ ] The correlation constants for the tube inlet are based on standard applications for small diameter tubes [ ] Constants for the entrance to the inlet plenum were developed by benchmarking to UPTF Test 11 [ ] The response to RAI 1.1 contains details concerning the correlation and the UPTF benchmark.

a) A benchmark of ROSA-IV Test SB-CL-18 was performed with the S-RELAP5 code in response to RAI 1.1.

b) Comparisons of steam generator up-side tube levels resulting from the benchmark used to support RAI 1.1 (see Figure 1-4 and Figure 1-5) demonstrate that the CCFL correlations used in the model hold water up in the steam generators in a conservative manner, delaying draining back into the vessel.

RAI 2:

Section 3 of the TR describes the core bypass modeling. Since the S-RELAP5 code will be applied to plants with upper head spray nozzles.

2.1 Please demonstrate through a comparison to test data that S-RELAP5 can properly capture the effect of core bypass on system response following a SBLOCA.

2.2 Show that loop seal hydraulic behavior is properly captured as well as the core two-phase and liquid level responses in the core during uncovery.

2.3 Present data comparison to the clad temperature at the PCT location. Please see NUREG/CR-4438 for a description of these tests and data.

Response to RAI 2:

The SEMISCALE tests are specific to a 5 percent cold leg break. Core bypass ranges from 0.9 percent for S-LH-1 to 3.0 percent for S-LH-2. Figure 2-1 and Figure 2-2 are comparison plots of the SEMISCALE reactor vessel liquid level and fuel element surface temperature from NUREG/CR-4438. In the case of higher bypass, the vessel level depression during the manometric or loop seal clearing phase is reduced and the subsequent drain-down of the vessel during core boil-off is delayed. The higher bypass test results in no core uncovery during loop seal depression and a reduced peak cladding temperature. Peak cladding temperatures (PCT) during boil-off differ by about 80 K for the tests.

The ROSA-IV test, SB-CL-18, will be used to demonstrate the capability of S-RELAP5 to capture the influence of core bypass on the system response following a SBLOCA rather than the SEMISCALE tests. The ROSA-IV test, SB-CL-18, is similar to the SEMISCALE tests. AREVA's base model for the ROSA-IV SB-CL-18 test simulates a 5 percent cold leg break with a core bypass of 1.2 percent. Two sensitivity studies were performed to examine the effects of core bypass; bypass flows of 0.4 and 2.2 percent were chosen for the studies.

Figure 2-3 to Figure 2-9 present the plots that contain the results of the base case transient, each of the two bypass sensitivity studies, and test data for the ROSA-IV SB-CL-18 test. The system pressure transient characteristics for the 5 percent cold leg break are similar for the three cases. Draining of the steam generators occurs at roughly the same rate for all of the cases. However, the steam generator in the intact loop drains somewhat sooner for the low-bypass case.

A preference develops during core uncovery for steam release from the core exit to the break via the broken loop hot leg for the low-bypass case; the intact steam generator fluid and accumulator liquid collect in the intact-side loop seal, but the broken-side loop seal is nearly cleared. This occurs because of the exaggerated manometric characteristics associated with the lack of a bypass path from the upper head to the downcomer that restricts steam flow via the broken loop cold leg.

In comparison to the low-bypass case, the base and high-bypass cases allow steam to escape from the upper head to the break via the broken cold leg. Core exit pressure is slightly lower; steam passing to the loops passes preferentially through the intact loop hot leg, pushing liquid to the vessel and, ultimately, to the break via the broken cold leg.

Complete loop seal clearing, which would have the increased mixture level in the core and limited core element heatup, did not occur for any of the cases evaluated. The core heatup did show a sensitivity to core bypass. The low-bypass case exhibited the highest cladding temperature, about 110 K above the base case; whereas, the high-bypass case resulted in lower temperature, about 40 K below the base case.

Response to RAI 2.1:

The sensitivity of an SBLOCA event progression has been demonstrated by studies performed with a ROSA-IV S-RELAP5 model. Loop seal clearing in the broken leg is nearly achieved when bypass is limited. With higher core bypass, the loop seals do not clear. All cases predict conservatively high fuel temperatures. During the core boil-off, the core level is progressively more depressed, and core heatup increases as bypass is reduced. The expected effect of core bypass on system response to a 5 percent cold leg break has been demonstrated.

Response to RAI 2.2:

The hydraulic behavior of the loop seals and core liquid level response during core uncovery was compared for three different core bypass cases. The low-bypass case (0.4 percent bypass relative to loop flow) exhibited a unique, but explainable, loop seal level progression following steam generator draining. The base case and high bypass case (1.2 percent and 2.2 percent bypass) exhibited similar loop seal behavior. The core liquid levels during core boil-off were progressively more depressed, and the core heatup was increased as the bypass was reduced.

Response to RAI 2.3:

The sensitivity studies performed for response to the RAI utilize a ROSA-IV model that is similar to the SEMISCALE tests. The ROSA-IV test simulates a 5 percent cold leg SBLOCA with core bypass value of 1.2 percent. Sensitivity studies for 0.4 percent and 2.2 percent were conducted for this response.

The difference in peak clad temperature response between the base case and the low-bypass case is an increase of about 110 K for a reduction in bypass value of 0.8 percent. An increase in PCT was anticipated, but is exaggerated by the loop seal behavior of the low-bypass case. Comparing the base case with the high-bypass case, the PCT was reduced by 40 K for an increase in bypass of about 1.0 percent. The result of this latter comparison is in good agreement with the SEMISCALE S-LH-1 and S-LH-2 test comparison, which demonstrates a reduction in PCT of 80 K for an increase in bypass value of 2.1 percent.

Figure 2-1: Vessel Liquid Level for 5 Percent SBLOCA Experiments S-LH-1 (0.9 Percent Bypass Flow) and S-LH-2 (3.0 Percent Bypass Flow)

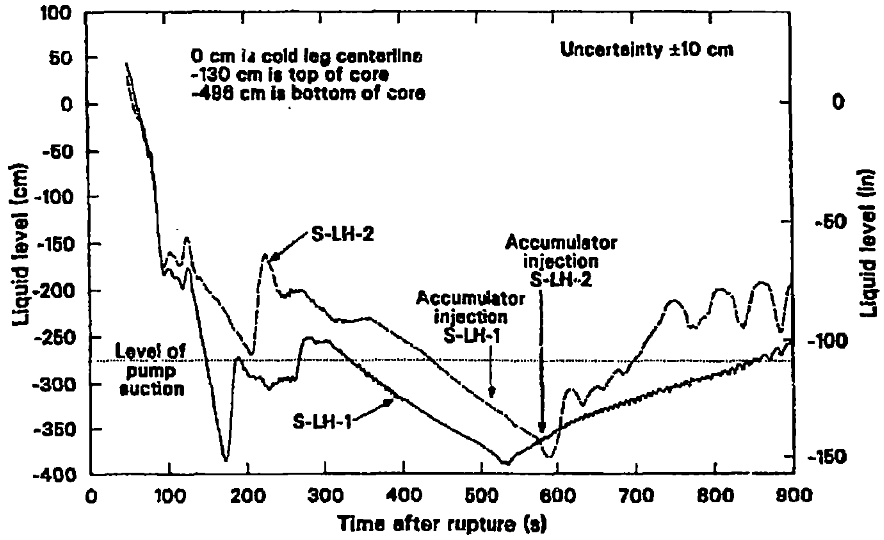


Figure 2-2: Core Heater Rod Temperature during 5 Percent SBLOCA Experiments S-LH-1 (0.9 Percent Bypass Flow) and S-LH-2 (3.0 Percent Bypass Flow)

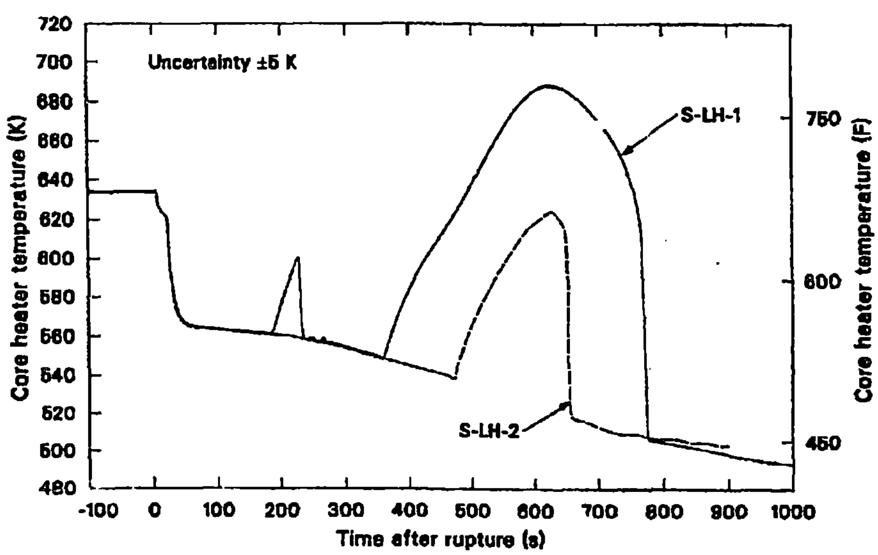


Figure 2-3: Pressurizer Pressure Transient Response Comparison, ROSA-IV Core Bypass Sensitivity Study

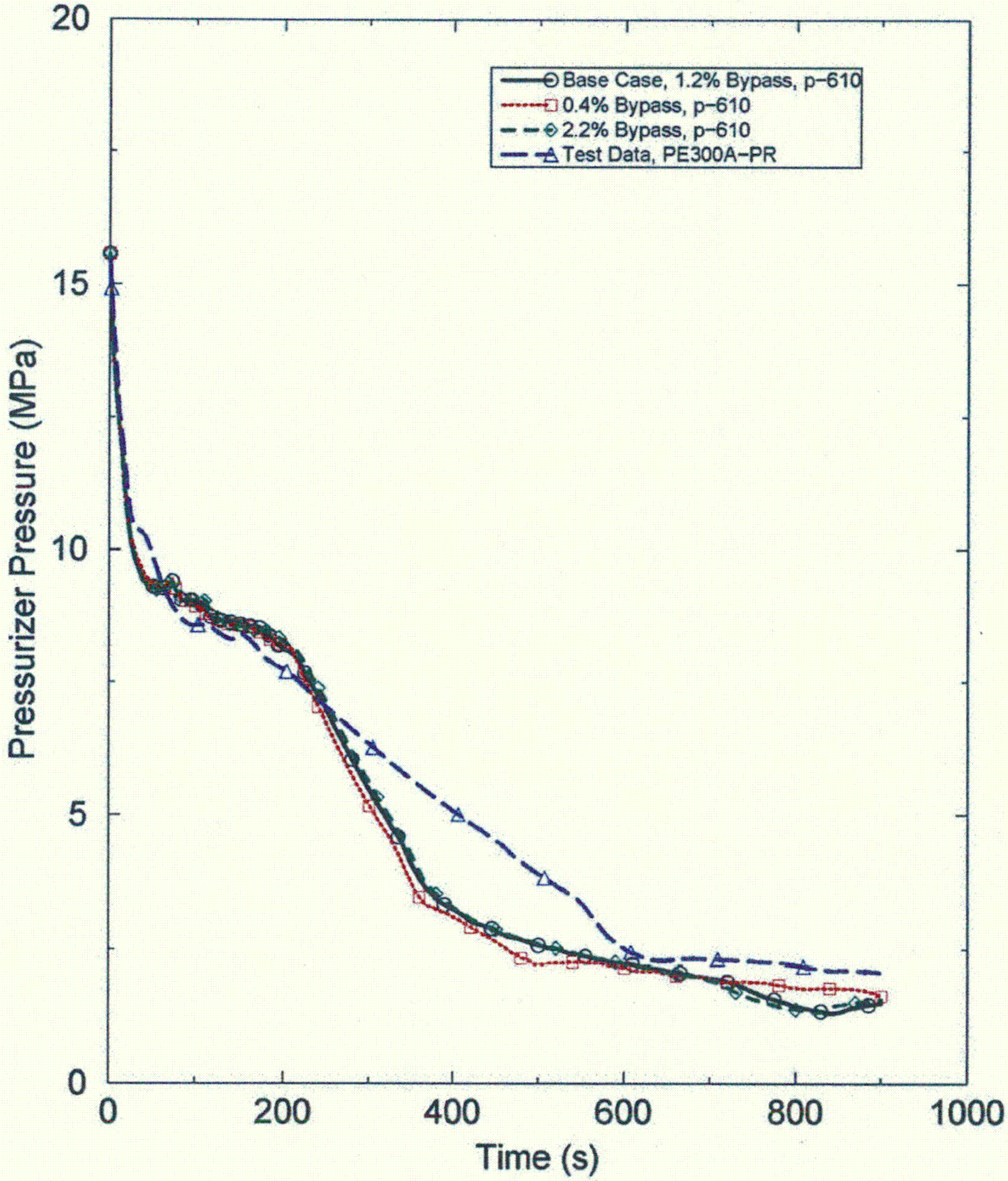


Figure 2-4: Intact Loop – Steam Generator Tube Up-side Level, ROSA-IV Core Bypass Sensitivity Study

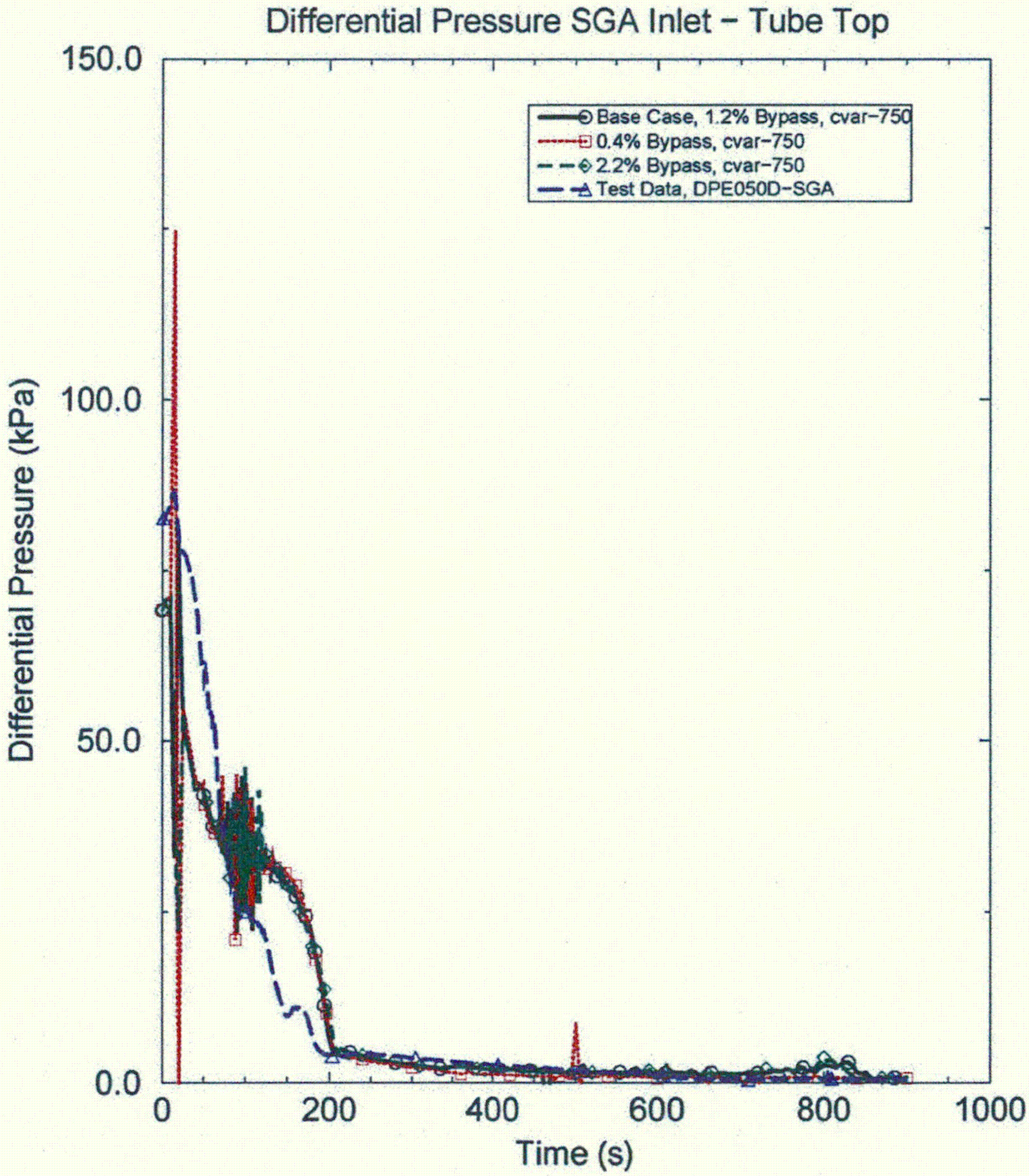


Figure 2-5: Broken Loop – Steam Generator Tube Up-side Level, ROSA-IV Core Bypass Sensitivity Study

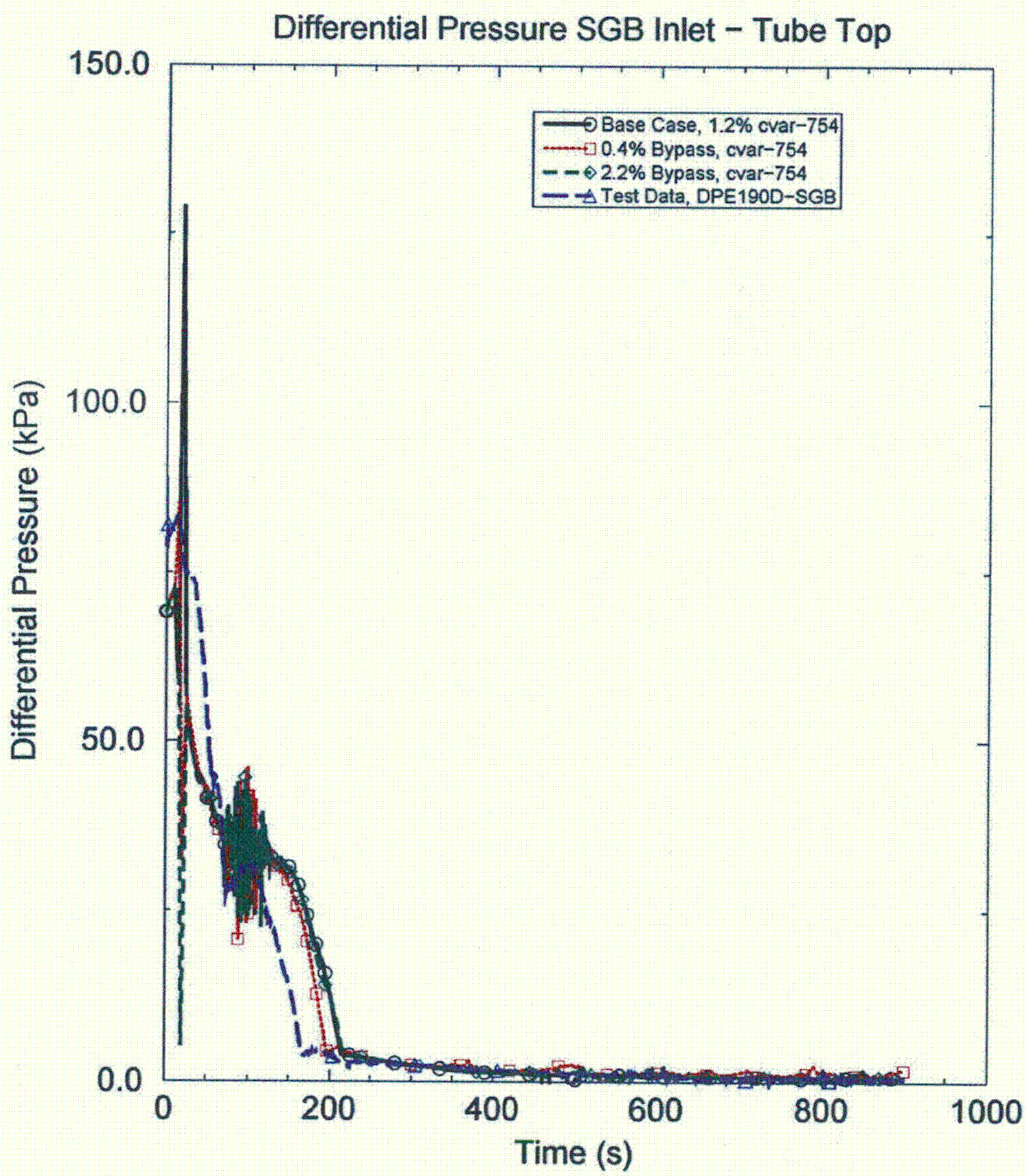


Figure 2-6: Intact Loop – Loop Seal Up-side Level, ROSA-IV Core Bypass Sensitivity Study

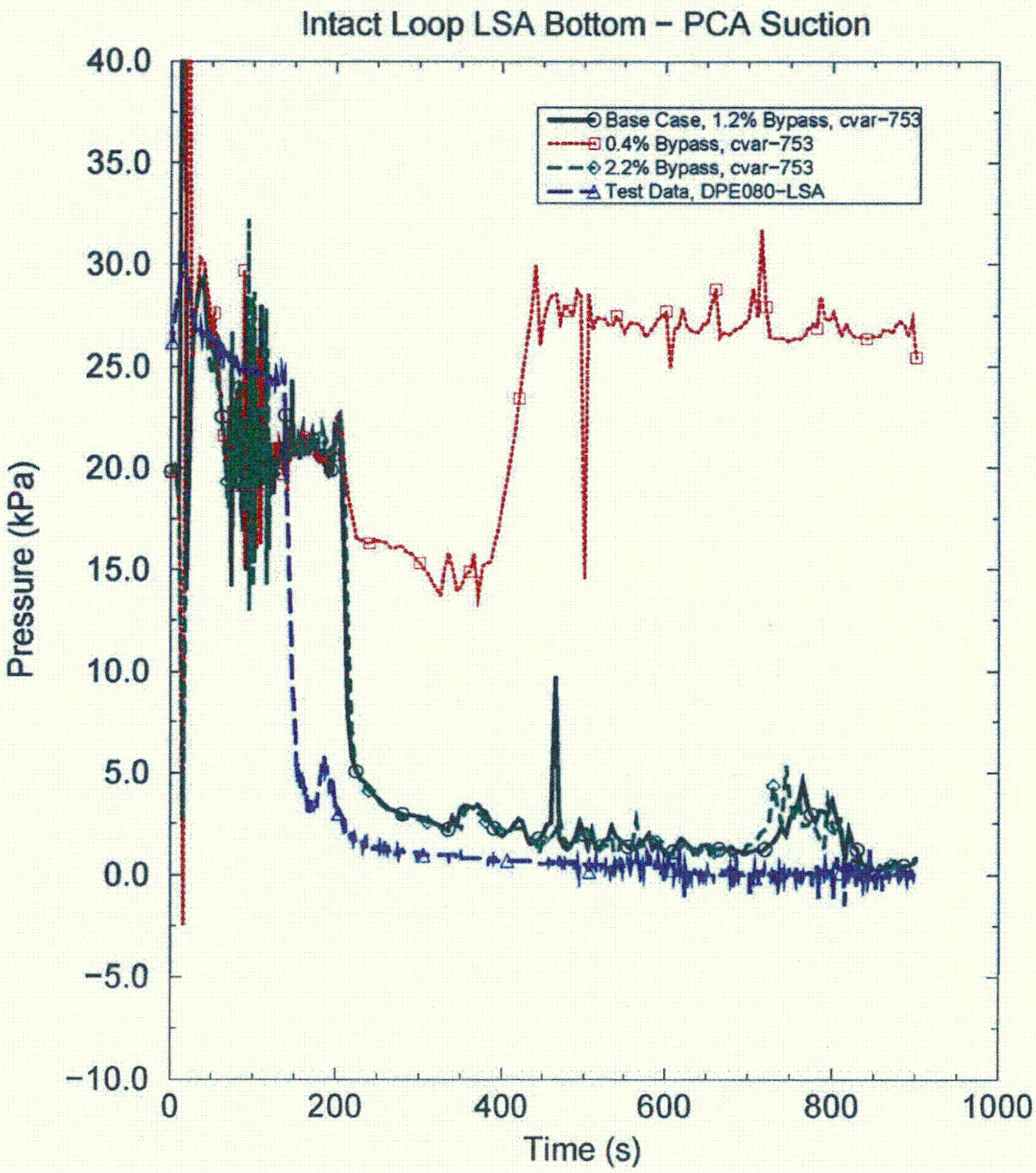


Figure 2-7: Broken Loop – Loop Seal Up-side Level, ROSA-IV Core Bypass Sensitivity Study

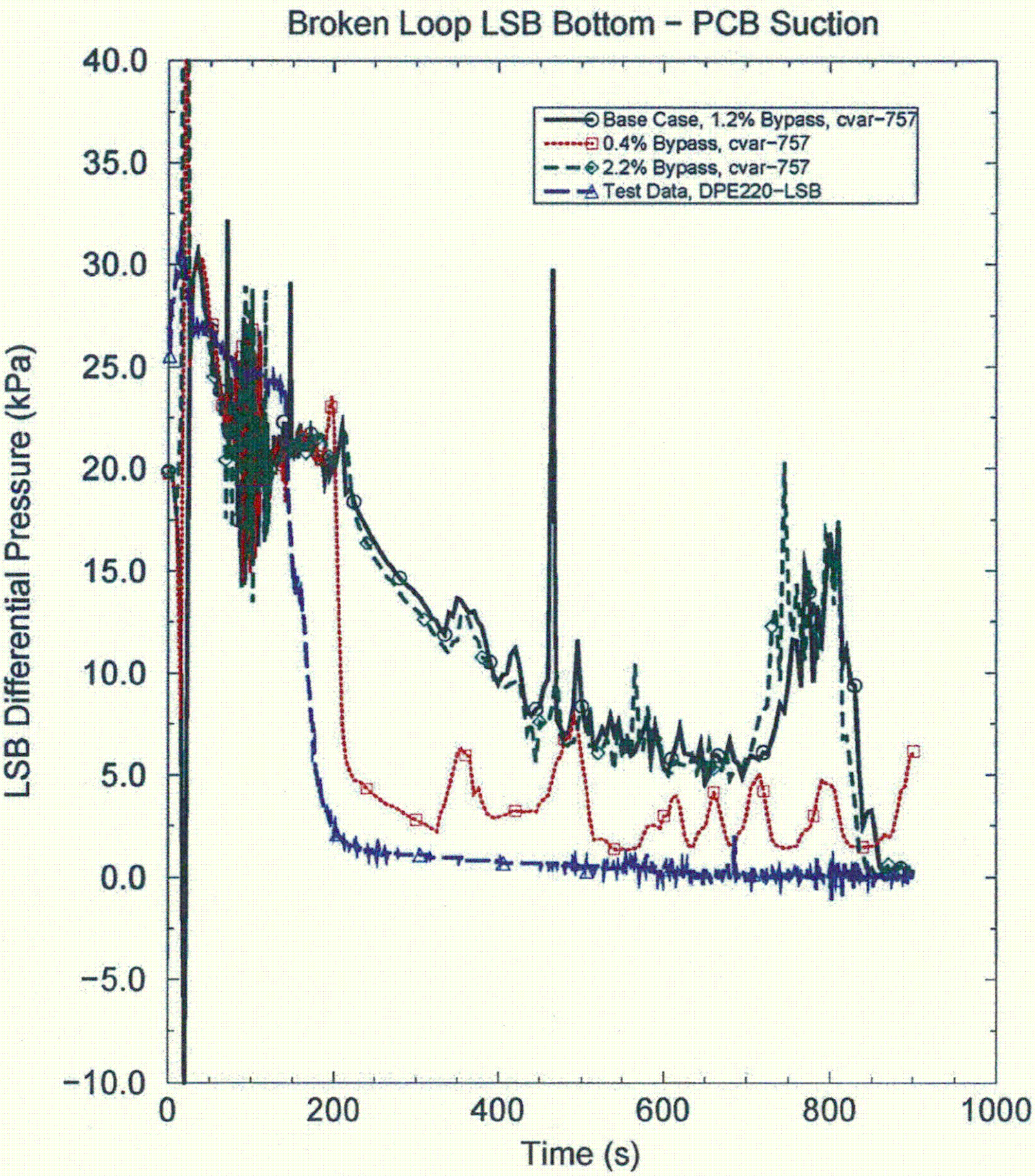


Figure 2-8: Core Level, ROSA-IV Core Bypass Sensitivity Study

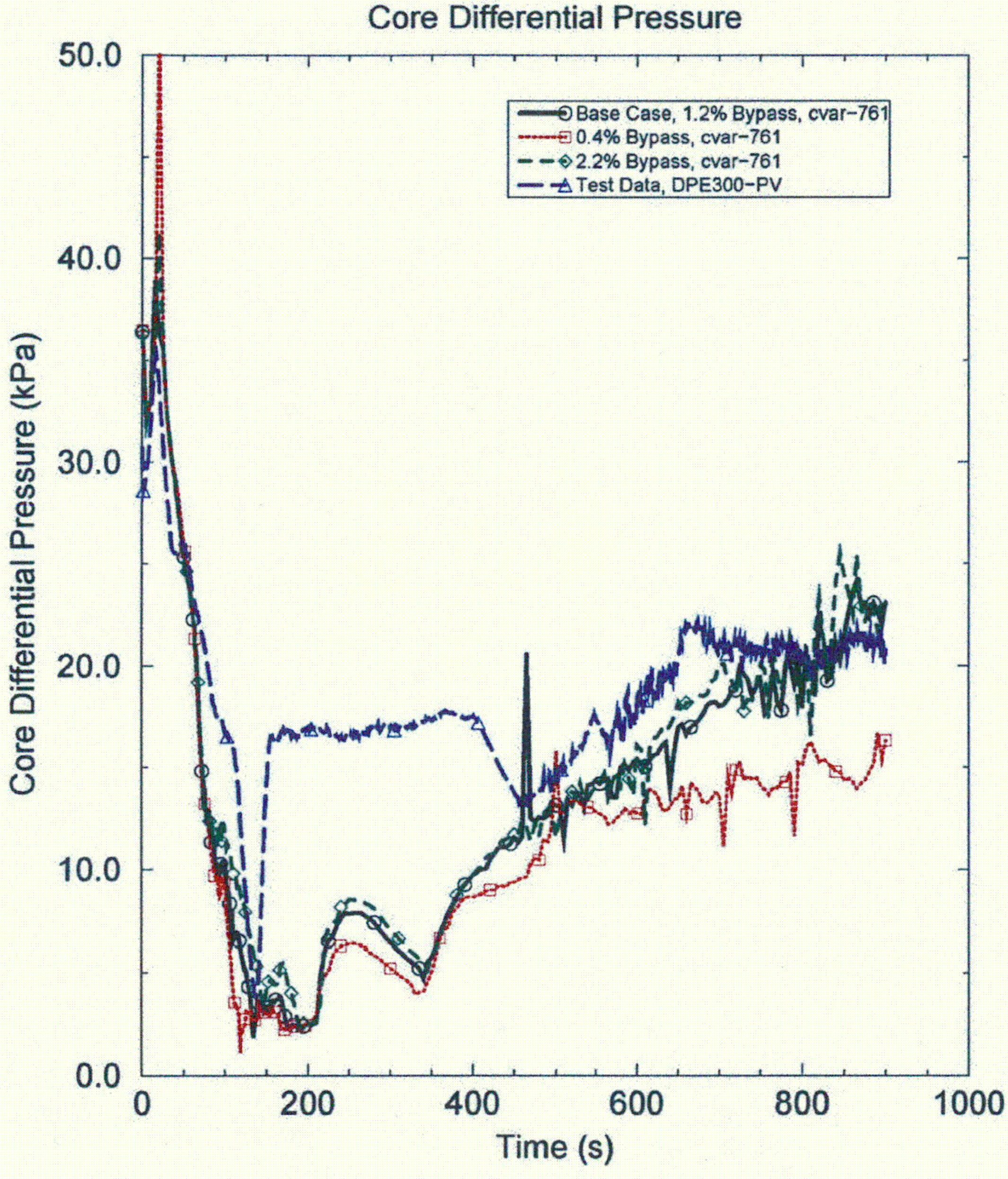
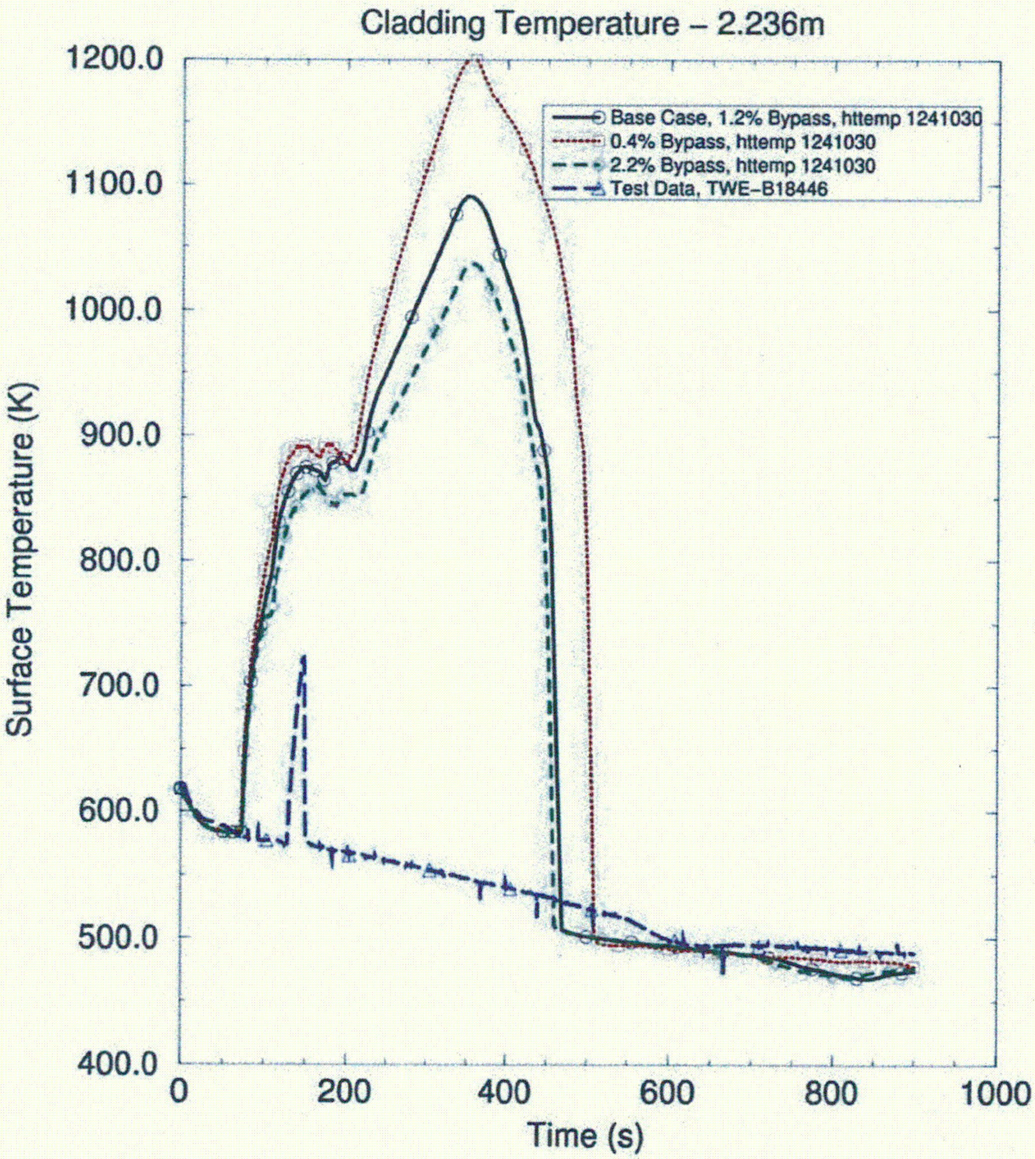


Figure 2-9: Peak Cladding Temperature, ROSA-IV Core Bypass Sensitivity Study



RAI 3:

3.1 Please demonstrate that the downcomer modeling for the case when the reactor coolant pumps (RCPs) are operating capture the proper hydraulic phenomena.

a) Should the downcomer levels decrease toward the bottom of the downcomer (cross-over to the lower plenum), demonstrate that the model simulates the two-phase flow communication properly between the downcomer and lower plenum as the fluid transitions from a low quality two-phase mixture to vapor.

b) Also, as vapor is pumped into the down comer from the cold legs describe how the code models the penetration of vapor into the downcomer liquid region at the surface and show that entrainment of downcomer liquid as vapor passes from the intact loops is properly accounted for in the model.

3.2 Please justify the applicability of the S-RELAP5 physical models and modeling techniques to SBLOCAs with the RCP running.

a) Please present validation of the model against integral and separate effects test data.

Response to RAI 3:

AREVA benchmarked the S-RELAP5 code against the LOFT L3-6 test to justify the applicability of the computer code S-RELAP5 (SR-5 in figure legends) physical models and modeling techniques to small break loss of coolant accidents (SBLOCAs) with the reactor coolant pumps running. The test simulates a 2.5 percent small break (4 inch equivalent diameter) in the cold leg of a large pressurized water reactor (PWR).The accumulators and low pressure injection system (LPIS) were not activated during the test. Safety injection water was provided directly to the downcomer by means of the high pressure injection system (HPIS). The reactor coolant pumps were not tripped until the end of the test.

AREVA used the LOFT L8-1 test for a S-RELAP5 code benchmark to capture the transition from a two-phase mixture to vapor in the downcomer and the communication between the downcomer and lower plenum. The LOFT L8-1 is an extension of the test L3-6 and was designed to investigate the core response following core uncovery. The test's objective was to allow the reactor vessel liquid level to drop below the top of the core and to produce a fuel rod heat-up. This was accomplished by terminating safety injection and tripping the pumps at the end of test L3-6.

The S-RELAP5 benchmark demonstrates the code's ability to accurately simulate the overall system response following a 4-inch equivalent diameter break in the cold leg with the primary coolant pumps running during the blowdown phase.

The S-RELAP5 simulation of the L3-6 test showed adequate core cooling by forced convection of a two-phase mixture during the pump running period. As the transient progressed, the void fraction in the reactor coolant system (RCS) increased steadily, reaching a stage when only highly voided fluid was pumped around the system and with liquid being present in the RCS lower regions or collecting in pockets at higher elevations.

During this period, the code captured correctly the fluid transition from a two-phase mixture to single-phase vapor in the downcomer region as well as the communication between the downcomer and the lower plenum. This is demonstrated by the comparisons of measurements and predictions of system mass inventory, coolant temperatures, primary system pressure, and fluid densities. The S-RELAP5 code accurately predicted the core two-phase mixture collapse and core uncovery, followed by fuel cladding temperature rise and peak cladding temperature after the tripping of the pumps at the start of Test L8-1.

S-RELAP5 Model of LOFT L3-6 Facility

The S-RELAP5 model of the LOFT facility is presented in Figure 3-1 and Figure 3-2.

The methodology utilized for the S-RELAP5 benchmark follows as closely as practicable the guidelines documented in EMF-2328 (P)(A), PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, and supplemented in EMF-2328 (P)(A) Supplement 1 Revision 0.

Key features of the benchmark model include the following:

Table 3-1 compares the S-RELAP5 calculated initial conditions with the conditions reached during the test. A time sequence of the important events during Tests L3-6 and L8-1 is presented in Table 3-2.

Figure 3-3 through Figure 3-13 present several calculated results and comparisons with the measured parameters.

S-RELAP5 Code Prediction vs. LOFT L3-6 Test

The S-RELAP5 simulation of the L3-6 test was started from similar initial conditions to those established in the test (Table 3-1). The transient was initiated 5.8 seconds after reactor scram. The reactor coolant pumps (RCP) continued to operate at constant speed until they were tripped at 2371.4 seconds, when the intact loop pressure reached approximately 2.3 MPa. This was followed by the termination of HPIS, which marked the end of the L3-6 test simulation and the beginning of the L8-1 test.

Figure 3-3 compares the calculated and measured pressures in the intact loop cold leg. The calculated pressure compares well with the test data. The code predicts a slightly higher pressure between 600 and 1700 seconds, but it converges with the measured data at the end of the test.

The break flow rate prediction (Figure 3-4) shows good agreement with the measured data and the predictions are within the data uncertainty band of 15 percent for 50 to 1435 seconds and +/- 0.75 kg/s for 1435 to 2400 seconds.

The predicted primary system mass inventory is shown in Figure 3-5 where it is compared to the lower and upper bound of the test data. The calculated primary system mass inventory remains within the bounds of the data. After 2100 seconds, the calculation shows that the HPIS flow rate slightly exceeds the break flow rate, which caused the system mass inventory to increase steadily. However, the calculated results still remained within the test data envelope. At the end of the L3-6 test, the calculated results and the upper test data converge to approximately 1100 kg.

Other significant variables are also predicted well by the code: Figure 3-6 shows an overlay of the calculated downcomer pressure with the measured data at approximately the same elevation; Figure 3-7 shows the downcomer temperature tracking the measured values very well.

The entire loop seal liquid inventory is depleted and the fluid at the break becomes all steam at approximately 1650 seconds. Prior to this time, the primary system mass is distributed uniformly throughout the system and the fluid in the intact loop is a homogeneous mixture. This is indicated by the similar densities in the hot and cold legs of the intact loop, as shown in Figure 3-8. Comparison of the calculated hot and cold densities with the test measured density in the intact leg (Figure 3-8) shows a good prediction by the S-RELAP5 code. As the primary pressure continues to decrease, the liquid pockets held up in the steam generator inlet plenum and the upward bend of the hot leg relocates within the hot leg, causing the rise in density shown in Figure 3-8 after 1650 seconds.

The flow at the core exit becomes highly voided due to the declining pump head and inability of the driven flow to entrain the liquid to the upper regions of the core after 1650 seconds. As a result, the liquid carried over from the lower plenum begins to accumulate in the core regions, resulting in a decrease in the core mixture quality. Figure 3-9 illustrates the impact of the liquid accumulation on the core void fraction after this time. This is also captured in Figure 3-10, which shows the collapsed core level rising approximately 0.3 meters, as the liquid from the vessel lower regions is retained in the core. The collapsed mixture level reached the bottom of the active core (Figure 3-10); however, the core remained cooled by the pumps forced flow. The measured and calculated cladding temperatures shown in Figure 3-11 follow saturation and no core heat-up occurs.

S-RELAP5 Code Prediction vs. LOFT L8-1 Test

The RCPs were tripped at 2371.4 seconds, marking the end of the L3-6 test and the beginning of the L8-1 test. The reactor vessel liquid mixture collapsed to the lower core elevations (approximately 0.5 m from the core bottom) (Figure 3-10), exposing the high power core regions to a steam environment at the time the pumps were tripped. The fuel rod temperature excursion began at the higher power elevations (Figure 3-13). The S-RELAP5 code predicted a Peak Cladding Temperature (PCT) of 683K at elevation 1.04 m, while the L8-1 test measured PCT value is 637K at elevation 0.75 m. Although the elevation of the calculated PCT does not exactly match the measured location, the calculated values are conservatively higher and the rate of cladding temperature rise compares well. The brief stagnation in the cladding temperature increase at approximately 2420 seconds is due to a small amount of liquid entering the peak temperature node 15, since the mixture level at the lower elevations rose briefly.

The accumulator and HPIS A and B systems were initiated after the break was isolated at 2460.2 seconds. The flow of subcooled emergency core cooling system water caused condensation in the downcomer and the lower plenum regions, as captured in Figure 3-12. Subsequently, the core was quenched within seconds (Figure 3-13) and the accumulator flow was terminated marking the end of the L8-1 test.

Table 3-1: Initial Conditions for Test LOFT L3-6

Parameter	Test L3-6	S-RELAP5
Core Power (MWt)	50	50
Hot Leg Pressure (MPa)	14.87 +/- 0.14	15.00
Hot Leg Temperature (K)	577.1 +/- 1.8	578.1
Cold Leg Temperature (K)	557.9 +/- 1.1	558.9
Mass Flow Rate (kg/s)	483.3+/- 2.6	483.3
Pressurizer Level (m)	1.18 +/- 0.11	1.18
Steam Generator Secondary Pressure (MPa)	5.57 +/- 0.06	5.58
Steam Generator Secondary Level (m)	0.22 +/- 0.03	0.27
Steam Flow Rate (kg/s)	27.8 +/- 0.1	26.3

Table 3-2: Sequence of Events for LOFT L3-6/L8-1 Tests

Event Time (sec)	Reactor Scram	Break Initiated	IL Voiding	RCPs Trip	HPIS A Isolated	Break Isolated	Max Clad Temperature	Accum/SI Initiated	Accum Terminated
L3-6/L8-1	-5.8	0.0	31.4	2371.4	2428.3	2460.4	~2467.0	2463.0	2506.6
S-RELAP5[1]	-5.8	0.0	29.2	2371.4	2428.4	2460.4	2474.7	2474.2	2506.6
L3-6					L8-1				

1. The S-RELAP5 timing of events presented in this table is shifted 5.8 seconds back from the actual transient run time in order to account for the reactor scram time of negative 5.8 seconds in the test (in the S-RELAP5 calculation the reactor scrammed at 0.0 seconds) to allow for a direct comparison of timing of events. The plots shown within this response are based on reactor scram at time=0.0s.

Figure 3-1: S-RELAP5 Nodalization Diagram of LOFT L3-6 Facility

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Figure 3-2: S-RELAP5 Model of LOFT L3-6 Downcomer and Core Regions

Figure 3-3: Comparison of Cold Leg Pressure for the L3-6 Test

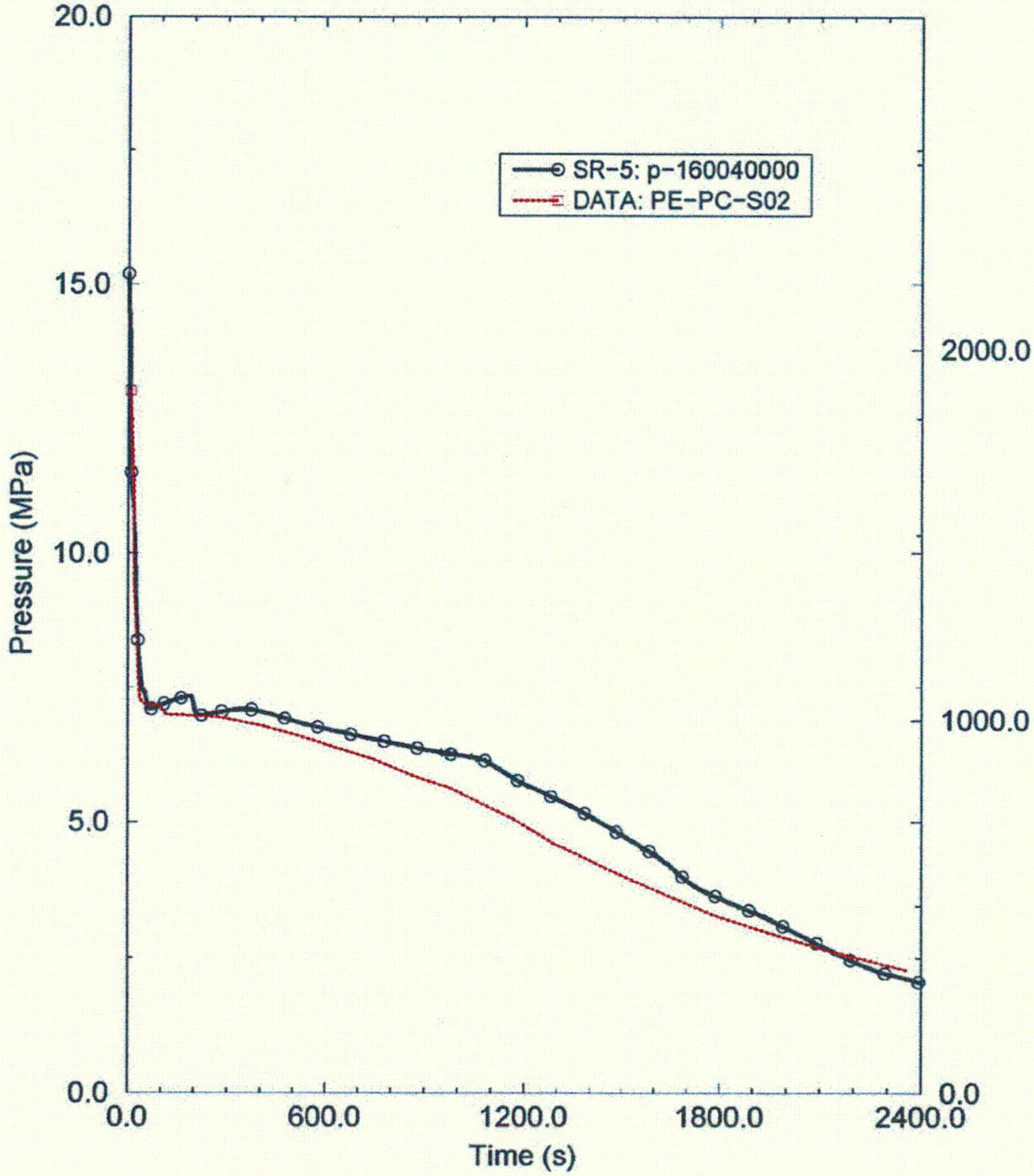


Figure 3-4: Comparison of Break Mass Flow Rate for the L3-6 Test

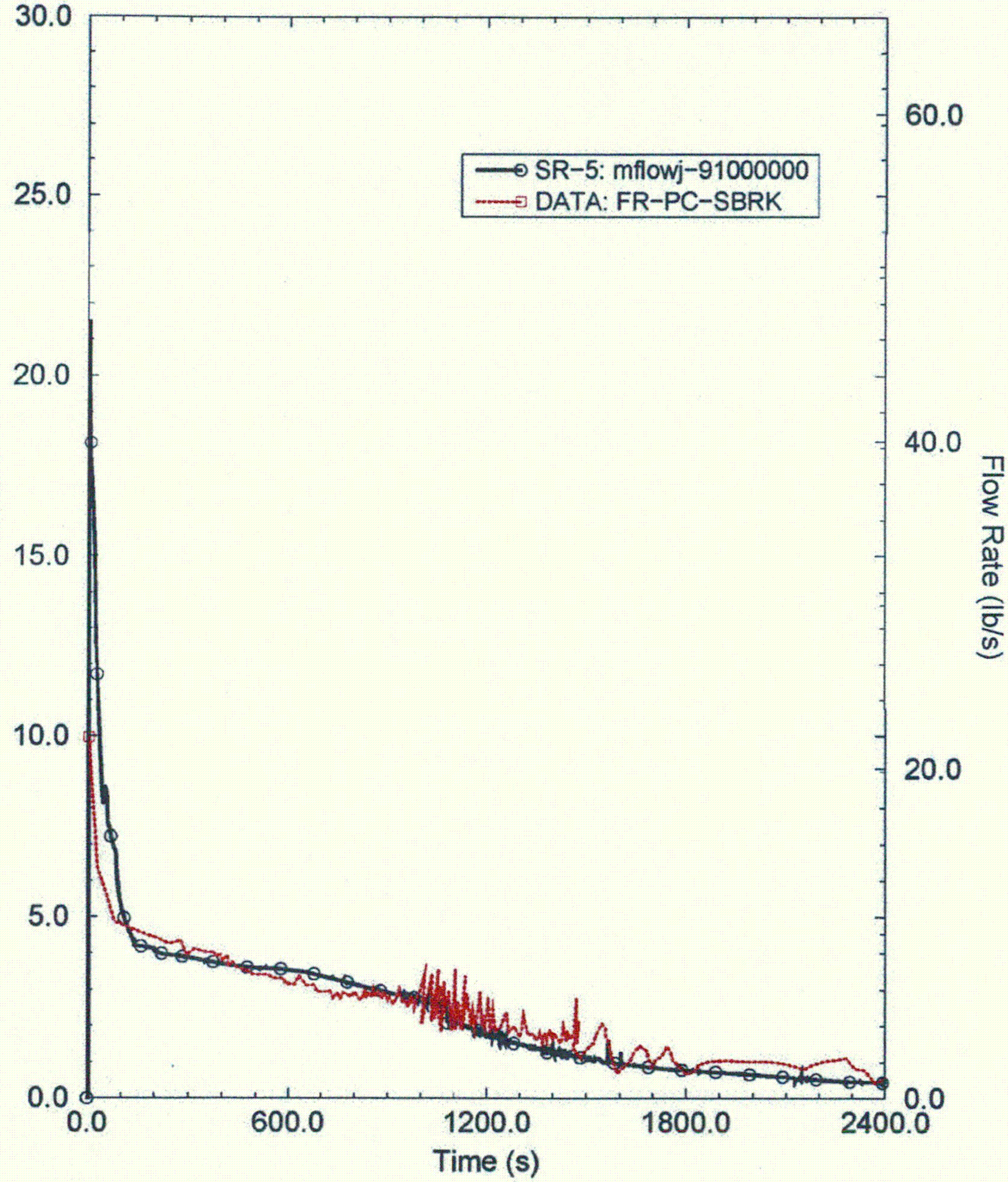


Figure 3-5: Comparison of Primary System Mass Inventory for the L3-6 and L8-1 Tests

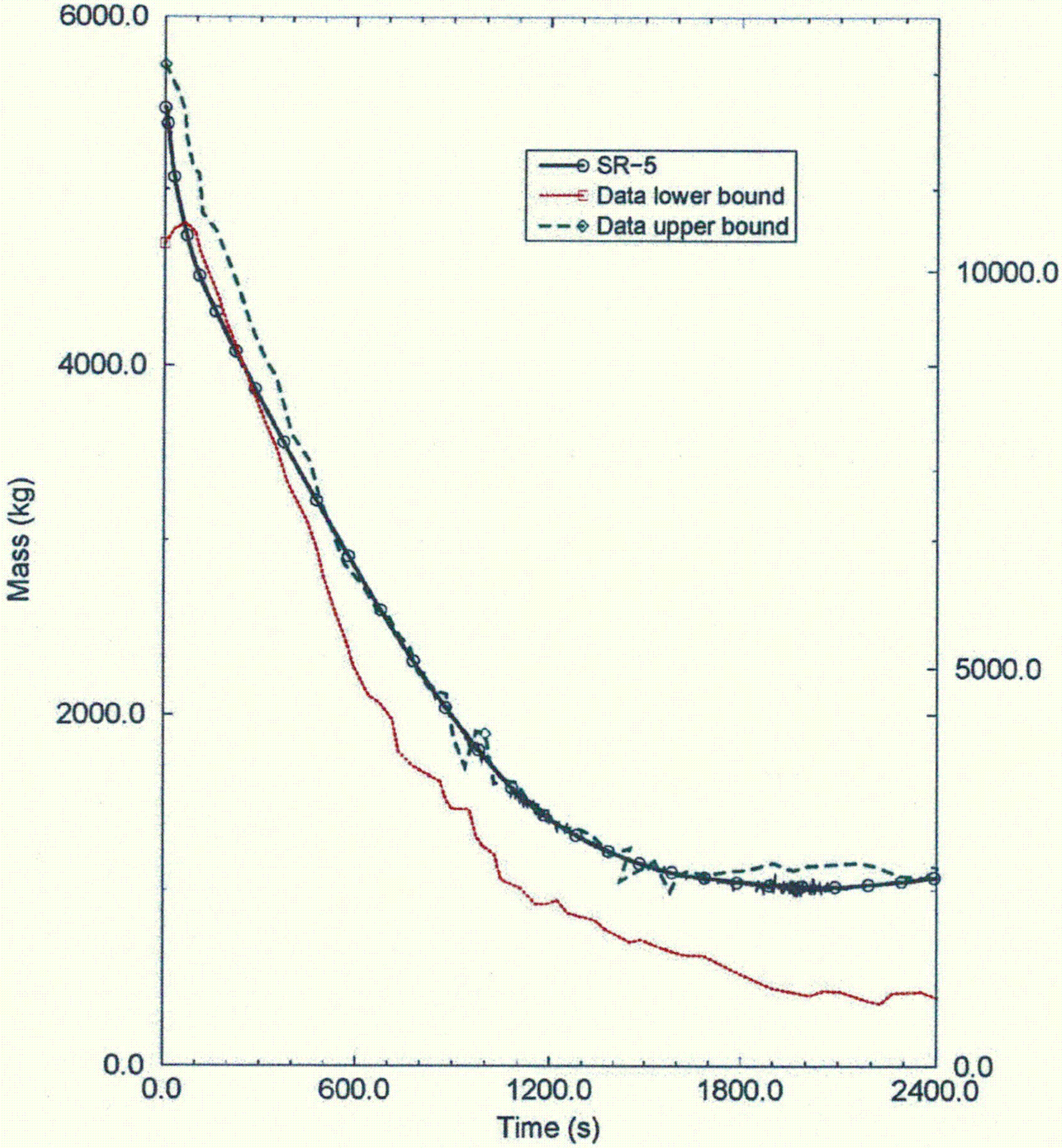


Figure 3-6: Comparison of Downcomer Pressure for the L3-6 Test

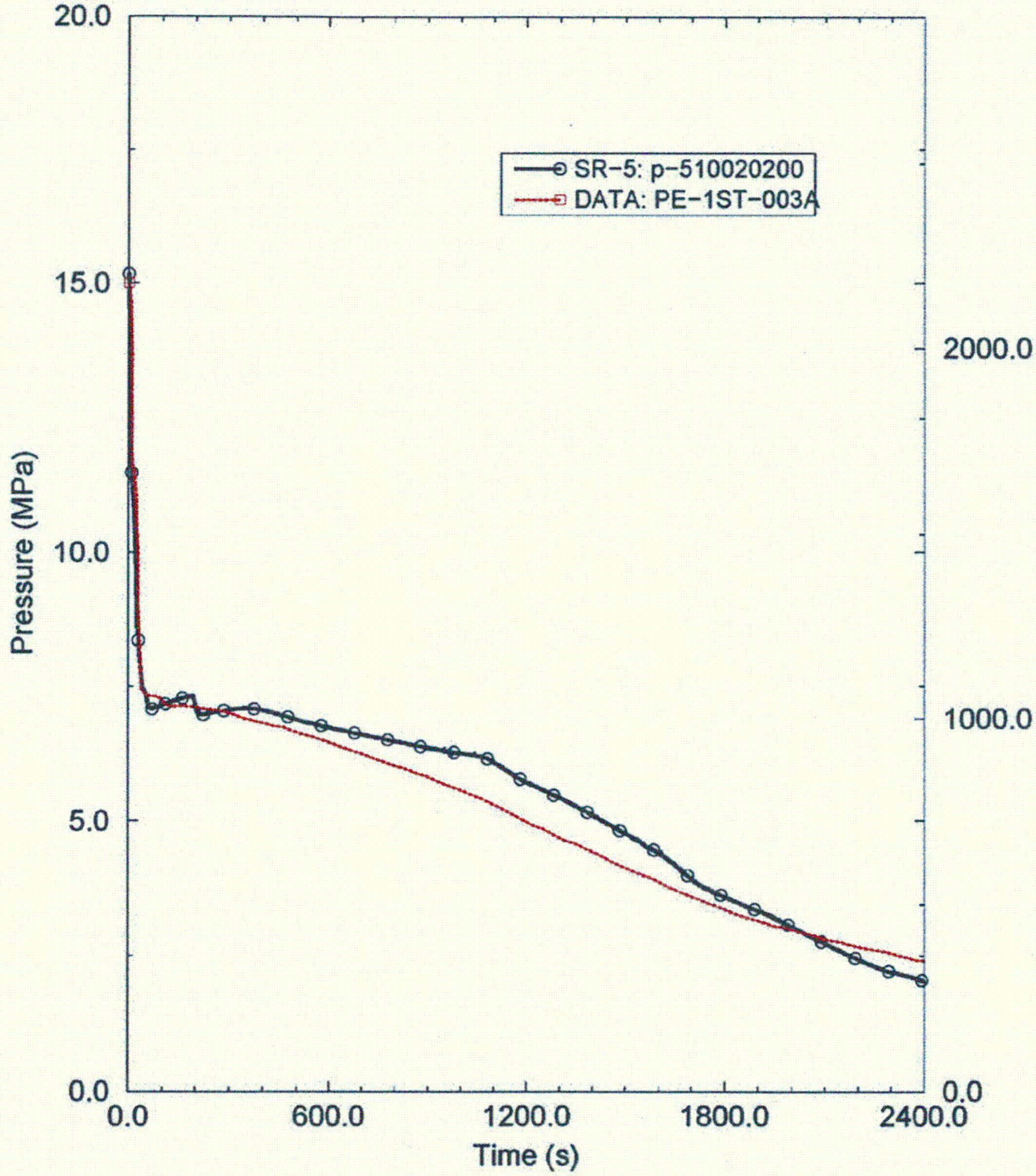


Figure 3-7: Comparison of Downcomer Temperature for the L3-6 Test

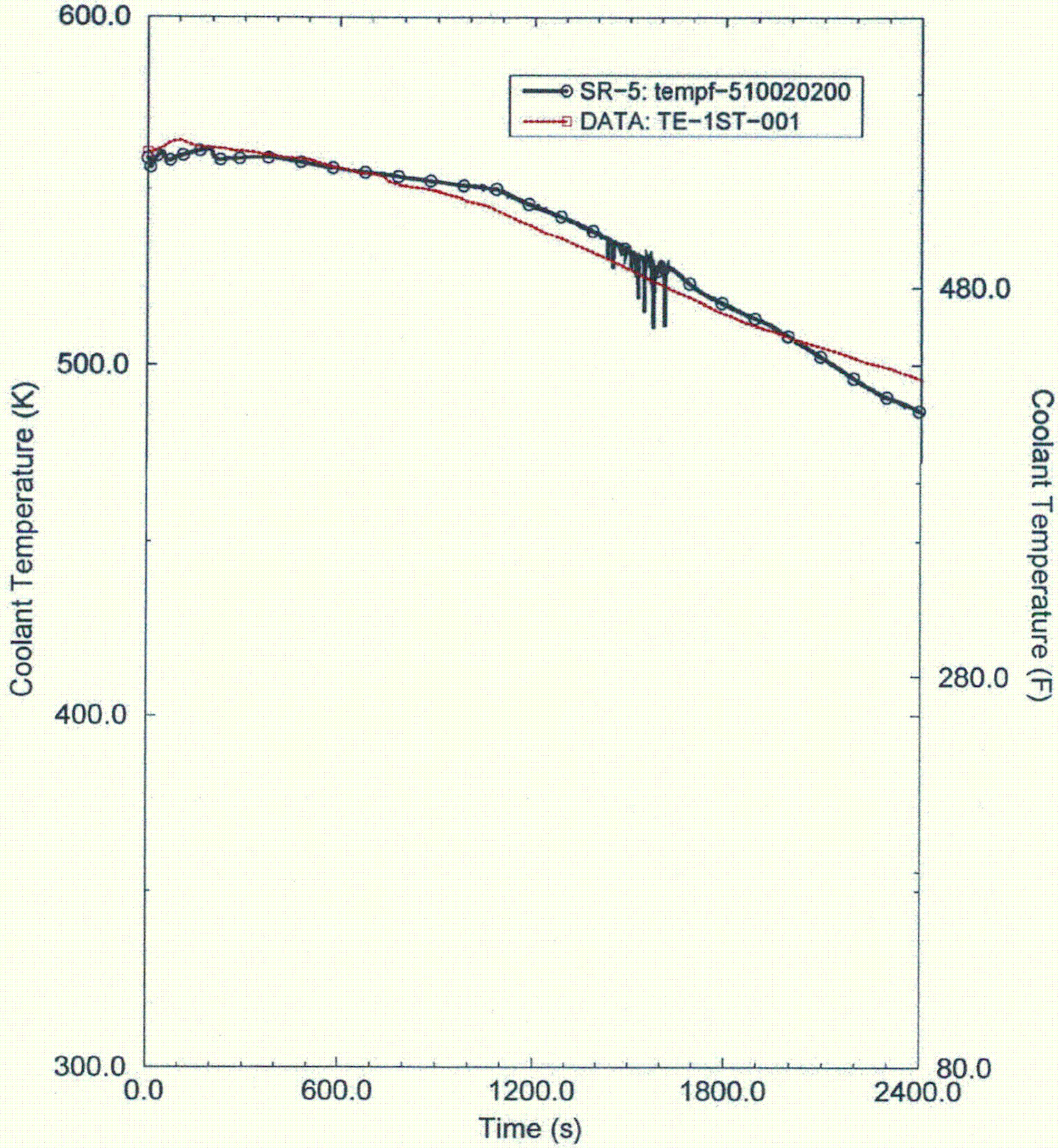


Figure 3-8: Comparison of the Intact Loop Density for the L3-6 Test

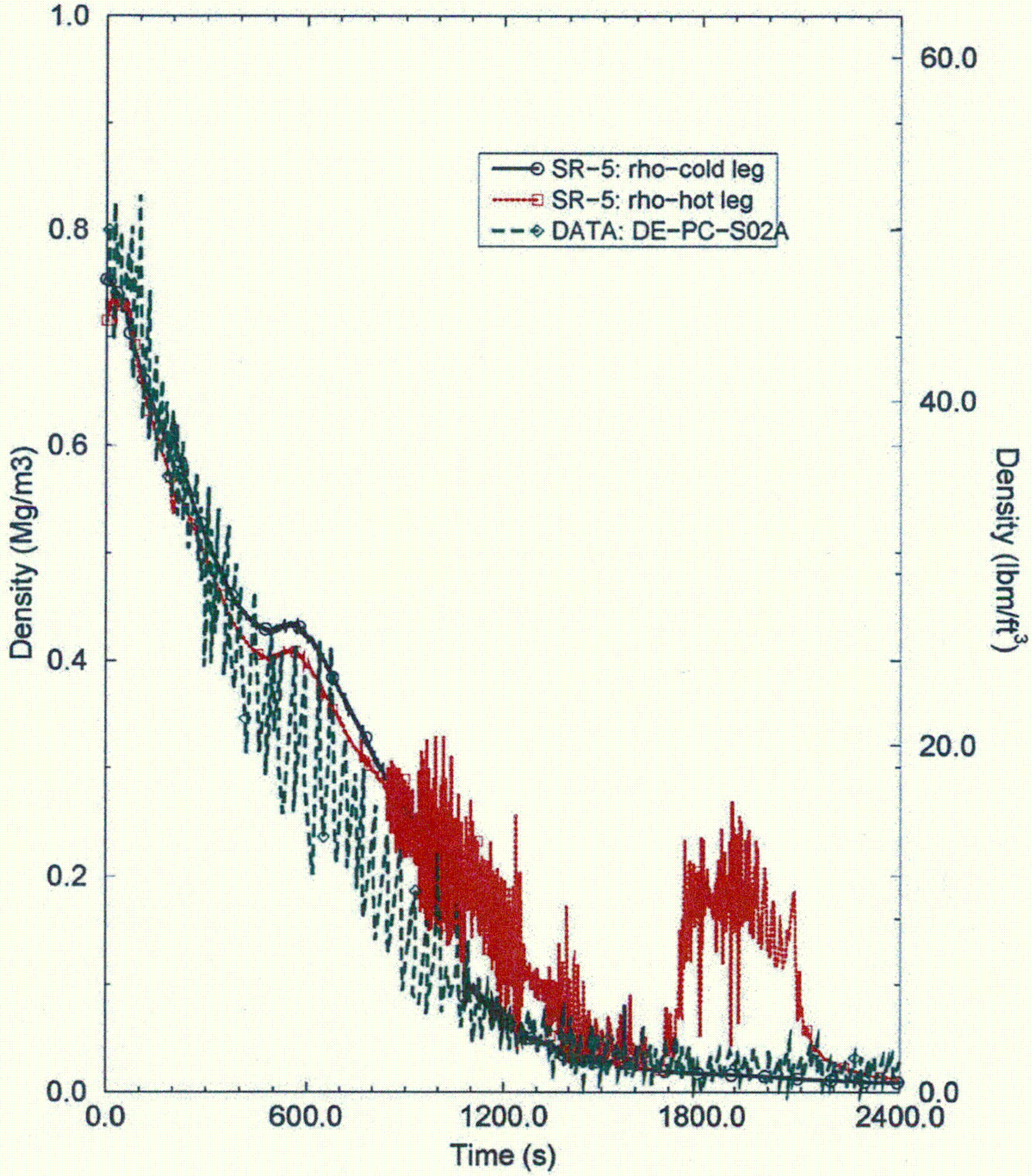


Figure 3-9: S-RELAP5 Prediction of Reactor Vessel Void Fraction during the L3-6 Test

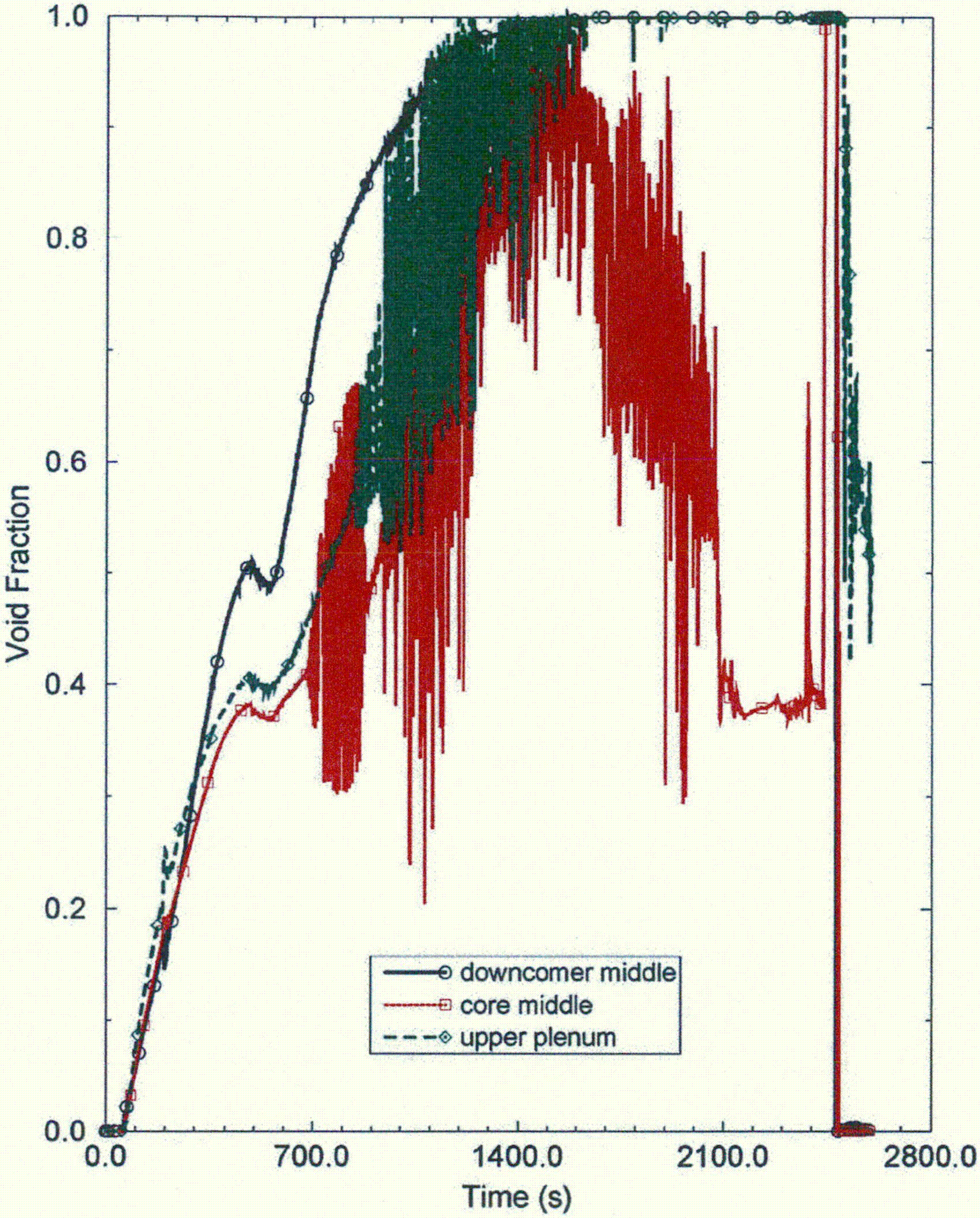


Figure 3-10: S-RELAP5 Prediction of Core Collapsed Liquid Level during the L3-6 and L8-1 Tests

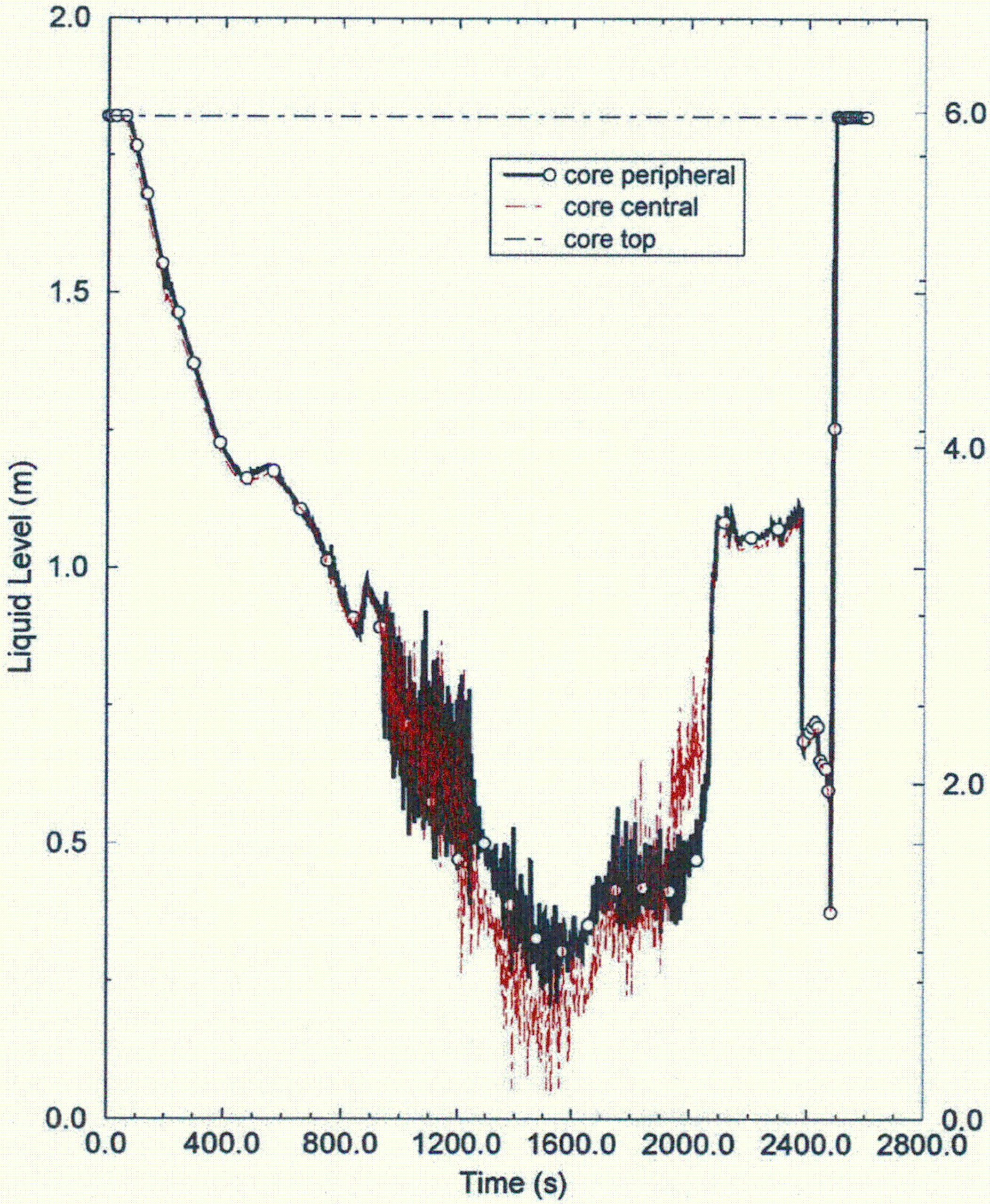


Figure 3-11: Comparison of Hot Rod Cladding Temperature for the L3-6 Test

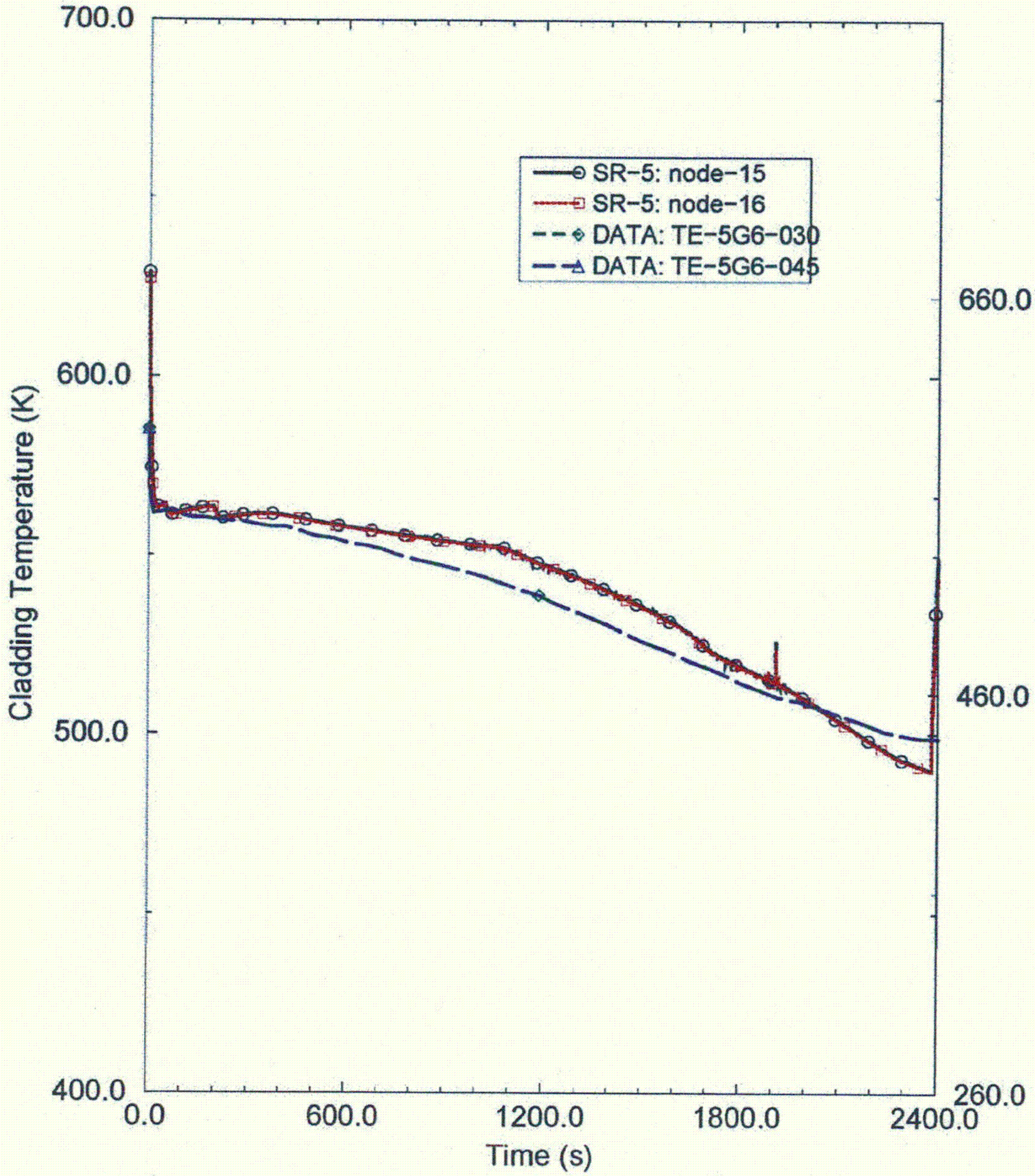


Figure 3-12: Comparison of Fluid Temperature at Vessel Bottom for L8-1 Test

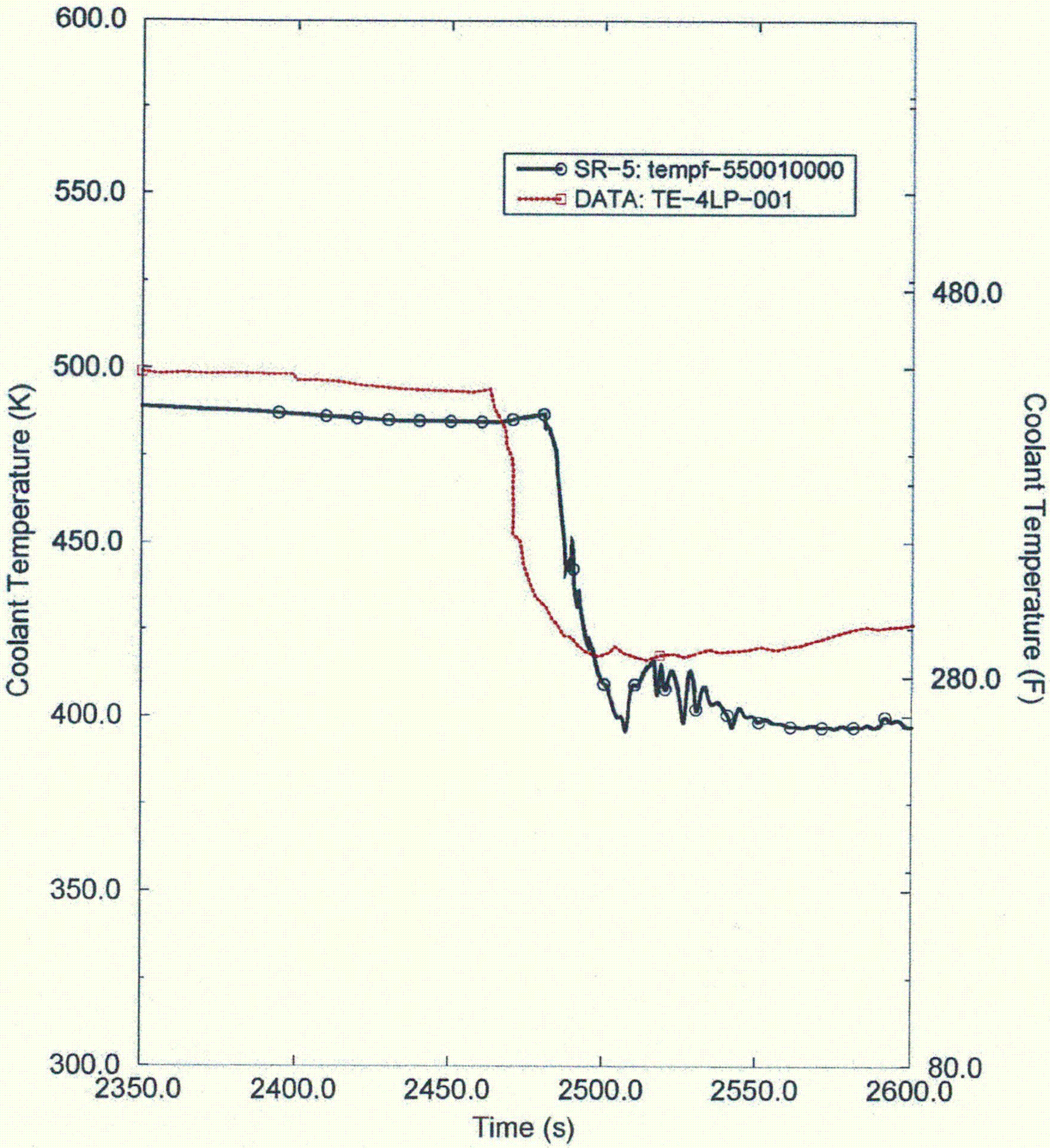
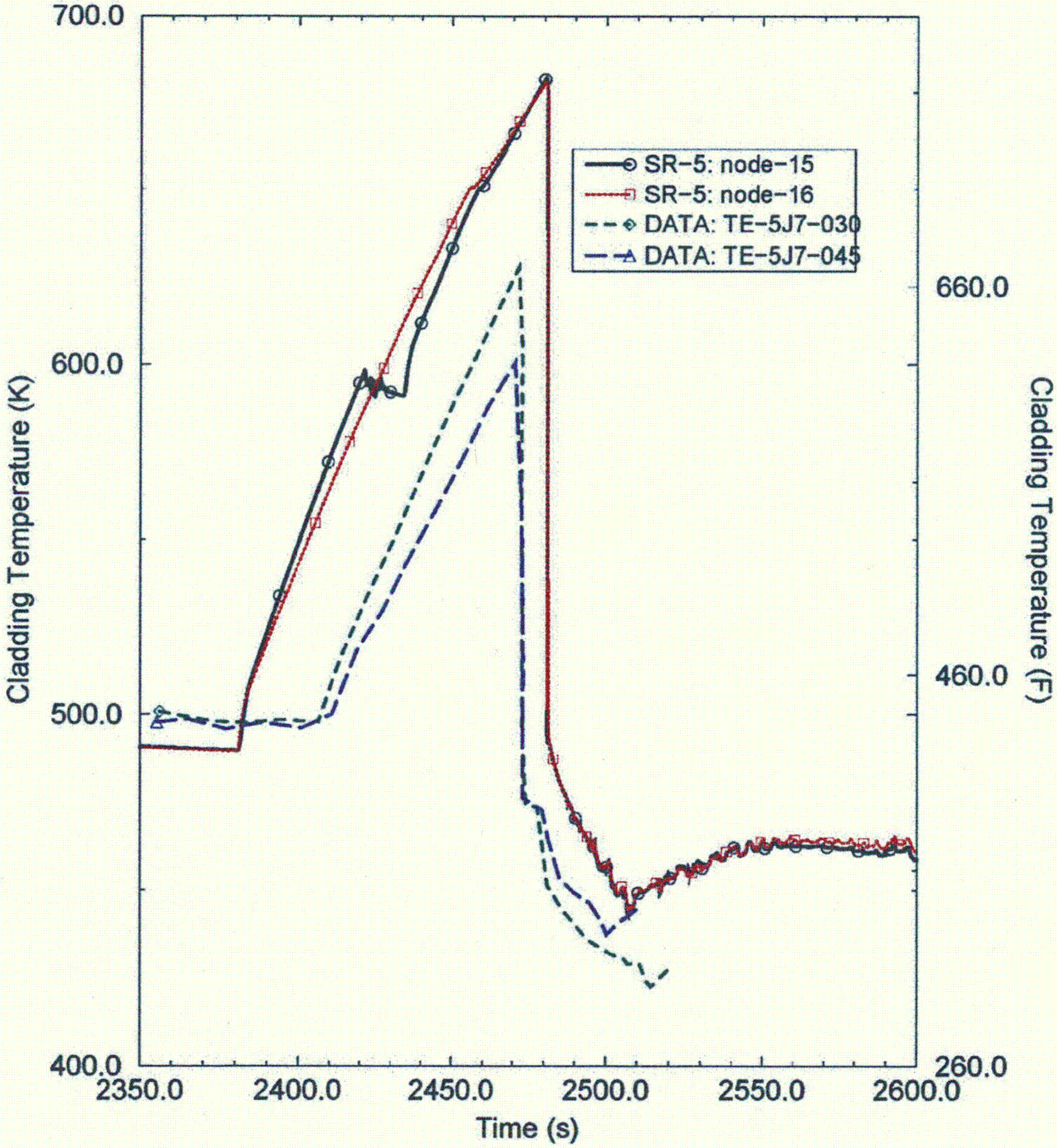


Figure 3-13: Comparison of Hot Rod Cladding Temperature for the L8-1 Test



RAI 4:

Section 7 discusses the approach to loop seal clearing following an SBLOCA. There are concerns with the limiting break analysis when multiple loop seal clearing behavior results in the suction legs. Typically, PCT is maximized when only the broken loop seal clears due to the increased resistance of vapor flow through only one loop versus multiple venting loops.

4.1 Please provide an analysis of an appropriate sample plant with and without the proposed modification. Show the impact on PCT for these two cases.

4.2 Also, describe any changes to the downcomer model that impacts entrainment of liquid out the break.

4.3 To validate the suggested approach, please show that the scaling approach to set the maximum break size for which only one loop seal clears, predicts test data for loop seal clearing in different facilities (ROSA test facility, as well as SEMISCALE Tests S-07-10 and S-07-10D, for example). Please see EGG-SEMI-5201 for these test data. These tests included loop seal effects and the impact on long term core uncovery.

a) Please provide plots of the loop seal levels as well as the core two-phase and liquid level responses.

b) Also provide the relevant transient plots to demonstrate that the key phenomena, including loop seals completely or partially cleared, are properly simulated.

(The Bethsy facility also provides additional integral data to validate the changes in EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0.)

Response to RAI 4:

The loop seal clearing approach has been modified from what was originally presented in EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0. The revised approach to be included in Section 7.1 and Section 7.3.1 of EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0 is shown in the excerpt below.

The AREVA methodology prescribes a biasing approach to promote single loop clearing in the broken leg for break sizes less than a pre-set size. As the break size increases and multiple loop seal clearings begin to occur, [

] The sample problems provided in response to RAI-5 reflect this approach.

EMF-2328PA Supplement 1, Section 7.1 will be replaced by the text in quotes below when the approved version of the topical report is issued.

> "In order to bound the possibilities discussed below, and to ensure a conservative evaluation, the S-RELAP5 based SBLOCA EM [
>
>]

The first three paragraphs in EMF-2328PA Supplement 1, Section 7.3.1 will be replaced by the text in quotes below when the approved version of the topical report is issued.

[

]

References:

12. Tasaka, K., et.al, "Loop seal clearing and refilling during a PWR small-break LOCA" Proceedings of the U.S. Nuclear Regulatory Commission Sixteenth Water Reactor Safety Information Meeting, NUREG/CP-0097 Vol. 4

13. Liebert, J., Emmerling, R. "UPTF Experiment Flow Phenomena during Full-scale Loop Seal Clearing of a PWR" Nuclear Engineering and Design 179 (1998) 51-64

14. Kim, Yeon-Sik, Cho, Seok "An experimental investigation of loop seal clearings in SBLOCA tests" Annals of Nuclear Energy 63 (2014) 721–730

Response to RAI 4.1:

A CE 2x4 plant was chosen as an appropriate sample plant for this response. Figure 4-1 presents a generic CE 2x4 vessel downcomer configuration (Reference 1). The figure illustrates the naming convention adopted throughout this response to describe the loops (loop #1 is the intact loop (IL) and loop #2 is the broken loop (BL)) and cold leg locations. Cold legs 1A and 1B belong to the intact loop. Cold legs 2A and 2B belong to the broken loop.

To address RAI 4.1, three analyses are compared

- *1B_2B_Clearing Case* (multiple-loop seal clearing): [

  ]

- *2B_Clearing Case* (single-loop seal clearing): [

  ]

- *Base Case*: [

  ]

Table 4-1 shows the PCT values for the multiple-loop seal clearing cases (*Base Case* and *1B_2B_Clearing Case*) and for the single-loop seal clearing case (*2B_Clearing Case*). For the *Base Case* where intact legs 1B and 2A are allowed to clear, an increase in PCT of almost 300°F is predicted when compared to the case for which the intact leg 1B and the broken leg 2B clear. An increase of 40°F in PCT is seen when only the broken leg 2B is allowed to clear in comparison to the case where the broken leg 2B and the intact leg 1B are allowed to clear.

The vapor released at the break is expected to have less liquid content in cases where the broken leg clears when compared to cases where multiple-loop seal are allowed to clear. This difference is due to the liquid entrained in the downcomer from the vapor of the intact loops venting at the break. When only the broken leg clears, vapor flows directly out of the break and downcomer entrainment is minimized. Therefore, the liquid inventory loss at the break would be reduced and lower PCTs are expected.

[

]

Additional Information

The analyses discussed below show that three competing phenomena influence SBLOCA transient evolution for CE 2x4 sample plants with a break size close to the threshold for single/multiple loop seal clearing: downcomer liquid entrainment, steam flow resistance for venting at the break and the amount of liquid draining from the upper plenum following multiple/single loop seal clearing. The impact of these three phenomena on the PCT depends on which and how many loop seals experience clearing through the transient.

The void fraction in the horizontal part of the pump suction leg is shown in Figure 4-2, Figure 4-3, and Figure 4-4 for the *Base Case*, the *1B_2B_Clearing Case* and the *2B_Clearing Case*, respectively. Figure 4-2 shows that the intact leg 1B and 2A clear for the *Base Case*. Figure 4-3 shows that the broken leg 2B and the intact leg 1B clear for the *1B_2B_Clearing Case*. For the *2B_Clearing Case*, only the broken leg 2B clears as shown in Figure 4-4.

Figure 4-5 shows the integrated break flow comparison for the three cases analyzed. The inventory lost at the break is impacted by the downcomer liquid entrained by the vapor of the intact loops venting at the break. In the *Base Case*, where two intact legs (1B and 2A) are allowed to clear, a larger amount of downcomer liquid entrainment is predicted when compared to the other analyses performed. This is because all of the vapor vented at the break flows through the downcomer upper portion. For the *1B_2B_Clearing Case*, where the broken leg 2B and the intact leg 1B clear, more than half of the vapor is vented at the break through leg 2B for which downcomer liquid entrainment is limited. Therefore, the amount of downcomer liquid entrainment due to vapor from the intact loop 1B venting at the break, is reduced when compared to the *Base Case* predictions. For the *2B_Clearing Case*, where only the broken leg 2B clears, the analyses show that downcomer liquid entrainment is minimized. This behavior is shown in Figure 4-5, with the *Base Case* predicting the largest mass inventory lost at the break, followed by the *1B_2B_Clearing Case* and the *2B_Clearing Case*.

The downcomer collapsed liquid level comparison, shown in Figure 4-6, confirms the trend discussed above. Figure 4-6 also shows the effect of multiple-loop seal clearing vs. single-loop seal clearing on the vapor flow resistance for venting at the break. When vapor is venting through one loop only (*2B_Clearing Case*), the vapor flow resistance for venting at the break is larger than the case with two loop seals clearing (*1B_2B_Clearing Case*). A larger vapor flow resistance creates a larger downcomer/core hydraulic head; therefore, downcomer head is increased for the *2B_Clearing Case* when compared to the *1B_2B_Clearing Case* (Figure 4-6).

After loop seal clearing, some liquid from the upper plenum and upper head is pushed into the hot leg and steam generator inlet plenum if vapor is venting through that hot leg. This behavior is shown in Figure 4-7 through Figure 4-10. After approximately 800 seconds, the mass inventory predictions for the hot legs and the steam generator inlet plenums are very similar for all cases analyzed. The primary system pressure is considerably lower and the vapor mass flow rate at the break is reduced by this time. Liquid from the hot legs can drain back into the upper plenum as shown in Figure 4-11. This coolant "reservoir" progressively falls into the core.

The core collapsed liquid level (Figure 4-12) is influenced by three physical phenomena:

- Downcomer liquid entrainment.
- Venting vapor at the break through multiple/single-loop seal clearing.
- Coolant inventory behavior above the core following multiple/single-loop seal clearing.

Downcomer liquid entrainment from the vapor of the intact loops venting at the break increases the amount of coolant lost from the vessel. Venting vapor at the break through the broken loop only increases the depth of the core uncovery because of a larger vapor flow resistance and a reduced coolant inventory draining from the upper plenum (Figure 4-11).

The core collapsed liquid level shown in Figure 4-12 shows that at approximately 1000 seconds, both the *Base Case* and the *2B_Clearing Case* predict core uncovery. Core uncovery is reached for the *1B_2B_Clearing Case* with a 200 second delay due to the combined effect of venting vapor at the break through multiple loops and a larger inventory of coolant draining from the upper plenum.

Figure 4-13 shows the PCT for the three analyses performed. The *Base Case* predicts the highest PCT due to the larger downcomer liquid entrainment. The different core heatup behavior between the *1B_2B_Clearing Case* and the *2B_Clearing Case* follows the behavior of core collapsed liquid level described above with the result that the PCT is slightly lower for the 1B_2B case.

Conclusions

The analyses discussed in this response show that when the broken leg clears then the effects of downcomer liquid entrainment, vapor flow resistance and draining of coolant from the upper plenum counter balance each other and the total effect on PCT for the break size analyzed is minimal when multiple/single-loop seal clearing cases are considered. Overall, an increase of 40°F was predicted for the single-loop seal clearing case PCT when compared to the multiple-loop seal clearing case PCT as shown in Table 4-1 and shown in Figure 4-13.

Conversely, when two intact legs are allowed to clear, as in the *Base Case*, the enhanced downcomer liquid entrainment results in a larger mass inventory being lost at the break and a larger depth of core uncovery when compared to the other two analyses discussed. The analyses show that S-RELAP5 predictions of downcomer liquid entrainment (shown to be excessively conservative in comparison with UPTF based correlations) impact the SBLOCA transient with two intact legs clearing. S-RELAP5 predicts a larger and faster core level drop, dryout and heatup with a consequent higher PCT (ΔPCT ~ 300°F) as shown in Table 4-1 and shown in Figure 4-13.

[ ]

References

1. EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

Table 4-1: Summary of the PCT for the 3.9 inch ID Break Size Cases Analyzed

Case ID	PCT (°F)	Leg Clearing
Base Case	1841	1B (in the IL) 2A (in the BL)
1B_2B_Clearing Case	1543	1B (in the IL) 2B (broken leg)
2B_Clearing Case	1583	2B (broken leg)

Figure 4-1: Generic CE 2x4 Vessel Downcomer Configuration

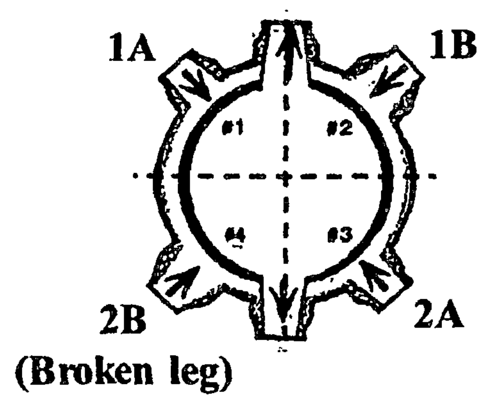


Figure 4-2: Void Fraction in the Horizontal Part of Pump Suction Leg for Base Case

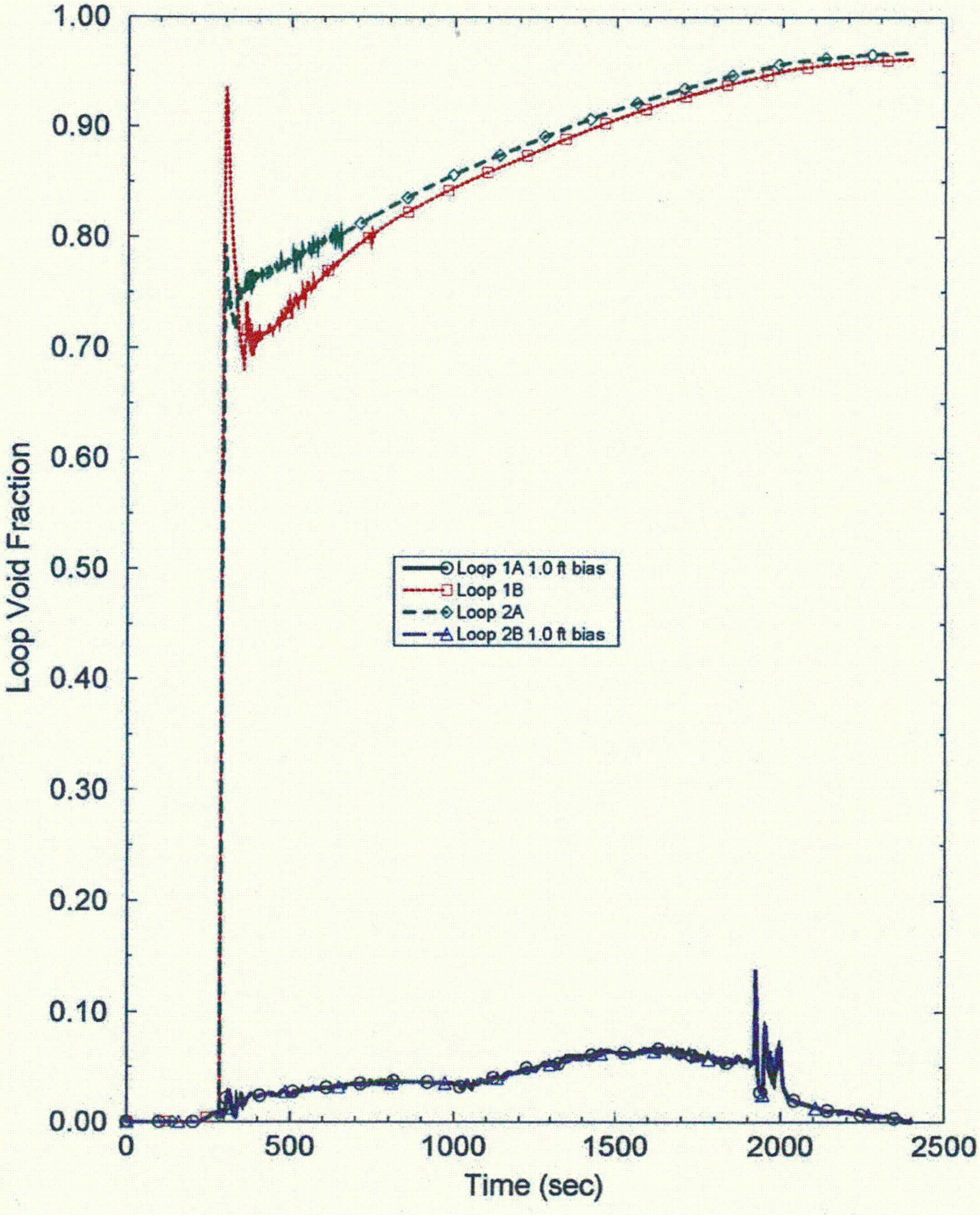


Figure 4-3: Void Fraction in the Horizontal Part of Pump Suction Leg for 1B_2B_Clearing Case

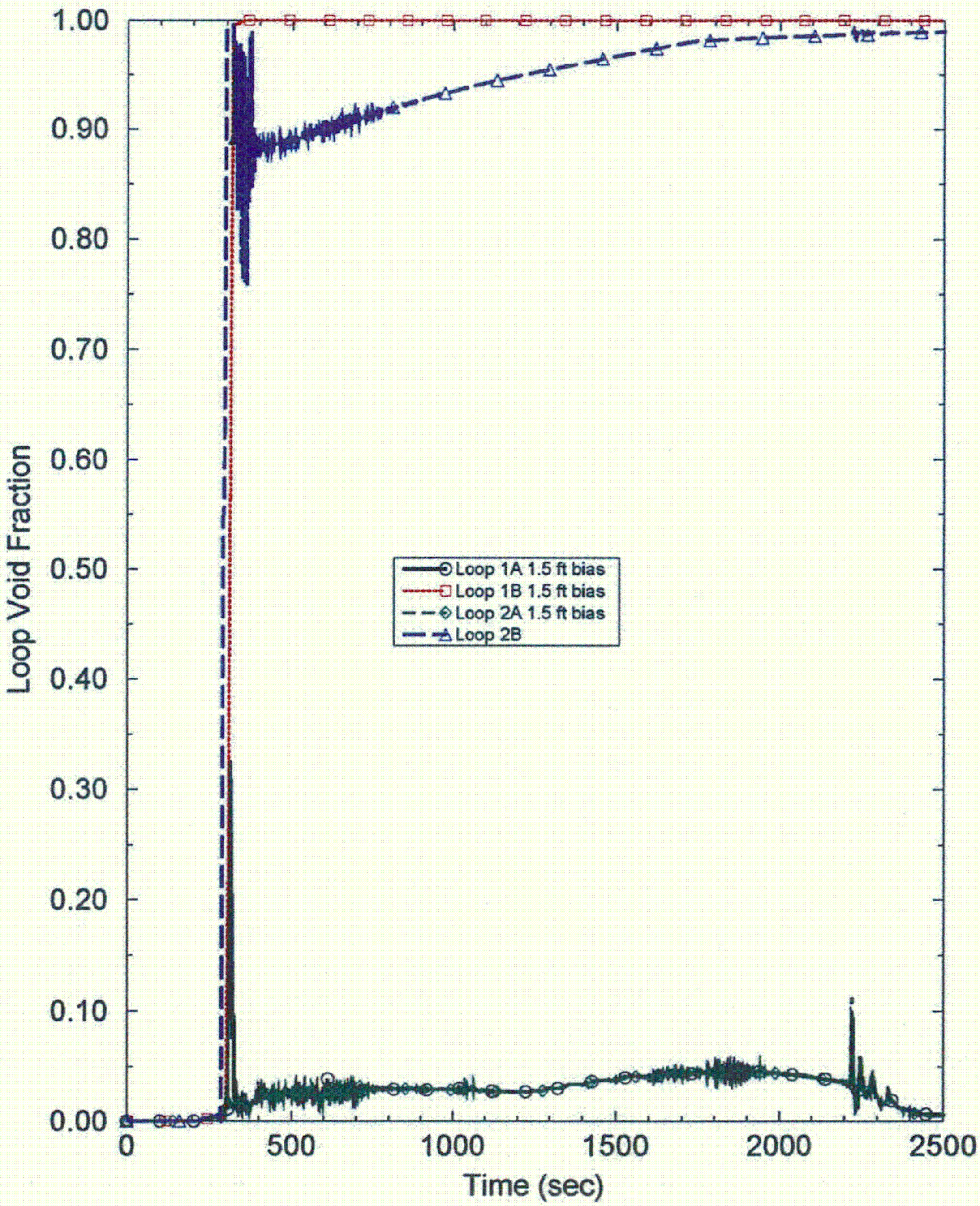


Figure 4-4: Void Fraction in the Horizontal Part of Pump Suction Leg for 2B_Clearing Case

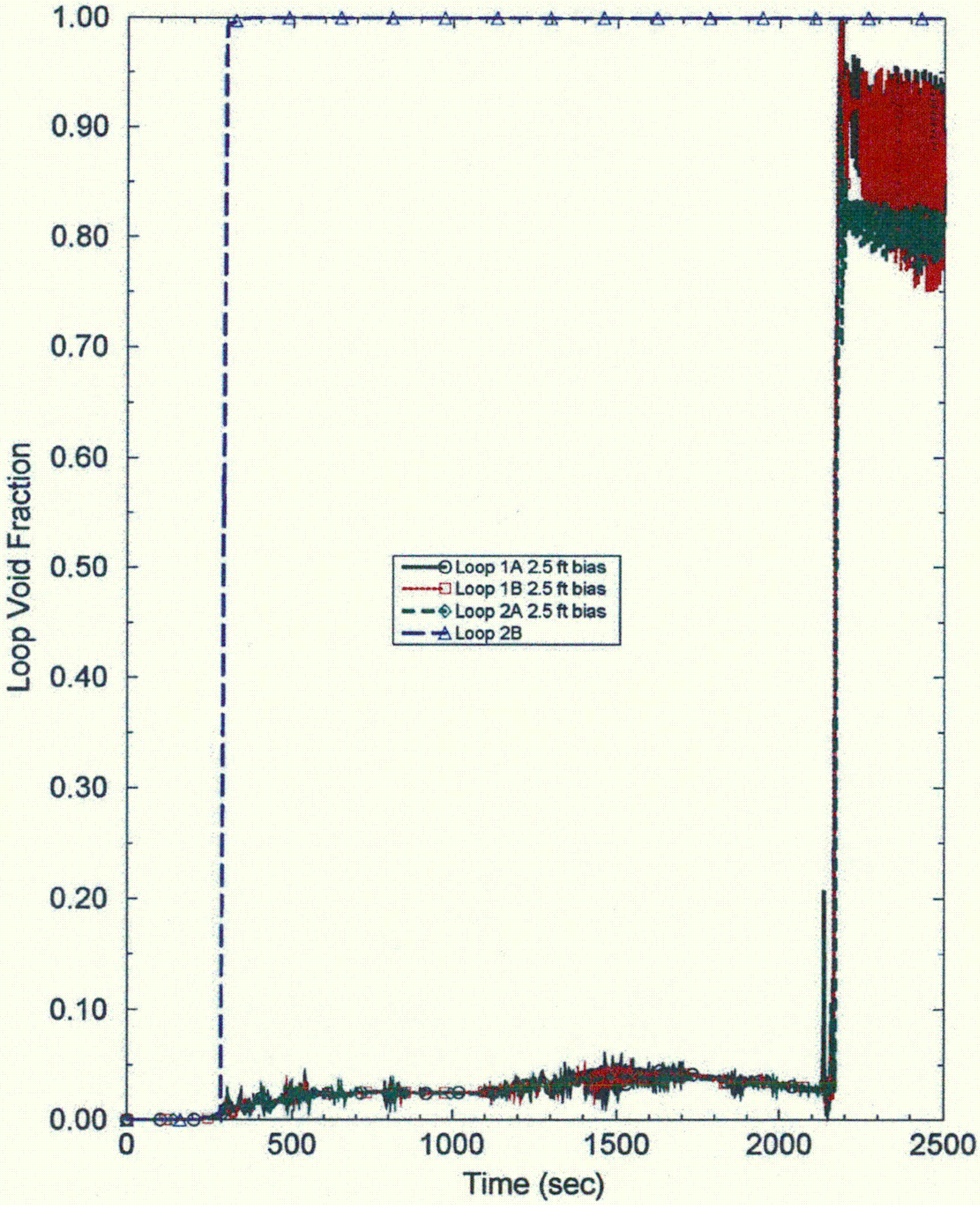


Figure 4-5: Integrated Break Flow Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

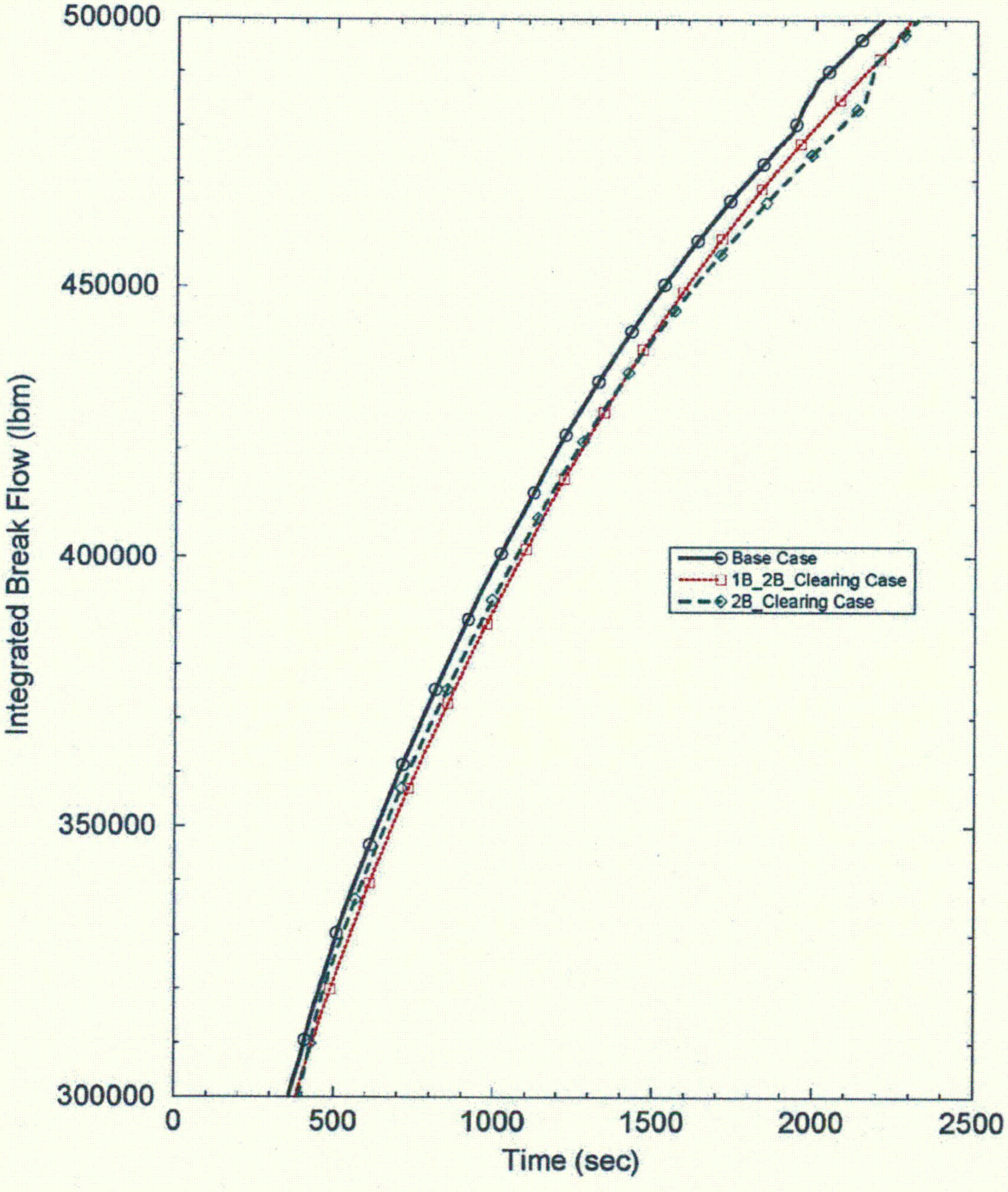


Figure 4-6: Downcomer Level Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

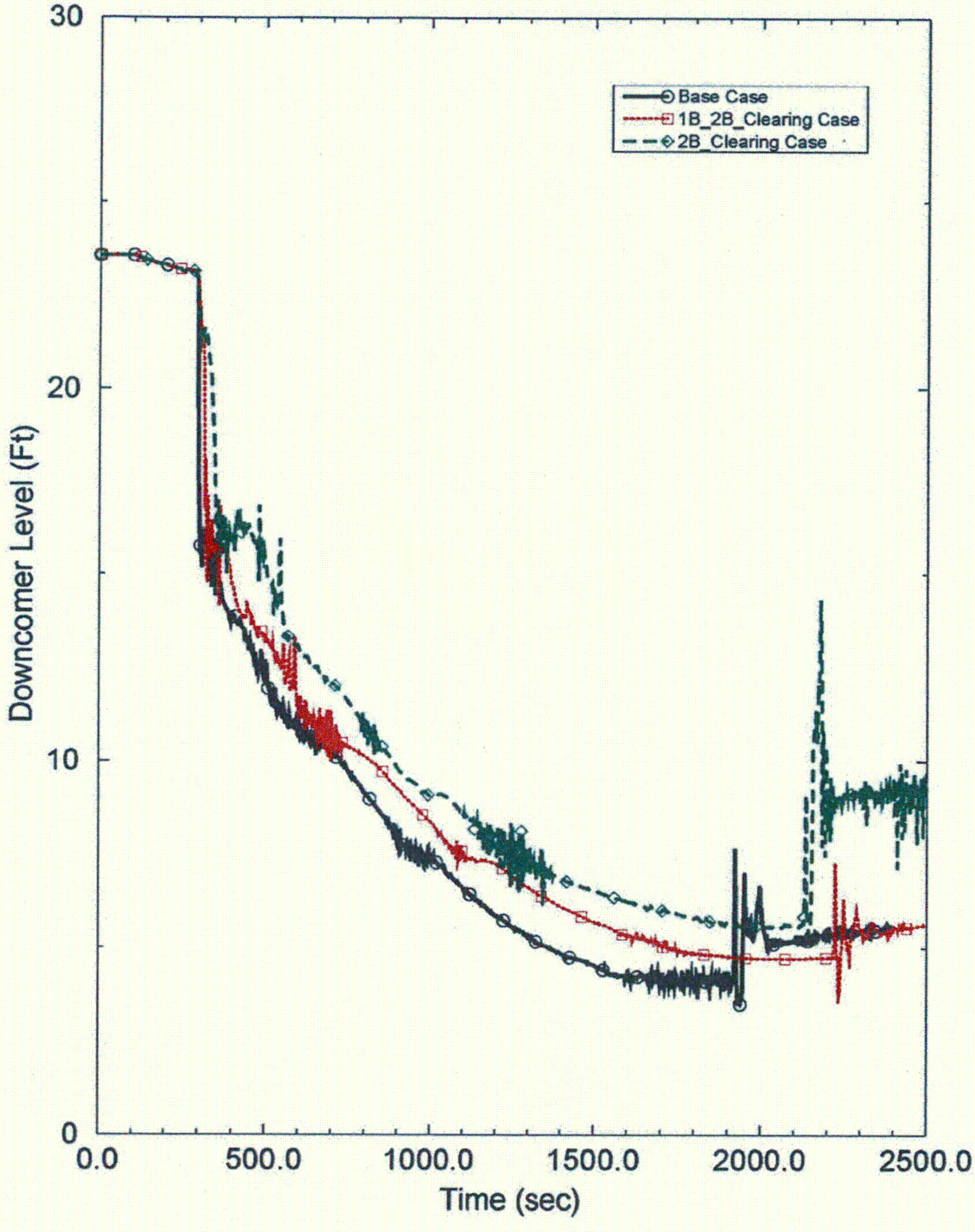


Figure 4-7: Hot Leg #1 Mass Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

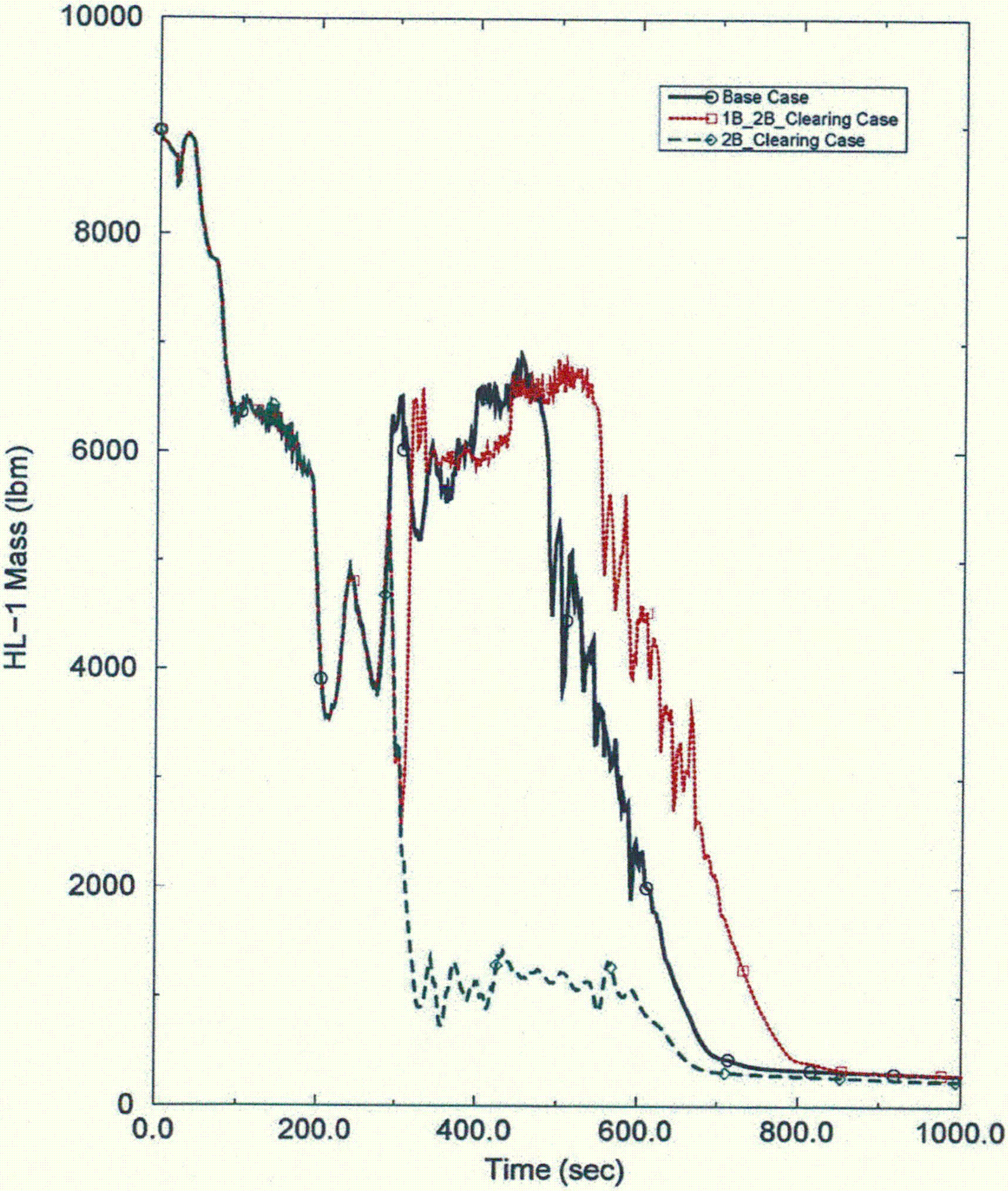


Figure 4-8: SG #1 Mass Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

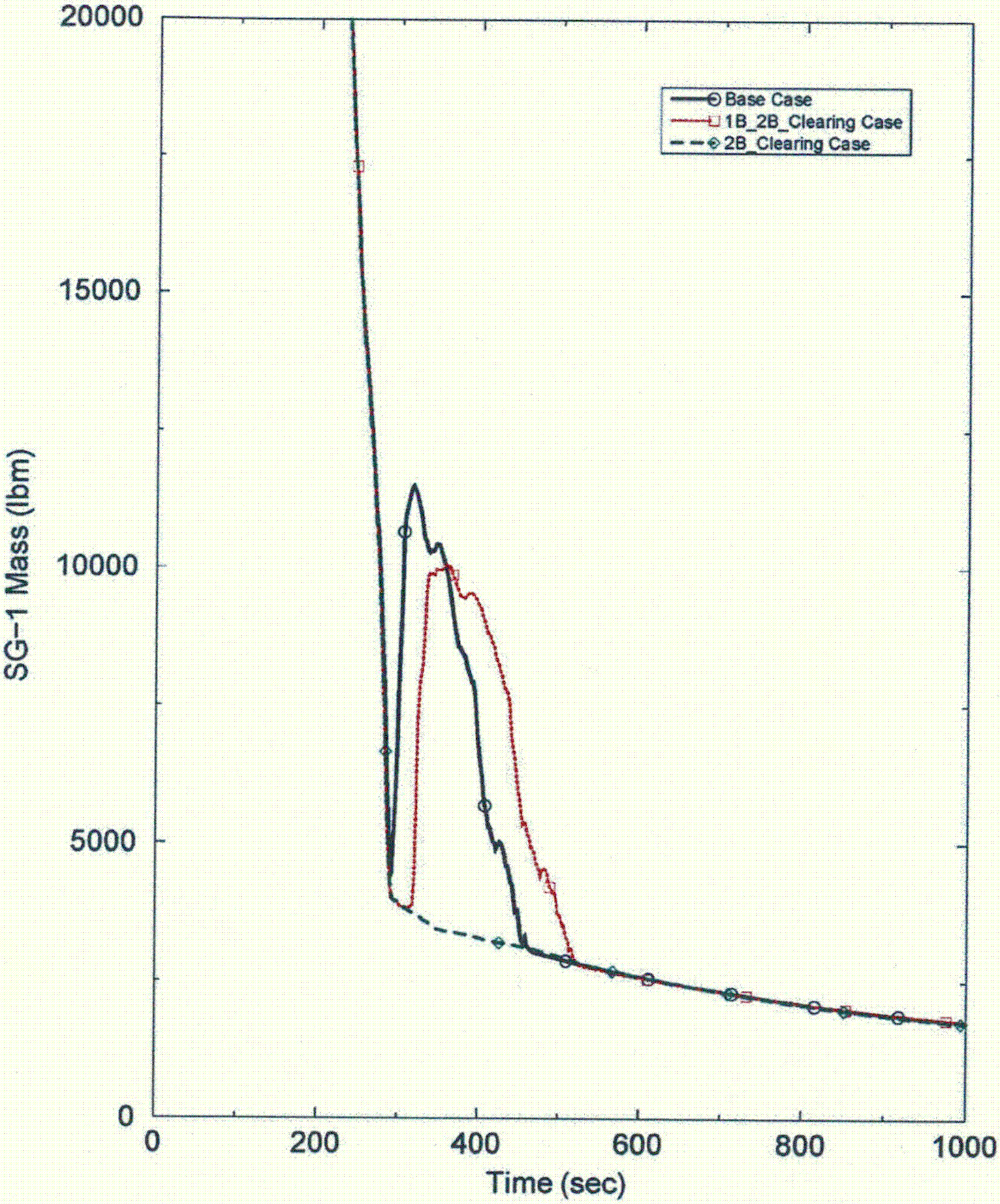


Figure 4-9: Hot Leg #2 Mass Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

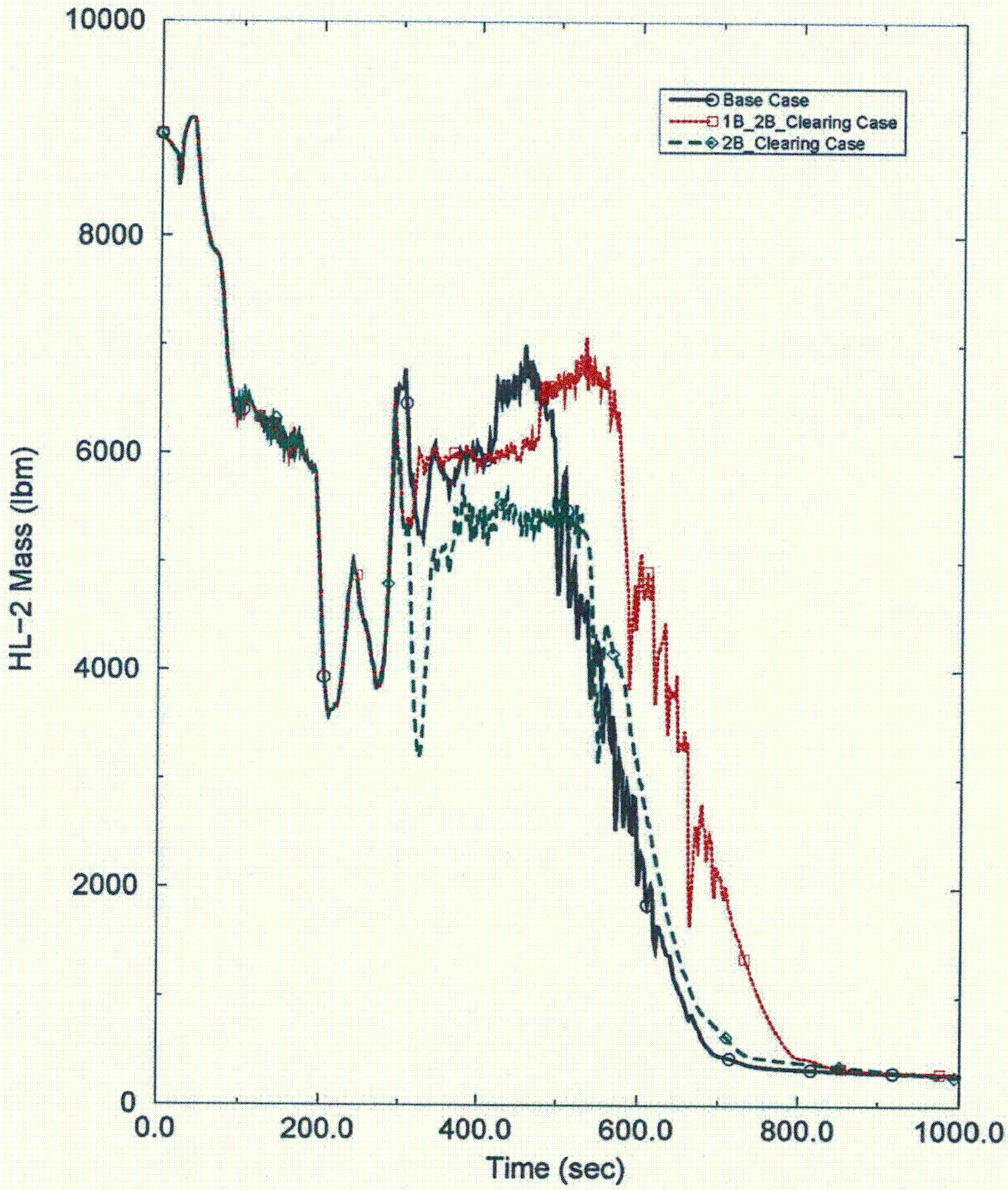


Figure 4-10: SG #2 Mass Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

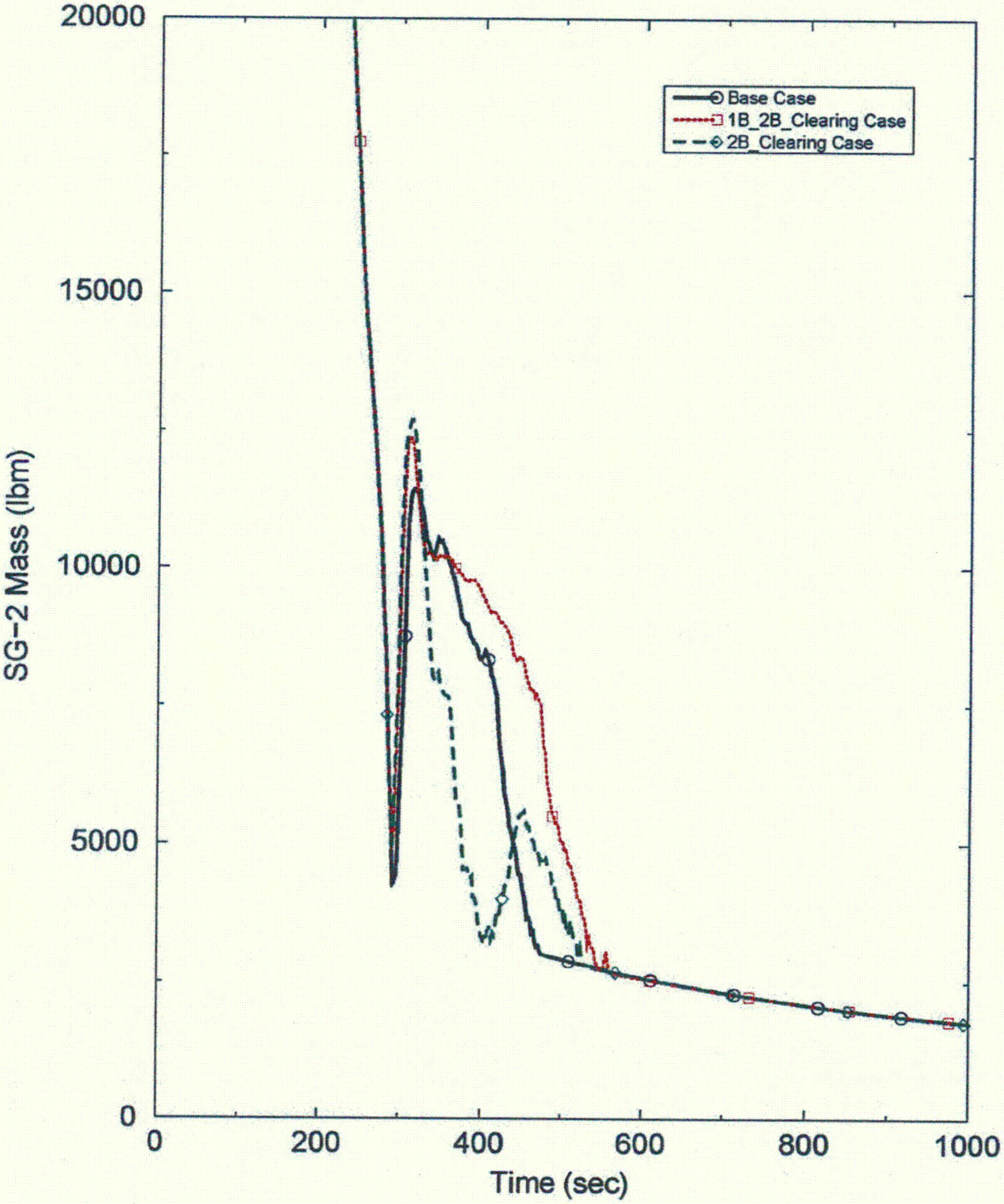


Figure 4-11: Upper Plenum and Upper Head Mass Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

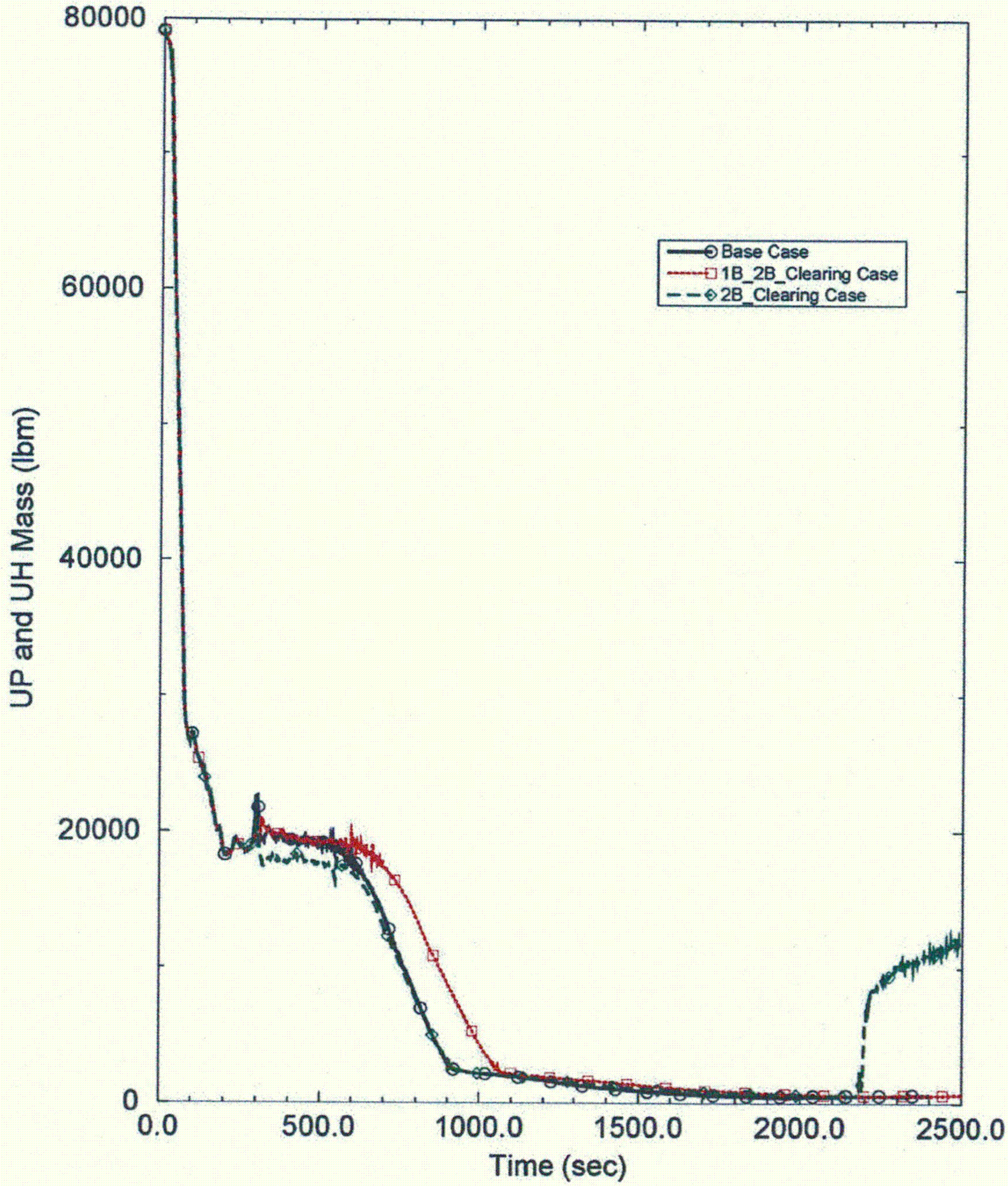


Figure 4-12: Core Collapsed Level Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case

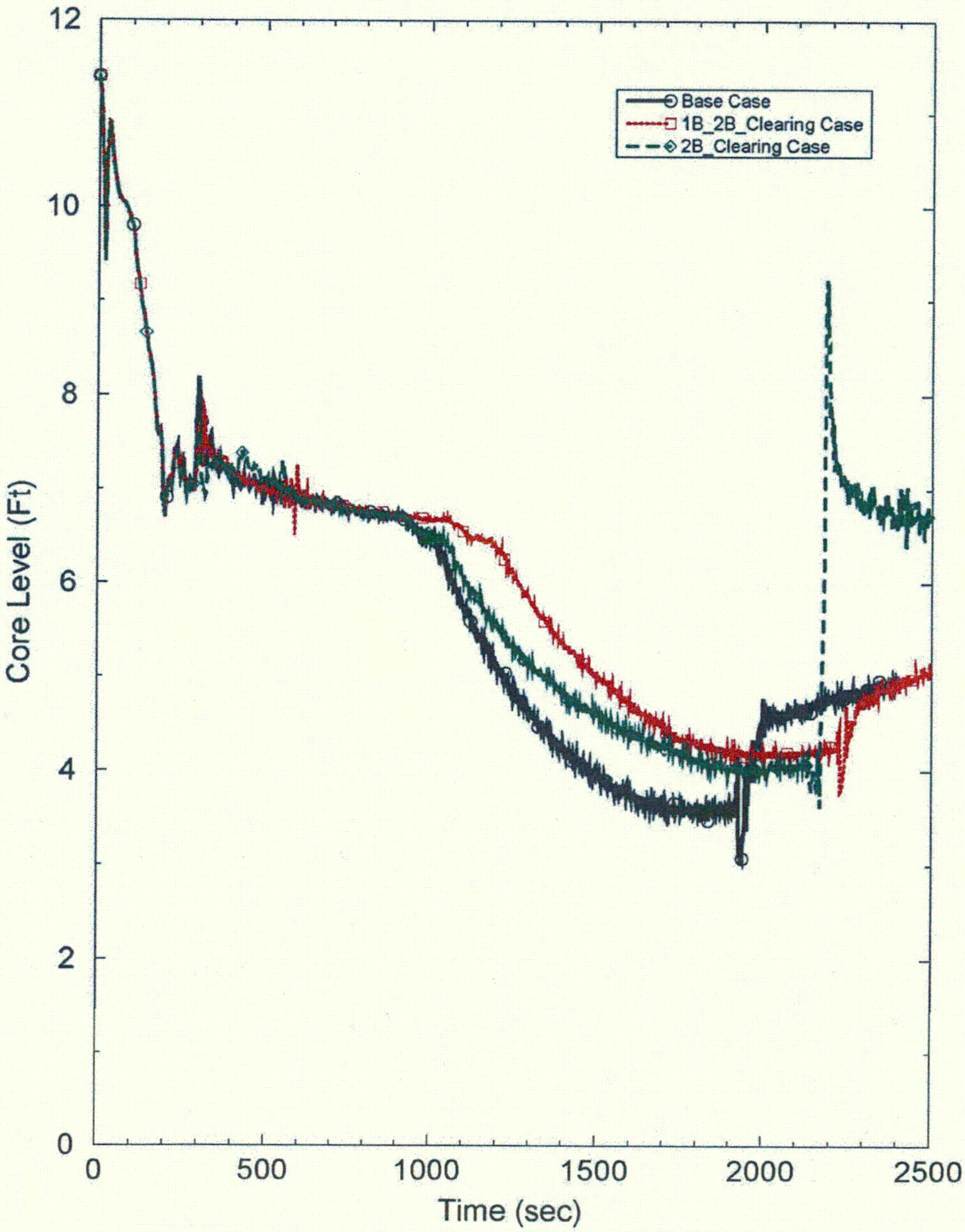
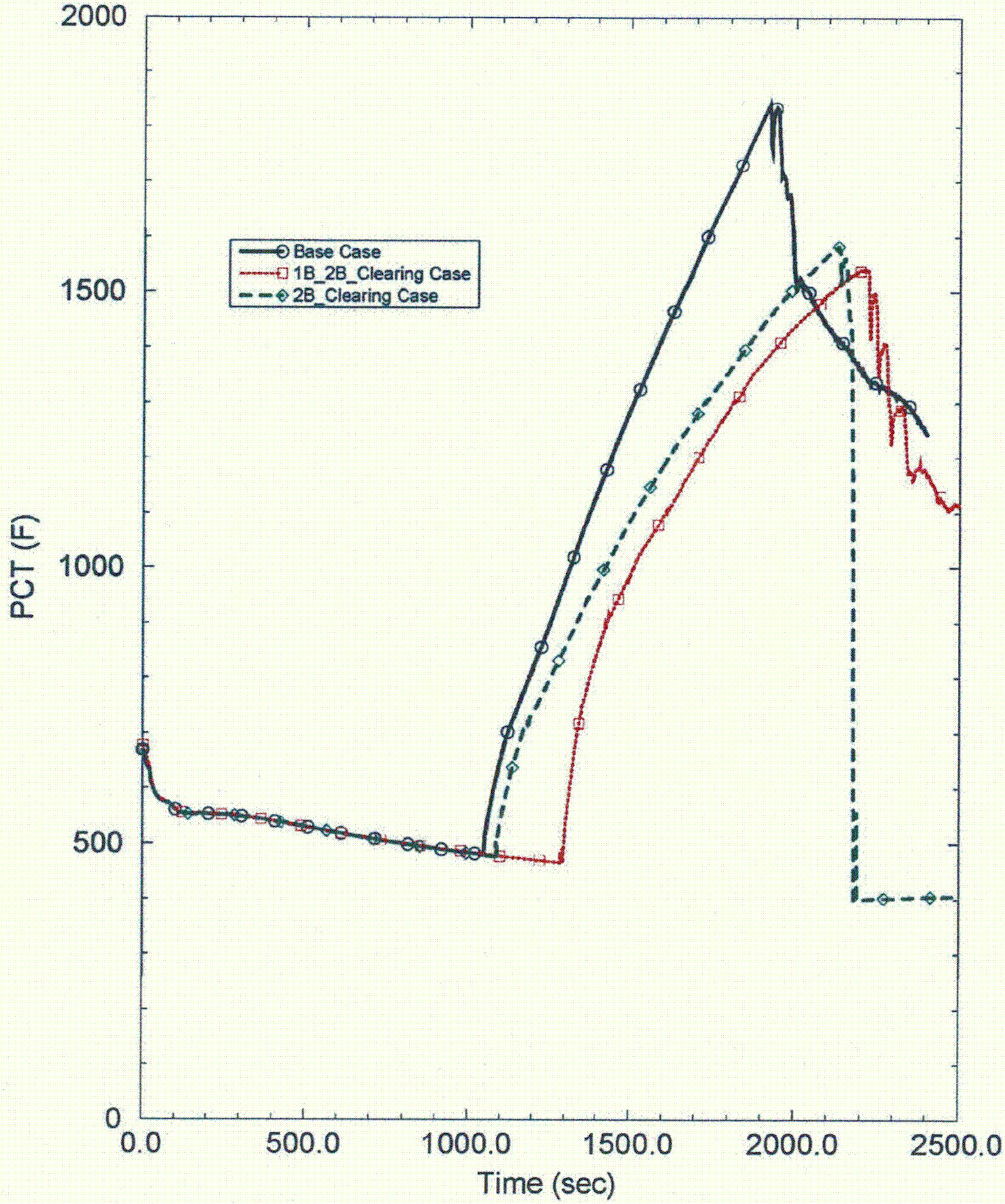


Figure 4-13: PCT Comparison Base Case/1B_2B_Clearing Case/2B_Clearing Case



Response to RAI 4.2:

The downcomer nodalization in EMF-2328 (P)(A) Supplement 1 is unchanged from that in EMF-2328(P)(A). The only change in Supplement 1, that is directly related to the downcomer, is that the [ ] Therefore, there have been no changes to the downcomer modeling that would impact entrainment.

Response to RAI 4.3:

The S-RELAP5 SBLOCA evaluation model (EM) applies [ ] (Section 7.0 of EMF-2328 (P)(A) Supplement 1). Table 4-2 provides loop seal clearing data from several SBLOCA tests conducted at the ROSA, BETHSY, EOS, SEMISCALE, LOBI, and UPTF facilities. These tests cover a range of break sizes and results over which the degree of loop seal clearing progresses from partial to complete clearing of all four loops. Figure 4-14 depicts the information in the table graphically. The image and data are consistent with Figure 7.2 of EMF 2328 (P)(A) Supplement 1, but additionally includes data from the UPTF facility.

All of the tests are scaled representations of a Westinghouse 4-loop NSSS, with the exception of BETHSY, which is a scaled 3-loop Framatome design, EOS, which is also a scaled 3-loop design, and UPTF, which is a full-scale simulation of the primary system of a 4-loop Siemens-KWU PWR. Each is characterized by the scaled break size it represents. Because the loop scale factors vary, the number of loops clearing has been interpreted to provide the effective result. For example, if all loops cleared in ROSA, the result listed is expressed as effectively clearing 4 loops.

If one loop clears in SEMISCALE, the result may be listed as 1 or 3 loops clearing because SEMISCALE simulates a broken loop and combines the three intact loops into one. The ROSA 1.9-inch case clears one of its loops (one ROSA loop equals two PWR loops) at approximately 2000 seconds, but the loop slowly refills. This case has, therefore, been interpreted as the lower bound for single PWR loop clearing. The data is further supplemented by the loop seal clearing conclusions from the ROSA program (Table 4-3, reproduced from Reference 3) which demonstrate repeatability and consistency.

[ ] The expected transition point for a Westinghouse 4-loop plant design from the loop seal biasing approach is superimposed on the figure.

Table 4-2: Loop Seal Clearing Versus Break Size for Various Tests (Ref. 1 and 2)

Table 4-3: ROSA Loop Seal Clearing Results (3)

Figure 4-14 Effective Loop Seal Clearing Versus W4 Plant Equivalent Break Size

References:

1. BAW-10168PA, Revision 3, "RSG LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," December 1996.
2. Liebert, J. Emmerling, R. "UPTF Experiment Flow Phenomena during Full-scale Loop Seal Clearing of a PWR" Nuclear Engineering and Design 179 (1998) 51-64
3. Tasaka, K., et.al, "Loop seal clearing and refilling during a PWR small-break LOCA" Proceedings of the U.S. Nuclear Regulatory Commission Sixteenth Water Reactor Safety Information Meeting, NUREG/CP-0097 Vol. 4