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November 21, 2005
L-05-169

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit Nos. 1 and 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Responses to a Request for Additional Information (RAI dated September 30, 2005) in Support of License Amendment Request Nos. 302 and 173

On October 4, 2004, FirstEnergy Nuclear Operating Company (FENOC) submitted License Amendment Request (LAR) Nos. 302 and 173 by letter L-04-125 (Reference 1). This submittal requested an Extended Power Uprate (EPU) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 and is known as the EPU LAR.

On April 13, 2005, FENOC submitted LAR 320 for BVPS Unit No. 1 by letter L-05-069 (Reference 2). This submittal requested the Technical Specification changes necessary for operation of BVPS Unit No. 1 with the replacement steam generators and is known as the RSG LAR.

By letter dated September 30, 2005, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) pertaining to LAR Nos. 302 and 173. Since approval of the RSG LAR is expected well before approval of the EPU LAR, and a number of the questions in the September 30, 2005 RAI apply to the RSG LAR, an effort has been made to identify the EPU RAI questions that also apply to the RSG LAR.

Enclosure 1 provides the FENOC response to the September 30, 2005 RAI, as modified to remove information considered to be proprietary. Enclosure 2 contains the complete responses to the September 30, 2005 RAI, including the information proprietary to Westinghouse Electric Company LLC. Enclosure 2 also includes an affidavit provided by Westinghouse, the owner of the subject proprietary information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations.

A001

Enclosure 3 contains corrections to RAI responses submitted on July 8, 2005 by FENOC letter L-05-112. None of the identified corrections, nor the responses and supplemental information provided by this transmittal, will impact the proposed Technical Specification changes, the results of the EPU safety analysis or the conclusions drawn, or invalidate the no significant hazards consideration submitted in References 1 or 2.

No new regulatory commitments are contained in this submittal. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Licensing, at 330-315-7243.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 21, 2005.

Sincerely,

Larry R. Freeland

Enclosures:

1. Non-Proprietary Responses to RAI dated September 30, 2005
2. Affidavit and Proprietary Responses to RAI dated September 30, 2005
3. Corrections to FENOC Letter L-05-112

References:

1. FENOC Letter L-04-125, License Amendment Requests 302 and 173, dated October 4, 2004.
2. FENOC Letter L-05-069, License Amendment Request 320, dated April 13, 2005.

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

L-05-169 ENCLOSURE 1

Non-Proprietary Responses to RAIs dated September 30, 2005

REQUEST FOR ADDITIONAL INFORMATION (RAI)

RELATED TO FIRSTENERGY NUCLEAR OPERATING COMPANY (FENOC)

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2)

EXTENDED POWER UPRATE (EPU)

DOCKET NOS. 50-334 AND 50-412

By letter dated October 4, 2004, as supplemented by letters dated February 23, May 26, June 14, July 8, July 28, and September 6, 2005, Agencywide Documents Access and Management System (ADAMS), Accession Nos. ML042920300, ML051160426, ML051160429, ML051160431, ML051530376, ML051670270, ML051940575, ML052140310 and ML052550373, FENOC (the licensee) proposed changes to BVPS-1 and 2 operating licenses to increase the maximum authorized power level from 2689 megawatts thermal (MWt) to 2900 MWt rated thermal power (RTP) or approximately 8%. The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's application against the guidelines in the EPU review standard, RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003, and determined that it will need the additional information identified below to complete its review. As the analyses used to support EPU also may support the replacement steam generator (RSG) license amendment request (LAR) for BVPS-1, dated April 13, 2005, as supplemented August 26, 2005, these questions also apply to the RSG LAR. These second round RAI questions are numbered the same as those in the NRC staff's May 5, 2005, RAI for continuity.

Question E.2 (Applicable to RSG & EPU)

Comparison of the high-head safety injection (HHSI) pump head flow curves for pre-EPU and EPU conditions for the SBLOCA analyses show the EPU head flow curve has increased by about 10%. Please explain how this was accomplished. Also, the flow into the two intact loops should be based on the two minimum-loop injection measurements from the surveillance testing. Please confirm this approach in arriving at the head flow curve for the SBLOCA analyses.

Response:

The rotating assemblies for each High Head Safety Injection (HHSI) pump, has or will be replaced prior to implementation of Extended Power Uprate (EPU) for both BVPS-1 and -2. The new rotating assemblies have a maximum flow limit which is 25 gpm higher than those being replaced. This change allows the HHSI system throttling valves to be set at a higher runout flow which has the effect of lowering the overall system resistance. This has the effect of increasing the HHSI flow at runout conditions and all Reactor Coolant System (RCS) pressure conditions. Additionally, improved instrumentation has or will be installed at both BVPS-1 and -2. This reduces the flow measurement uncertainty, which is accounted for in the HHSI calculations which develop the HHSI system throttle valve settings. A change is also proposed in the EPU and Replacement Steam Generator (RSG) License Amendment Requests to modify the Reactor Coolant Pump (RCP) seal injection minimum pump discharge pressure. This change has the effect of increasing the analytical resistance used to represent the seal injection path. Since RCP seal injection is not isolated during a safety injection alignment, it represents a loss to the system, as it is not credited as injection flow. A higher resistance for seal injection has the effect of reducing this loss and allowing a higher injection flow for the same pump performance. The combined effects of these changes result in the increased HHSI flow used for EPU analyses.

The HHSI throttling procedures specify a minimum and maximum total system flow as well as injection path balance criteria. The balance criteria, currently requires that the difference between the highest and lowest injection path flow at near runout conditions not exceed 10 gpm. In developing the EPU HHSI curve, it is conservatively assumed that the injection paths are at the balance limit with the two lowest injection paths 10 gpm lower than the highest injection path flow. The two injection paths with the lowest flow are assumed to inject, while the injection path with the highest flow is assumed to spill.

Question E.3 (Applicable to RSG & EPU)

The NRC staff's RELAP5 calculations show that the peak clad temperature (PCT) for a severed injection line is about 1750 °F. Although the accumulators do inject into the core, the increased core uncovery with the reduced injection capacity produces a rapid heat-up and high clad temperatures before termination of the event by the accumulators. Using BVPS-1 and 2-specific information, p[ease show that the NOTRUMP code does not produce temperatures for this condition that are higher than the worst break in the bottom of the cold leg.

Response:

As part of the new Small Break LOCA (SBLOCA) analysis (Reference E.3-1) that includes a revised break spectrum, a six inch break size is also provided which represents a sheared off Emergency Core Cooling System (ECCS) line with flow spilling to containment back-pressure. For this case, no ECCS flow is assumed in the faulted loop and delivered flow to the intact loops is penalized because of the back-pressure assumptions. The results show that this case is non-limiting with respect to the other cases analyzed. This should encompass the scenario as requested.

Reference:

E.3-1. FENOC Letter L-05-168, "Supplement to License Amendment Request Nos. 320 (Unit No. 1 TAC No. MC6725) and 302/173 (Unit No. 1 TAC No. 4645/Unit No. 2 TAC No. MC4646)" dated October 28, 2005).

Question E.5 (Part 1) (Applicable to RSG & EPU)

The response to this question was not acceptable. The NRC staff's position is that the worst small break should be identified in all SBLOCA evaluation model (EM) submittals. The analysis approach of only investigating integer break sizes does not meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.46, paragraph (a)(1)(I) that states "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." The worst small break is typically that break (or one where the accumulators inject for a very short period of time) where the reactor coolant system (RCS) depressurizes to a minimum pressure that remains just above the accumulator action pressure. Please provide the results of the BVPS-1 and 2-specific analysis related to the limiting break.

Response:

To address this concern, as well as others presented in the Request for Additional Information (RAI) process, a new SBLOCA analysis was performed. In this re-analysis, a tighter break spectrum was performed. Instead of the typical 1-inch equivalent size variations, [ ]a,c in break size were investigated in the size range from [ ]a,c where SBLOCA Peak Clad Temperature (PCTs) for the Beaver Valley Power Station (BVPS) plant type occurs. This break size interval showed a good characteristic behavior and that no extreme outliers likely existed within the range considered. Given this and the overall conservatism that exists within the model, the [ ]a,c is considered adequate and the concern addressed.

As discussed in the meeting with the Nuclear Regulatory Commission (NRC) on October 13, 2005, the worst break observed for the BVPS SBLOCA transients is not the scenario described by the staff above. In the break size range where the results are limiting (2 to 3 inches equivalent size), the RCS always depressurizes to where the accumulators inject. It is just a question of whether the ECCS has been able to introduce enough mass into the RCS such that the clad temperature heat-up is terminated prior to this point. Westinghouse investigated several break sizes in the break size range where the transient turned around solely on ECCS flow. The results are as follows:

[ ]a,c

Note that the transient oxidation values reflect pre-burst conditions. When burst occurs, two-sided metal-water reaction is introduced and these values increase dramatically. Under burst conditions, the 2.5 and 2.75 inch equivalent diameter breaks are clearly limiting with respect to cladding oxidation and PCT, respectively.

Figure E.5-1 through Figure E.5-6 present comparisons of the analysis results for the break spectrum range as described previously. It is noted that during the time frame where the core mixture level decreases are interrupted (Figure E.5-3), the corresponding downcomer mixture levels (Figure E.5-4) are well above that of the core/upper plenum region. Therefore, with adequate core steam venting capabilities (i.e. post loop seal clearing), the core and downcomer will attempt to come into manometric balance. Once the downcomer mixture level has been sufficiently depleted, the core mixture level once again begins to decrease. Some minimal liquid refluxing is also occurring during this time period which will assist in suspending the core mixture level depletion.

It is observed that as the break size increases, the RCS inventory depletion increases (Figure E.5-6) as does the predicted depth of core uncovery (Figure E.5-3). With increasing break sizes, the RCS depressurization (Figure E.5-5) also increases with increasing break size; therefore, even though the larger breaks uncover deeper, the accumulators inject earlier (Figure E.5-2) thereby terminating the cladding heat-up.

Question E.5 (Part 2) (Applicable to RSG & EPU)

Please explain and justify how the accumulator maximum water temperature is maintained at 105°F temperature assumed in the analysis.

Response:

Technical Specification 3.6.1.5 requires that the containment average building temperature stay below a maximum temperature of 105 °F. The accumulators are located in the lower level of the containment building. Past reviews of containment temperature surveillance data have indicated that the lower levels of containment are consistently below the overall average containment temperature. Since there are no other sources of heat input into the accumulators other than the surrounding ambient containment air, the accumulators would not exceed the containment maximum Technical Specification temperature limit of 105 °F.

Question E.5 (Part 3) (Applicable to RSG & EPU)

Also, were failures of instrument tube penetrations in the lower reactor vessel head evaluated? If so, what was learned from the evaluations?

Response:

No plant specific analyses were performed for the two BVPS units to support the EPU or RGS submittals with regards to failures of bottom mounted instrumentation (BMI) tubes. However, in response to the Davis Besse and South Texas Unit 1 events, a comprehensive Westinghouse Owners Group (WOG) program for both traditional Westinghouse and Combustion Engineering System 80 reactor vessels was developed several years ago to assess the impacts of a postulated leak or failure of one or more BMI nozzles. The WOG program included the following tasks.

- Historical information review to determine the extent to which BMI breaks have been analyzed and to determine the effort required to address the potential consequence of a BMI failure.
- Small Break LOCA analyses to evaluate the potential effect of various failures of BMI tubes. These results are then utilized to support a probabilistic risk assessment of BMI failures.
- A materials assessment which evaluates the potential for failure based on phenomenological considerations. This includes Failure Modes and Effects Analysis (FMEA)

- Evaluation of the effectiveness of the Emergency Response Guidelines (ERGs) in dealing with this postulated scenarios and provisions for recommending modifications to the guidance.

During the execution of this program, various organizations discussed the benefits of providing a coordinated fleet-wide response to BMI related issues. As such, a joint effort between the WOG, B&W Owners Group (BWOG) and Materials Reliability Project (MRP) was developed to provide this response. The effort culminated in the development of internal documentation which supports the various conclusions reached in regards to these issues. A meeting to present the WOG and BWOG results to the NRC was held on September 30, 2005. A summary of the observed LOCA response is provided below:

- Different plant groups demonstrate similar responses to the BMI small LOCA event. Evaluated thermal hydraulic analysis cases show that a Bottom Mounted Nozzle (BMN) break of approximately 1.0 inch equivalent diameter can be withstood under timely operator action (45 minutes) to depressurize without core uncovery.

a,c

Figure E.5-1 BVPS-1, Core Exit Vapor Temperature Comparison

a,c

Figure E.5-2 BVPS-1, Integrated Broken Loop Accumulator Injection Flow

a,c

Figure E.5-3 BVPS-1, Core Mixture Level Comparison

a,c

Figure E.5-4 BVPS-1, Downcomer Mixture Level Comparison

a,c

Figure E.5-5 BVPS-1, RCS Pressure Comparison

a,c

Figure E.5-6 BVPS-1, RCS Inventory Comparison

Question E.6 (Applicable to RSG & EPU)

As shown in Figure (Fig.) E.6-7, removing the erratic behavior in the two-phase level between 1400 and 2200 sec shows elimination of the cooling between 1400 and 1800 sec. However, since the minimum level is not turned around until about 3200 sec, what caused the cooling between 2200 and 2400 sec? It appears that the PCT should be increased by the difference in temperatures at around 2200 sec. Please explain.

Response:

The figure in question is shown in Figure E.6-1. The following information is provided to supplement the response.

The core level minimum occurs at approximately 2800 seconds (Figure E.6-2) as opposed to the 3200 seconds stated.

Since the core cooling boundary conditions, as passed from the NOTRUMP Thermal & Hydraulic (T&H) code, are identical beyond the 2200 second time period, the code will naturally attempt to re-converge on cladding temperature given that no cladding rupture was predicted to occur. In addition, the cladding temperatures which occur during this time frame are below the sensitive metal water reaction range (i.e. > 1700°F). As a result, the cladding temperatures will re-converge given the same T&H boundary. Since the cladding temperatures at ~2200 seconds are different between the reference and smoothed mixture level case, there will be a time period over which they will re-converge once the same boundary conditions are re-established (Figure E.6-3). Figure E.6-4 shows that the cladding heat flux for the smoothed mixture level case is higher than for the reference case. This is a direct result of the elevated cladding temperature at the time where identical boundary conditions were re-established. Subsequently, more heat is released from the cladding to the steam over this time period which slows the cladding temperature increase while the two cases re-converge. This re-convergence period takes approximately 300 seconds to occur and is essentially completed at approximately 2500 seconds. Had cladding rupture or PCTs above 1700°F occurred during the period of interest, the cladding temperatures may not have re-converged due to blockage and metal-water reaction rate affects.

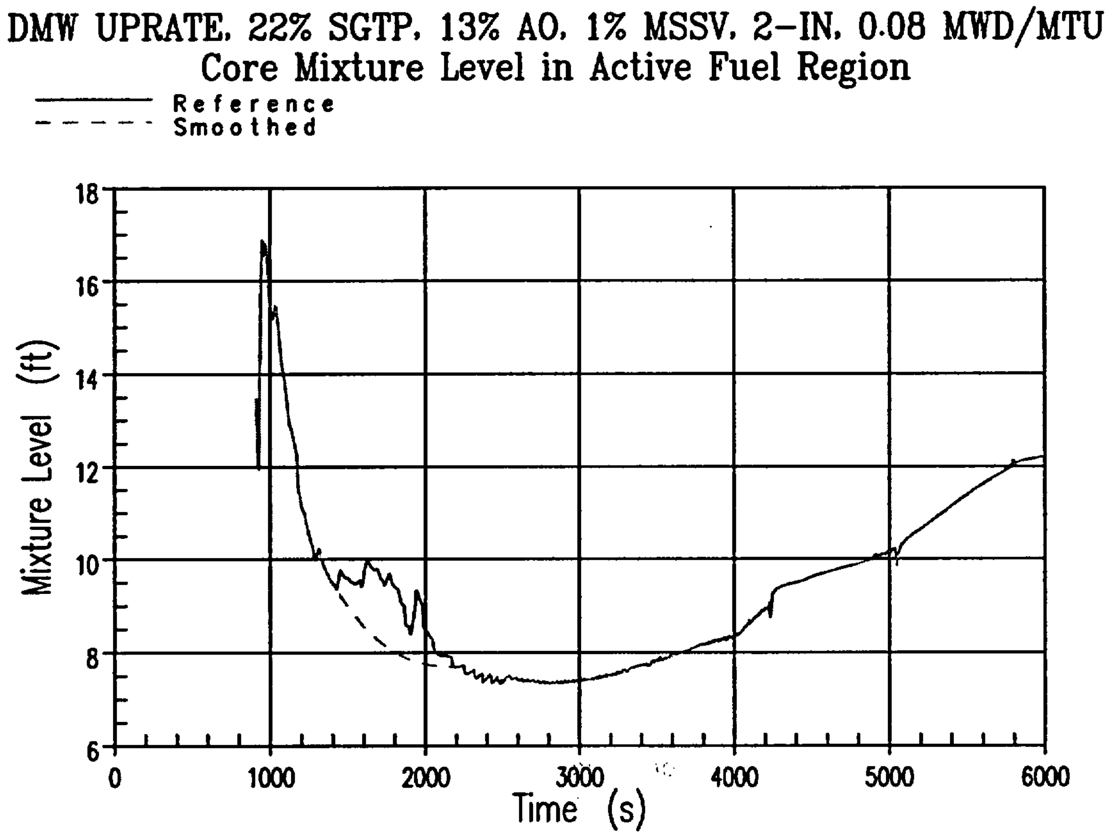

Figure E.6-1 BVPS-2 2-Inch PCT Response

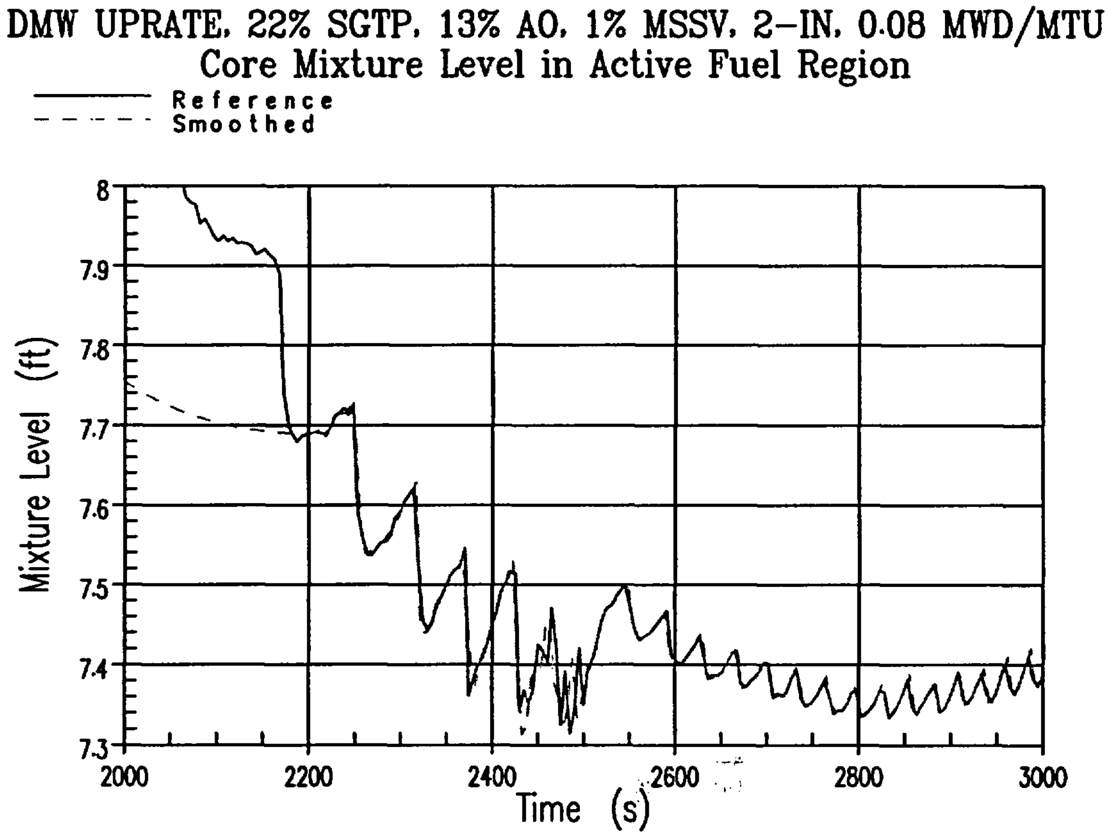

Figure E.6-2 **BVPS-2 2-Inch Core Mixture Level Response (2200-3000 sec.)**

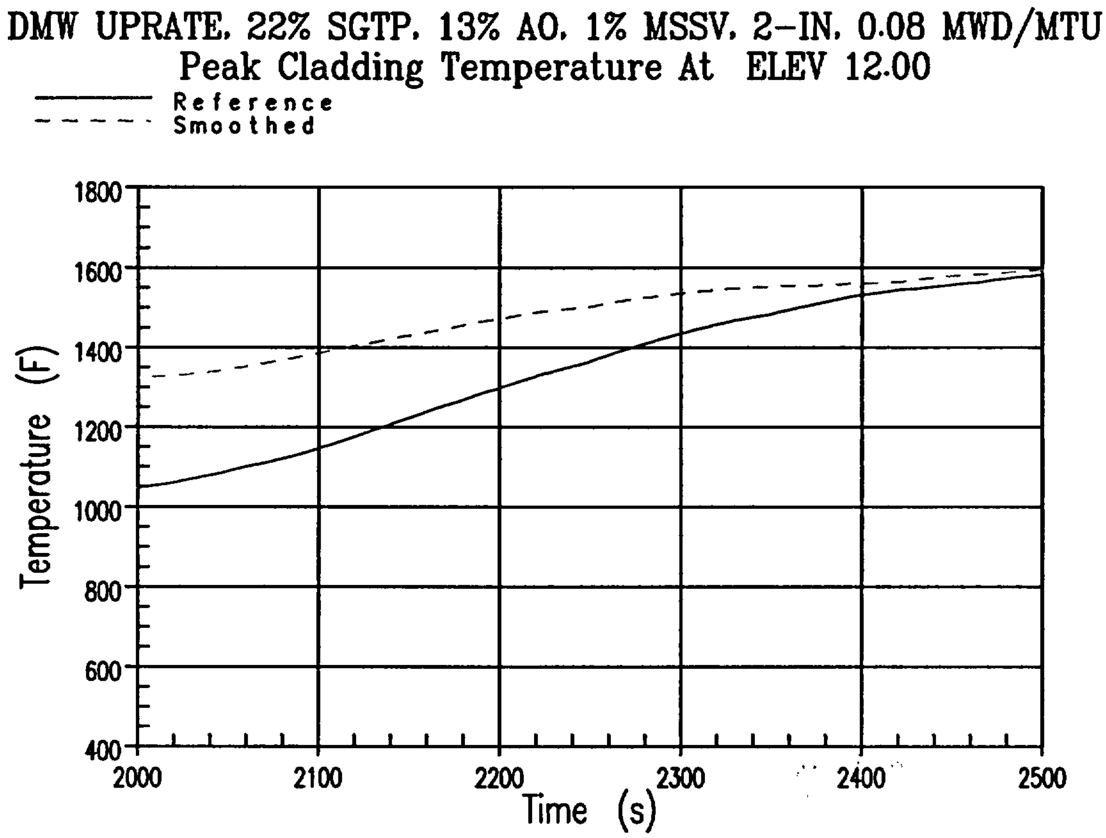

Figure E.6-3 BVPS-2 2-Inch PCT Response (2200-2500 sec.)

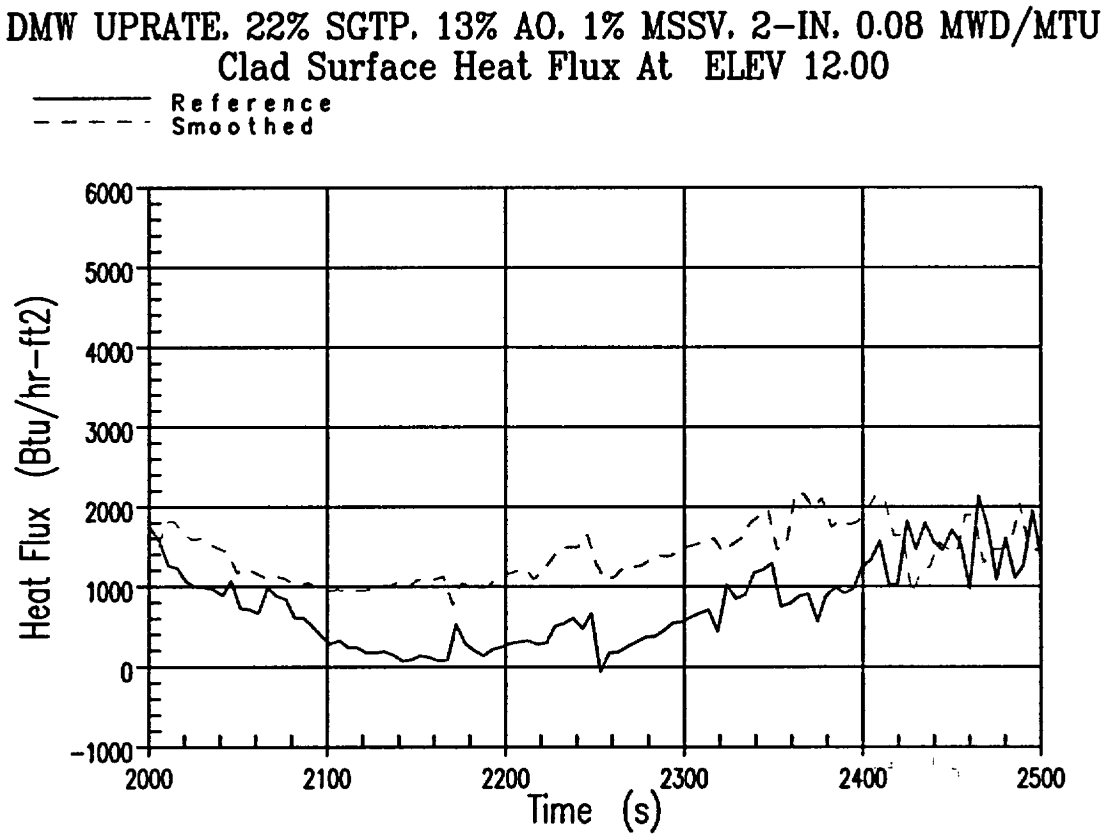

Figure E.6-4 BVPS-2 2-Inch Clad Surface Heat Flux (2200-2500 sec.)

Question E.7 (Applicable to RSG & EPU)

The modification to allow more than one loop seal to clear represents a change to the approved SBLOCA EM using the NOTRUMP code. From page 5-45 of WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP CODE," August 1985 clearly states only the broken loop seal is allowed to clear. It reads:

"To ensure the most severe core mixture level and cladding heatup transient response during a small break LOCA in compliance with the spirit of Appendix K to 10 CFR 50.46, a modification was applied to the NOTRUMP small break LOCA PWR model to assure conservative loop seal behaviors. This modification permits only one loop seal (the broken loop) to vent significant amounts of steam."

The EM further states: "Reiterating, break sizes larger than the threshold break size will realistically vent steam through more than one loop seal and in doing so will result in minimal core uncovery. The modification to assure conservative behavior is also applied to those breaks to ensure a continuum of response in terms of peak clad temperature when only the broken loop is artificially forced to vent steam."

The NRC staff requests that the analysis be repeated adhering to the approach approved in the topical report by allowing only the broken loop seal to clear.

Or alternatively, should Westinghouse choose to change the model to allow more than one loop seal to clear, then a full description of the new model, sensitivity studies, and appropriate benchmarking would be needed to allow the NRC staff to review this change. The additional benchmarking would include analyses of separate effects tests and integral tests experiments to validate the new model (i.e. SEMISCALE tests series S-LH, S-UT, etc.). Test S-07-10D shows the effect of loop seal clearing on the core two-phase level and the subsequent clad heat-up. Please also provide the comparisons of the new model with this test and show the effect of the new loop seal model on the long-term level in the core and PCT.

Since the NOTRUMP model combines both intact loops, a new nodalization model would need to be presented and justified. Please note that the RELAP5/MOD3 analysis of the BVPS-1 and 2 break spectrum shows only the broken loop clears for break sizes less than 6 inches in diameter.

The NRC staff also notes that the review schedule for the BVPS-1 and 2 EPU would be extended should Westinghouse choose to pursue modifications to the SBLOCA NOTRUMP EM loop seal model.

See also the additional technical concerns expressed for E.19.

Response:

To address this concern, as well as others presented in the RAI process, a new SBLOCA analysis was performed (Reference 1). In this analysis, only the faulted loop seal was allowed to clear for break sizes of less than **[ ]** ^{a,c}. This is consistent with Westinghouse practice since NOTRUMP was licensed.

Reference:

1. FENOC Letter L-05-168, "Supplement to License Amendment Request Nos. 320 (Unit No. 1 TAC No. MC6725) and 302/173 (Unit No. 1 TAC No. 4645/Unit No. 2 TAC No. MC4646)" dated October 28, 2005).

Question E.9 (Applicable to RSG & EPU)

For the 3-inch break, the top portion of the core has not been recovered and in fact the two-phase level is decreasing at 5000 sec. in Fig. E.8-5. If operator action is required to cool the core for small breaks during the short term, please identify the actions and show that the core can be recovered so that the core is quenched and oxidation is no longer relevant. Identify what actions are required and show the effect on PCT and oxidation. Identify the timing for operator actions and demonstrate that the latest time for action is reflected in the EOPs. What is the latest time allowed for the recommended operator actions? Also, since the two-phase level is below the top of the core at 5000 sec for the 3-inch break, RHR cannot be operated since the loops contain steam (re-establishment of single phase natural circulation by refilling the RCS with ECC liquid is expected to take a long time). Since RCS pressure is already low for the 3- and 4-inch breaks, depressurization with SGs and PORVs may not be timely nor effective. Please show the analyses that are required to show the core is quenched and oxidation is no longer increasing.

(E.9a. Same as E. 9 above for the 3-inch break.)

Response:

As far as limiting clad temperature and local oxidation is concerned, there are no specific actions required to keep the core cool or limit clad temperature in the near term. The NOTRUMP evaluation model (EM) makes no assumptions in this regard. However, in the longer term of stable and sustained quench, specific actions are taken by the plant operations staff to reduce the reactor coolant system (RCS) pressure. These actions reduce break flow while at the same time increasing make-up flow. While still not specifically credited here, this is considered relevant since the thermal-hydraulic response of the RCS is not well represented in the longer term because of restrictions placed on the EM out of convenience. Examples of these are the lower pressure limit on the Condensation of Safety Injection (COSI) condensation model and the lack of an enhanced condensation model on the make-up flow introduced by the accumulators. The behavior noted in this time frame (beyond 1 hour) is in a quasi-steady state. That is the RCS has depressurized to a point where break flow and make-up flow will approach an equilibrium value with respect to one another. However, with the accumulators in communication with the RCS during this time, the pressure is slow to change on it's own accord. If the code were executed long enough, it would eventually show that the core would completely re-quench, however, at some point the intent of 50.46 is met and long term cooling and cold shutdown conditions become the focus. None of the break size cases executed indicate any response to the contrary.

The significant operator action steps would be to cool the RCS down by dumping steam either from the steam generators (SG's) through the steam dump system or the atmospheric relief valves. This would be done at a rate not to exceed 100°F/hr to prevent pressurized thermal shock concerns. There is no upper time limit as to when an operator action must occur in this regard. As stated above, there is nothing in the simulation results or the plant conditions at later times to indicate that core temperatures would deviate from the near saturation conditions that exists in the longer term (i.e. post PCT and oxidation cessation) until full level recovery is achieved. [

] a,c

It is true that for breaks of this size range (greater than ~ 1 inch), normal RHR operation cannot be used. The volumetric flow out of the break is too large to permit refilling of the RCS to the point where RCS subcooling could be re-established and allow this to occur. Long term cooling would be in the form of emergency core cooling system (ECCS) sump recirculation. The fact that vapor exists in the RCS has little impact on this. Depressurization using the steam generators is effective because of the significant heat transfer area that is available. Figures E.9-1 through E.9-9 show cooldown behavior for three different break sizes i.e. 2, 3 and 4 inch equivalent diameter. In all of these simulations, a plant cooldown was started at approximately 1 hour into the transient. This time frame includes 4 minutes of EOP transition briefings by the operating crew (i.e. E.0 to E.1 and E.1 to ES-1.2). This is well in excess of what actually occurs. Plant simulator runs executed at BVPS showed that the maximum time elapsed before plant cooldown was initiated was less than 1 hour. Included in this time period were the operator briefings conducted prior to transitioning between EOPs. The attached results are based on BVPS-2, however, they are indicative of BVPS-1 (and typical of all SBLOCA's under the same circumstances) as well since there are no differences between the units that would significantly impact these results. As can be seen, in all of these cases, depressurization of the RCS using the SG's is effective. Note that low pressure safety injection (LPSI) system flows were not included in the runs that characterized the cooldown process. If included, these would provide additional flow into the vessel as part of the long term cooling phase.

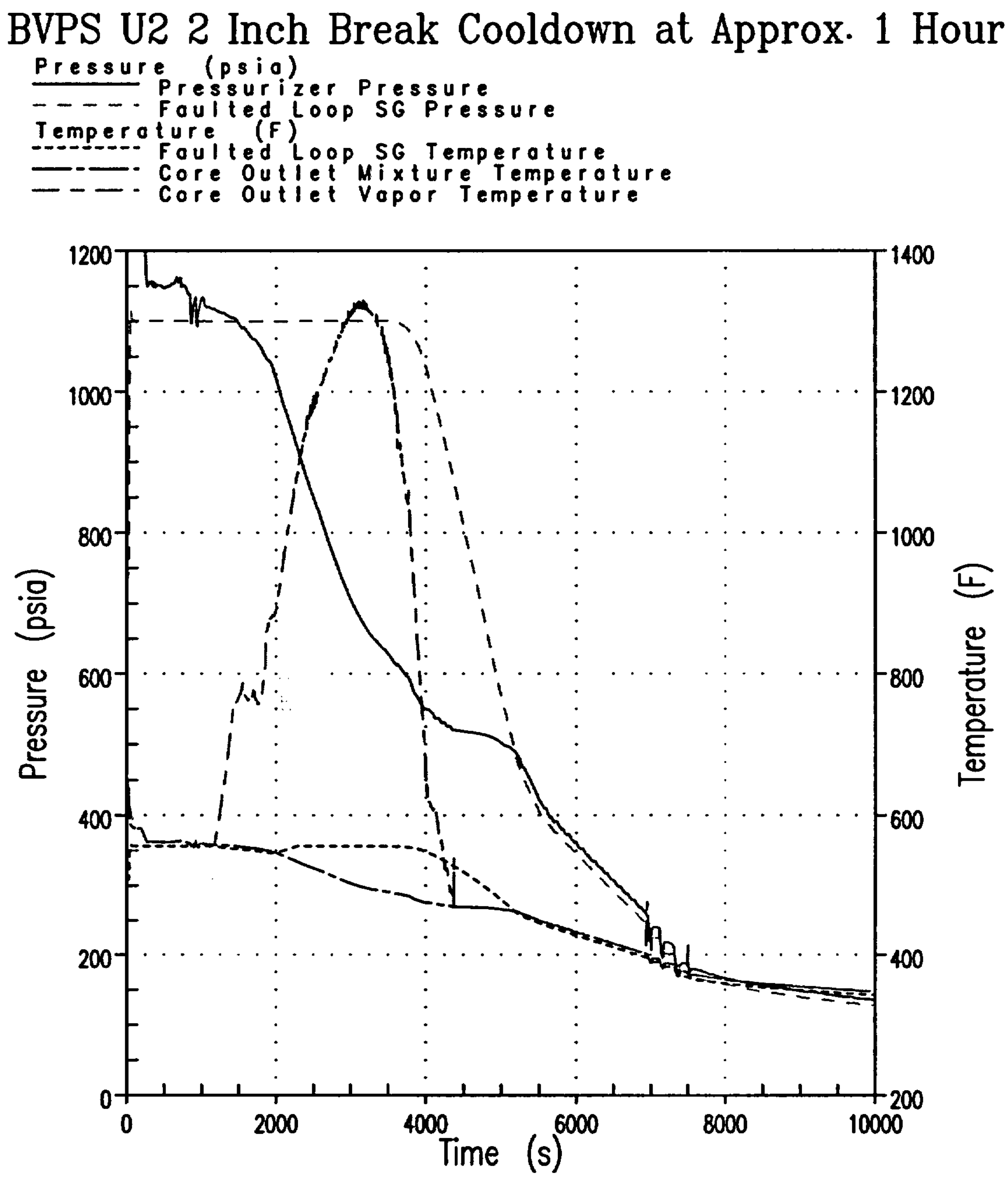
BVPS U2 2 Inch Break Cooldown at Approx. 1 Hour
Pressure (psia)
Pressurizer Pressure
Faulted Loop SG Pressure
Temperature (F)
Faulted Loop SG Temperature
Core Outlet Mixture Temperature
Core Outlet Vapor Temperature
Pressure (psia)
Temperature (F)
Time (s)
0
200
400
600
800
1000
1200
1400
2000
4000
6000
8000
10000

Figure E.9-1

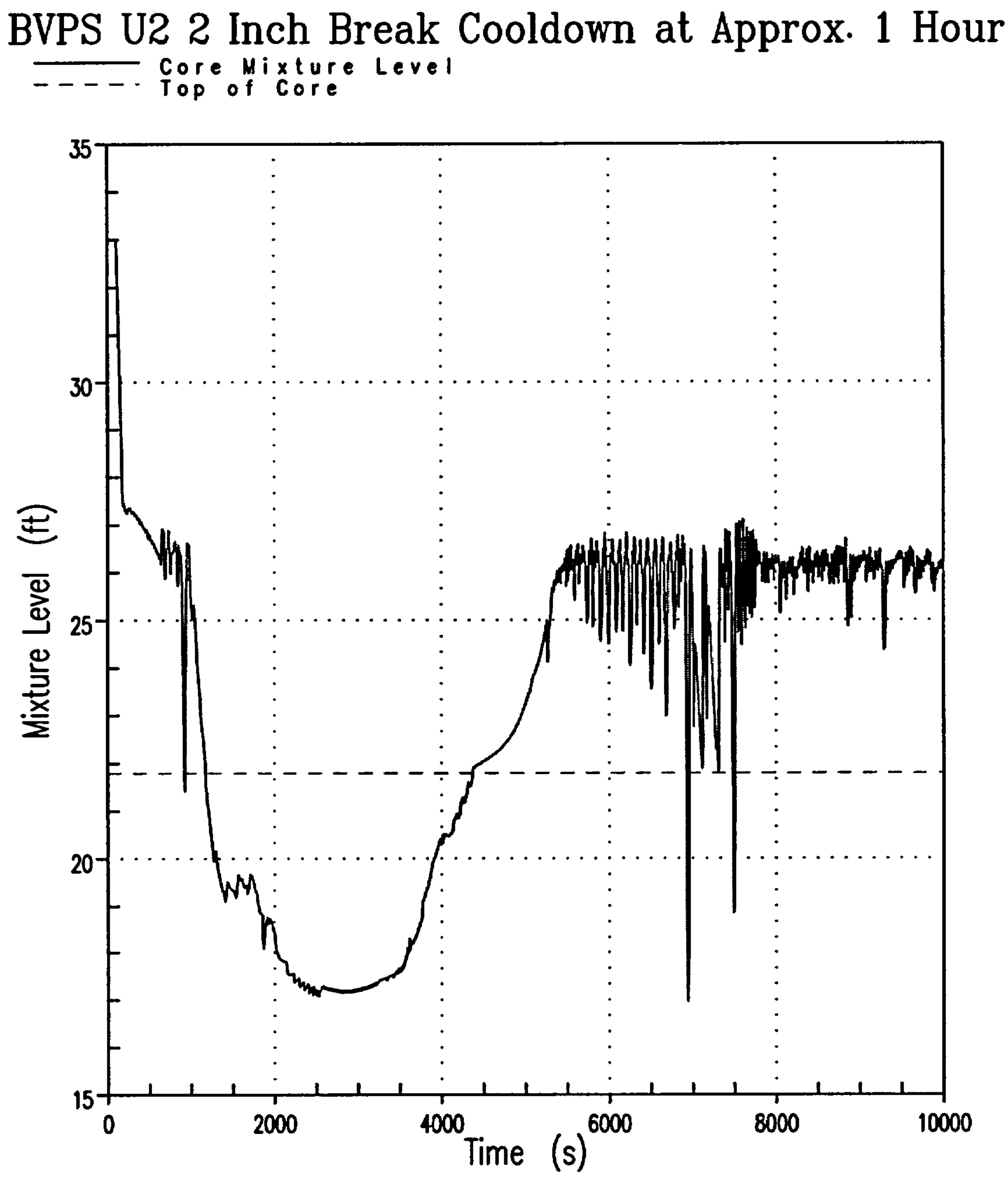
BVPS U2 2 Inch Break Cooldown at Approx. 1 Hour
Core Mixture Level
Top of Core
Mixture Level (ft)
Time (s)
35
30
25
20
15
0
2000
4000
6000
8000
10000

Figure E.9-2

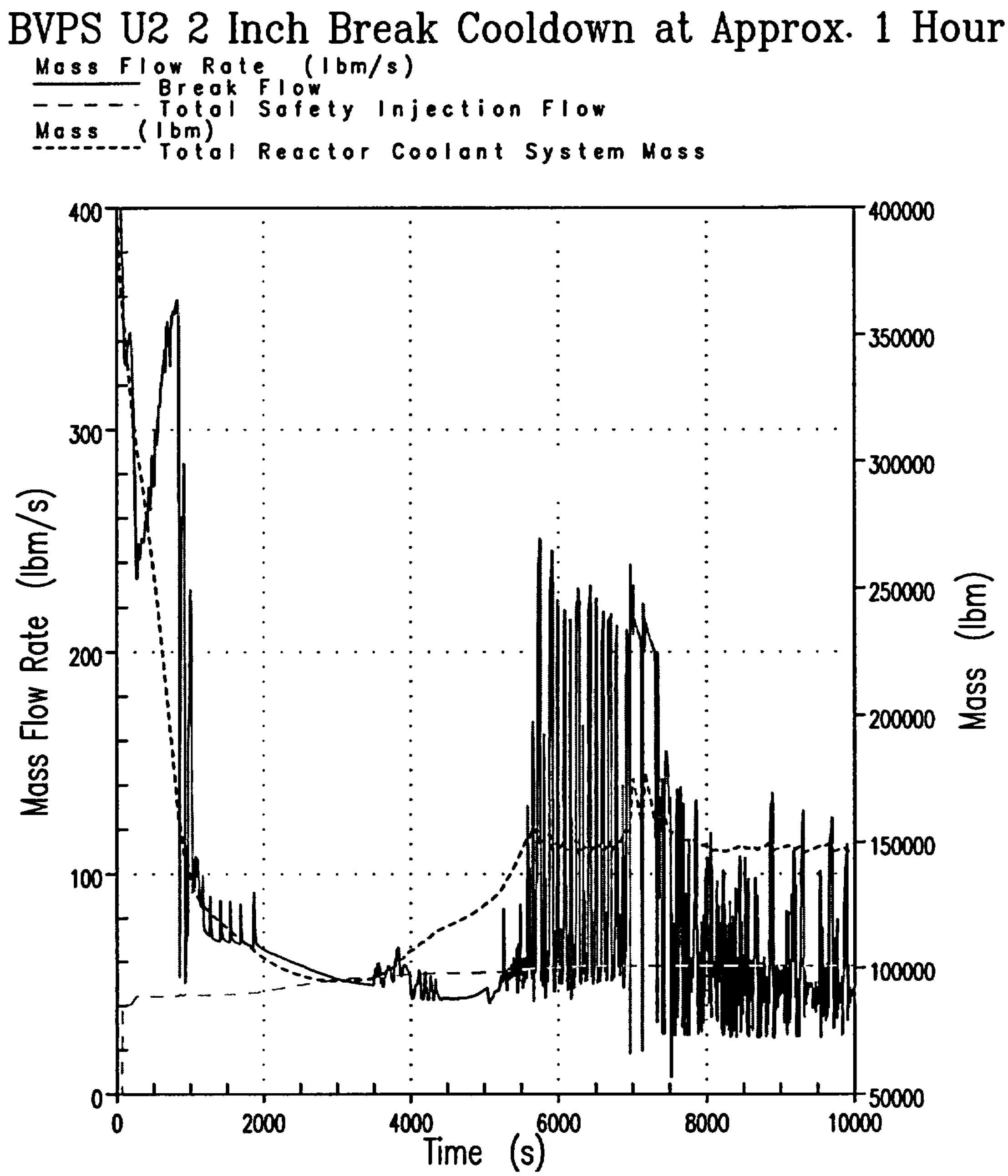
BVPS U2 2 Inch Break Cooldown at Approx. 1 Hour
Mass Flow Rate (lbm/s)
Break Flow
Total Safety Injection Flow
Mass (lbm)
Total Reactor Coolant System Mass
400
300
200
100
0
400000
350000
300000
250000
200000
150000
100000
50000
0
2000
4000
6000
8000
10000
Time (s)
Mass Flow Rate (lbm/s)
Mass (lbm)

Figure E.9-3

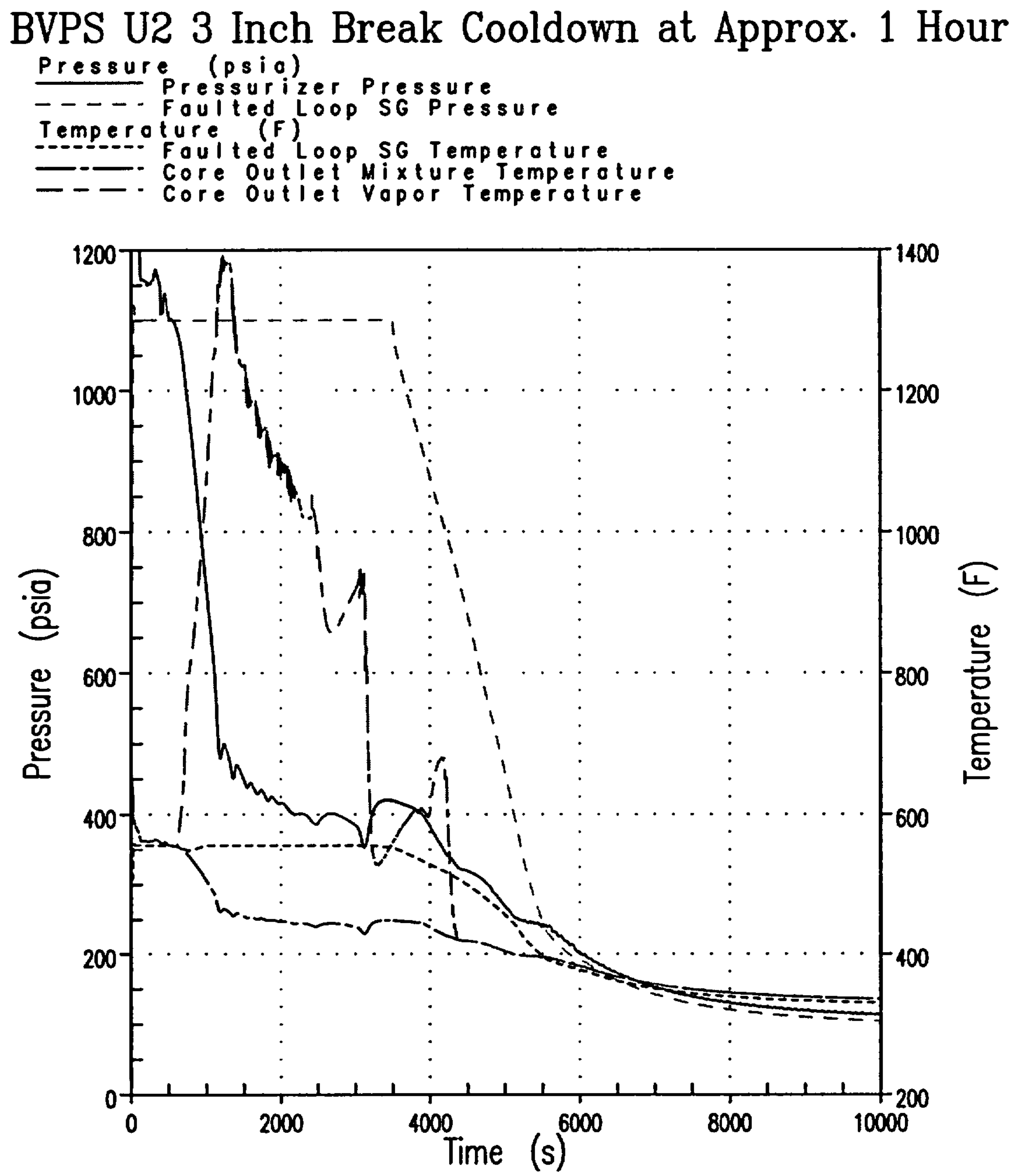
BVPS U2 3 Inch Break Cooldown at Approx. 1 Hour
Pressure (psia)
Pressurizer Pressure
Faulted Loop SG Pressure
Temperature (F)
Faulted Loop SG Temperature
Core Outlet Mixture Temperature
Core Outlet Vapor Temperature
Pressure (psia)
Temperature (F)
Time (s)
0
200
400
600
800
1000
1200
1400
2000
4000
6000
8000
10000

Figure E.9-4

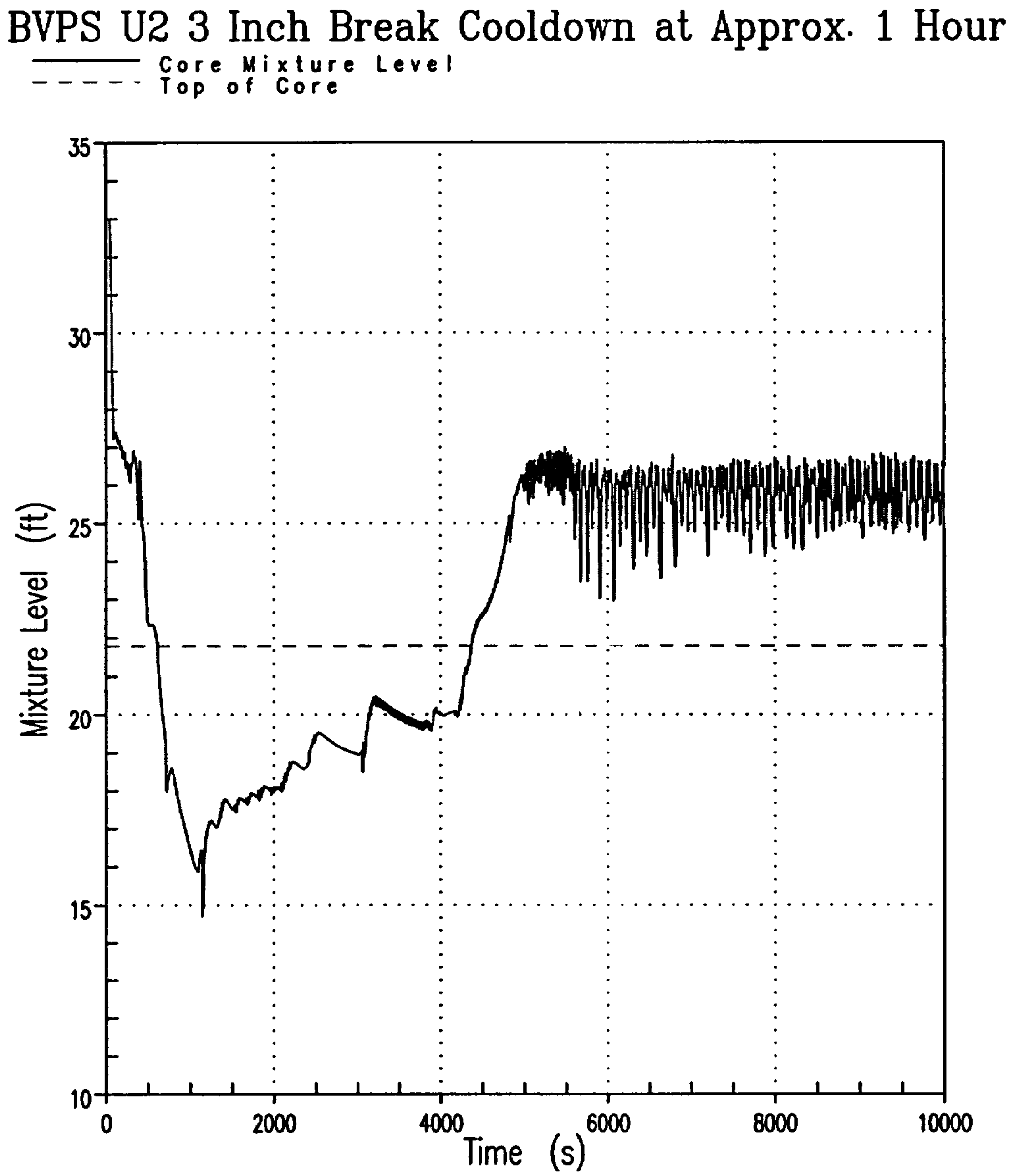
BVPS U2 3 Inch Break Cooldown at Approx. 1 Hour
Core Mixture Level
Top of Core
35
30
25
20
15
10
Mixture Level (ft)
0
2000
4000
6000
8000
10000
Time (s)

Figure E.9-5

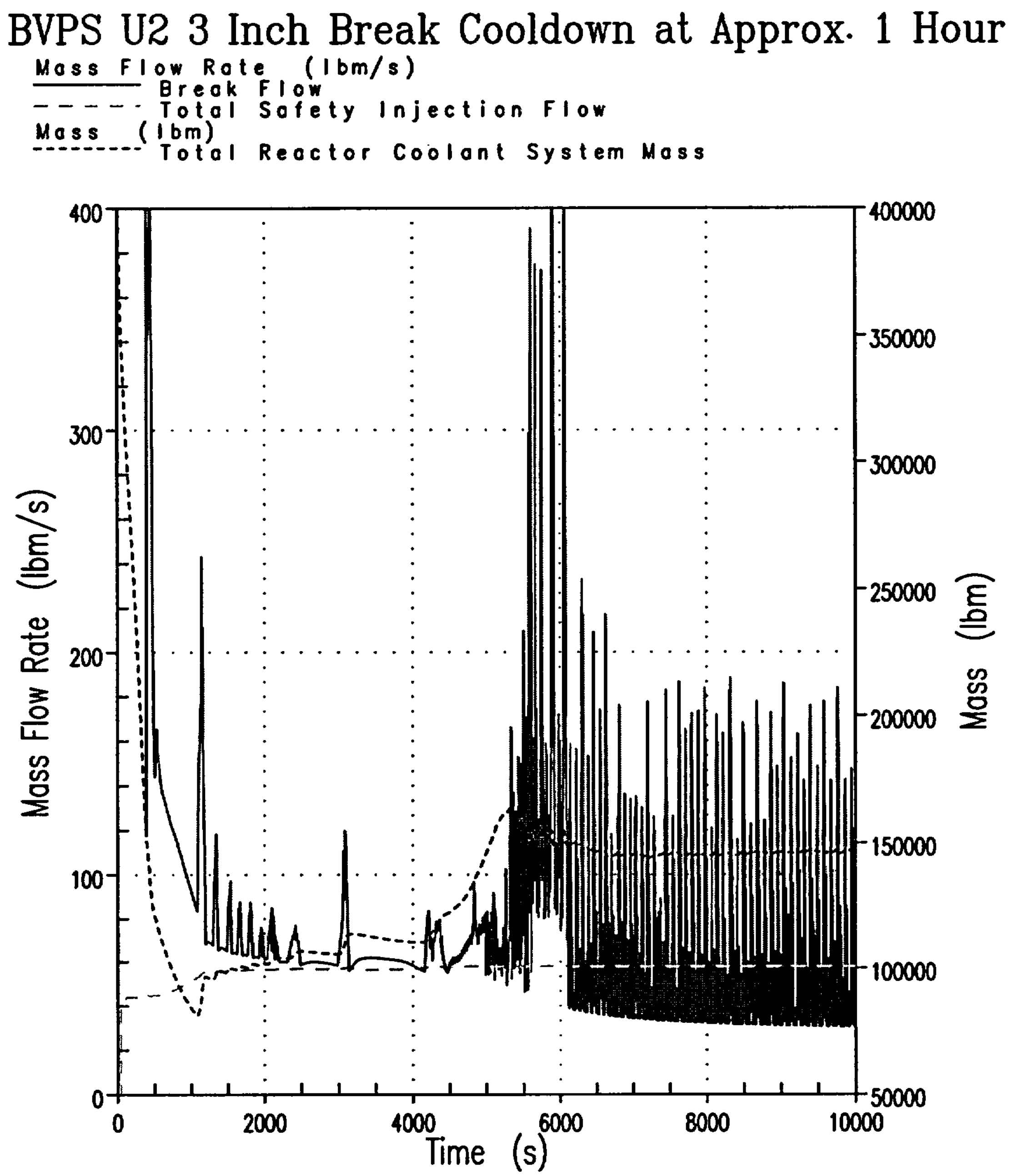
BVPS U2 3 Inch Break Cooldown at Approx. 1 Hour
Mass Flow Rate (lbm/s)
Break Flow
Total Safety Injection Flow
Mass (lbm)
Total Reactor Coolant System Mass
Mass Flow Rate (lbm/s)
Mass (lbm)
Time (s)
0
100
200
300
400
50000
100000
150000
200000
250000
300000
350000
400000
0
2000
4000
6000
8000
10000

Figure E.9-6

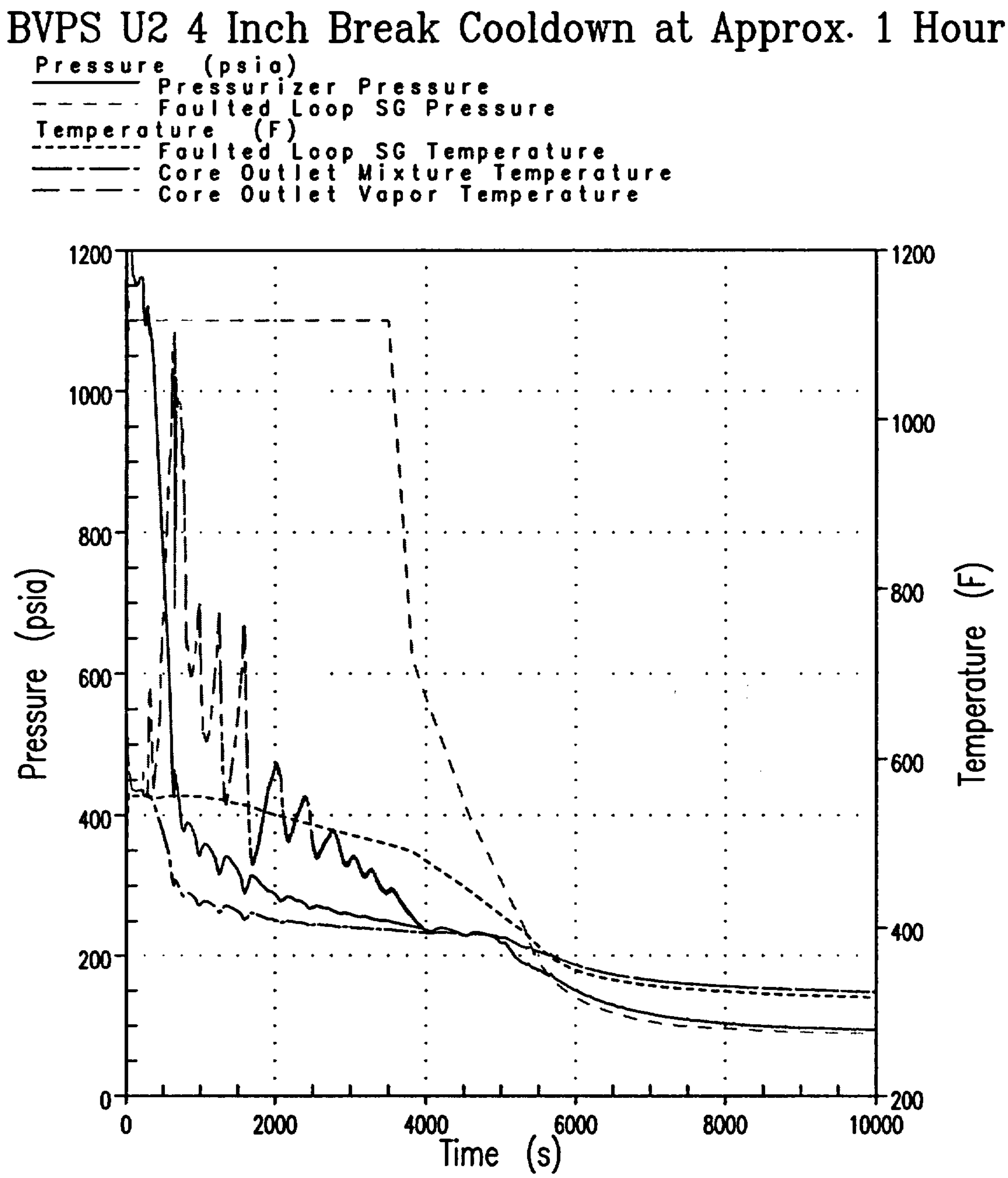
BVPS U2 4 Inch Break Cooldown at Approx. 1 Hour
Pressure (psia)
Pressurizer Pressure
Faulted Loop SG Pressure
Temperature (F)
Faulted Loop SG Temperature
Core Outlet Mixture Temperature
Core Outlet Vapor Temperature
Pressure (psia)
Temperature (F)
Time (s)
0
200
400
600
800
1000
1200
2000
4000
6000
8000
10000

Figure E.9-7

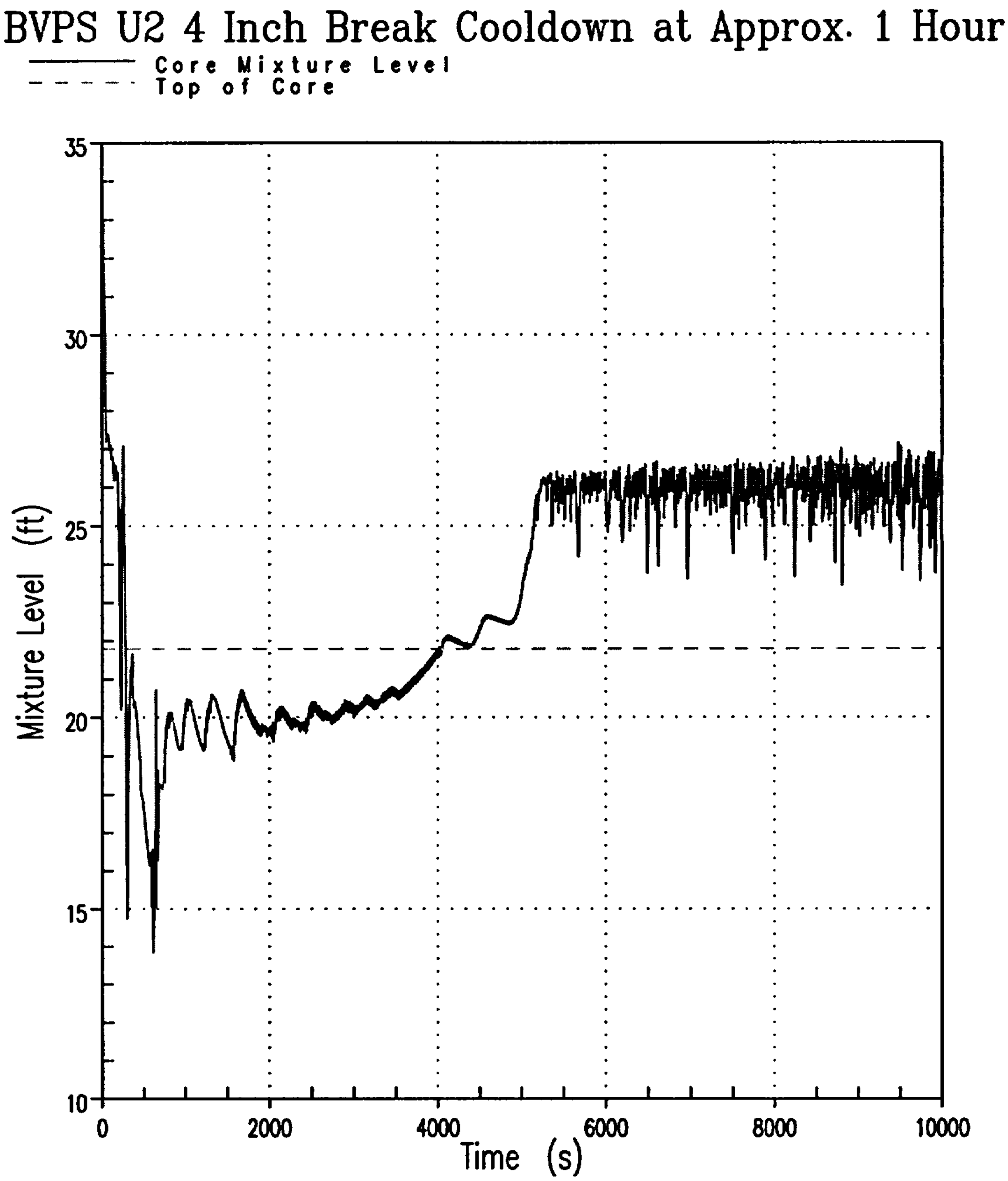
BVPS U2 4 Inch Break Cooldown at Approx. 1 Hour
Core Mixture Level
Top of Core
Mixture Level (ft)
35
30
25
20
15
10
0
2000
4000
6000
8000
10000
Time (s)

Figure E.9-8

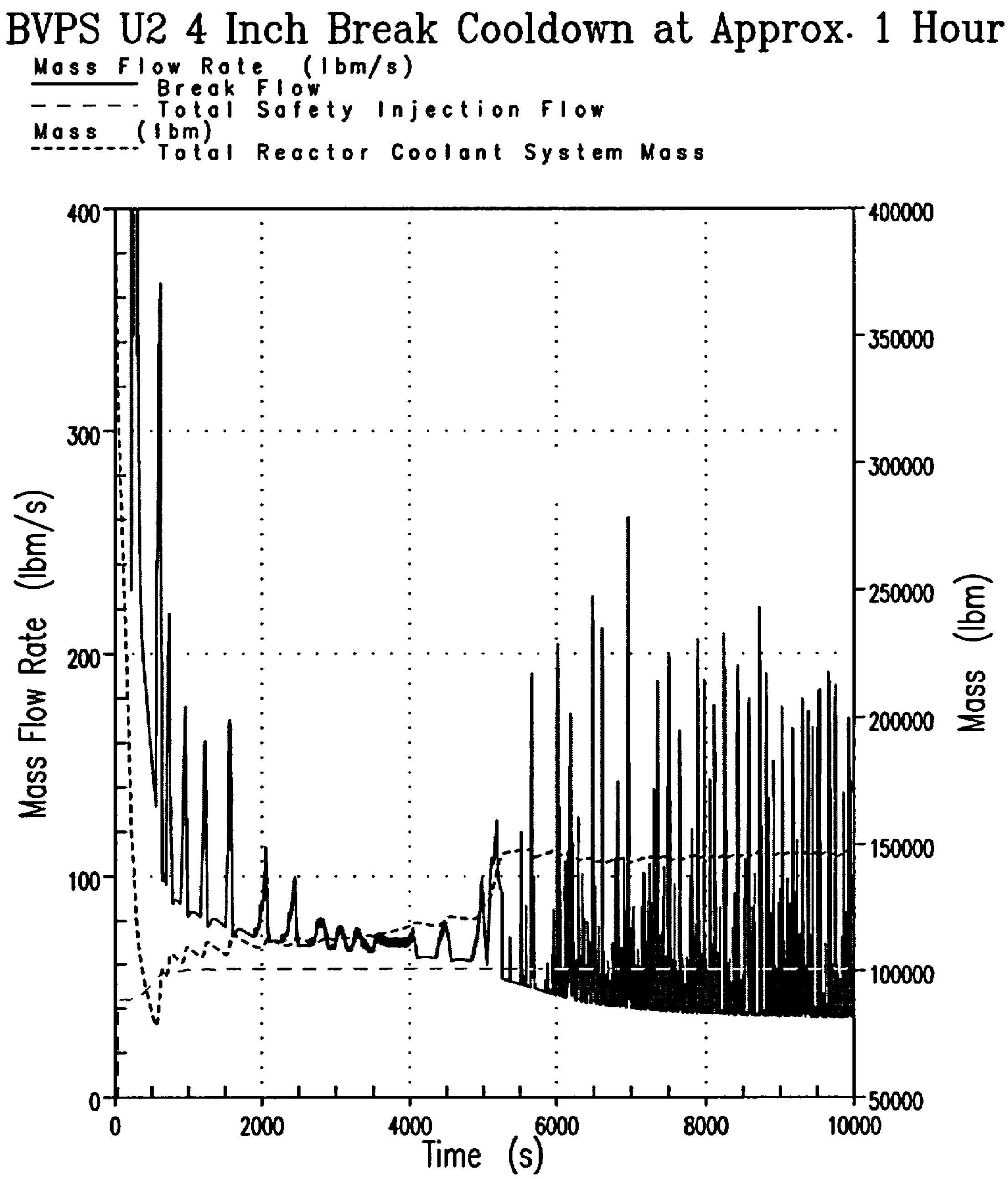
BVPS U2 4 Inch Break Cooldown at Approx. 1 Hour
Mass Flow Rate (lbm/s)
Break Flow
Total Safety Injection Flow
Mass (lbm)
Total Reactor Coolant System Mass
Mass Flow Rate (lbm/s)
Mass (lbm)
Time (s)
0
100
200
300
400
2000
4000
6000
8000
10000
50000
100000
150000
200000
250000
300000
350000
400000

Figure E.9-9

a,c

Figure E.9-10

a,c

Figure E.9-11

a,c

Figure E.9-12

[ ]a,c

Figure E.9-13

a,c

Figure E.9-14

a,c

Figure E.9-15

Question E.11 (Applicable to RSG & EPU)

For some plants, analyses of small breaks in the 0.5 ft^2 to 1.0 ft^2 range can show these breaks to be limiting since the core totally uncovers. While the duration of the uncovery is shorter for these larger small breaks, the PCT is dependent on the accumulator design to prevent temperatures from exceeding 2200°F. Please confirm, based on BVPS-1 and 2-specific conditions, that the NOTRUMP code produces less limiting PCTs for breaks 0.5 to 1.0 ft^2 compared to the limiting breaks controlled by HHSI (0.02 - 0.04 ft^2).

Response:

The nature of the Westinghouse Nuclear Steam Supply System (NSSS) is that all accumulators are at relatively high pressures (between 575 and 715 psia). As such, the Westinghouse NSSS designs are less sensitive to breaks in the 0.5- to 1.0-ft^2 size due to the nature of the transient (i.e. rapid depressurization to accumulator set-points and ultimately to Low Head Safety Injection (LHSI)).

To demonstrate the nature of the Westinghouse NSSS design response, a 0.5- and 1.0-ft^2 break was simulated with BVPS-1. Figure E.11-1 through Figure E.11-12 presents the results of this study. The results indicate, as expected, that the smaller break sizes remain limiting with respect to Peak Cladding Temperature (PCT) and cladding oxidation.

Figure E.11-1 through Figure E.11-6 presents the results of the 0.5-ft^2 break simulation for BVPS-1. Figure E.11-1 demonstrates the rapid depressurization characteristics for breaks in this size range. As can be seen, the RCS pressure drops rapidly to the accumulator actuation set-point (625 psia) in about 100 seconds. This is evident in Figure E.11-5 which shows the RCS inventory recovery observed beginning at that time. The core/upper plenum mixture level response for this simulation is presented in Figure E.11-2. As seen, no significant sustained uncovery is predicted for this simulation.

To supplement the results, the core nodal mixture region qualities were compared to the 90% quality limit. The 90 percent quality limit is discussed in Reference E.11-1 as a predictor of dryout conditions since void fractions indicative of the critical heat flux (as predicted by Griffith's modification of the Zuber equation) (Figure E.11-2 and Figure E.11-3) would be obtained at core exit qualities above this value. The core mixture level is then adjusted to be the minimum of the active mixture level and the lowest core node that exceeds this limit.

Figure E.11-4 presents the nodal quality check for the 0.5-ft^2 break size. For this break, the dryout quality is not approached except when the mixture level drops out of the specific core node. Since the quality limit was not exceeded for this simulation, the mixture level was not modified as seen in Figure E-11-5. Finally, Figure E.11-6 presents the PCT response which demonstrates that this break size is less limiting than predicted for the smaller break sizes previously analyzed.

Figure E.11-7 through Figures E.11-12 presents the results for the 1.0-ft^2 simulation for BVPS-1. Figure E.11-7 demonstrates the rapid depressurization characteristics for breaks in this size range. As can be seen, the RCS pressure drops rapidly to the accumulator actuation set-point in about 60 seconds. This is evident in Figure E.11-9 which commences to recover RCS inventory at that time. The raw core/upper plenum mixture level response for this simulation is presented in Figure E.11-8. It can be observed that a period of blowdown uncovery is predicted for this simulation case. To further supplement this level response, the dryout onset comparison was once again performed.

Figure E.11-10 presents the nodal quality check for the 1.0-ft^2 break size. For this break, the dryout quality is exceeded during the blowdown period. Since the quality limit was exceeded, the mixture level was modified as described earlier. Figure E.11-11 presents the raw and adjusted core mixture levels for this transient simulation. Finally, Figure E.11-12 presents the PCT response for this simulation utilizing the adjusted core mixture level response. As can be seen, this response is also much less limiting than that observed for the smaller break sizes analyzed in the NOTRUMP EM break spectrum.

[ ] a,c

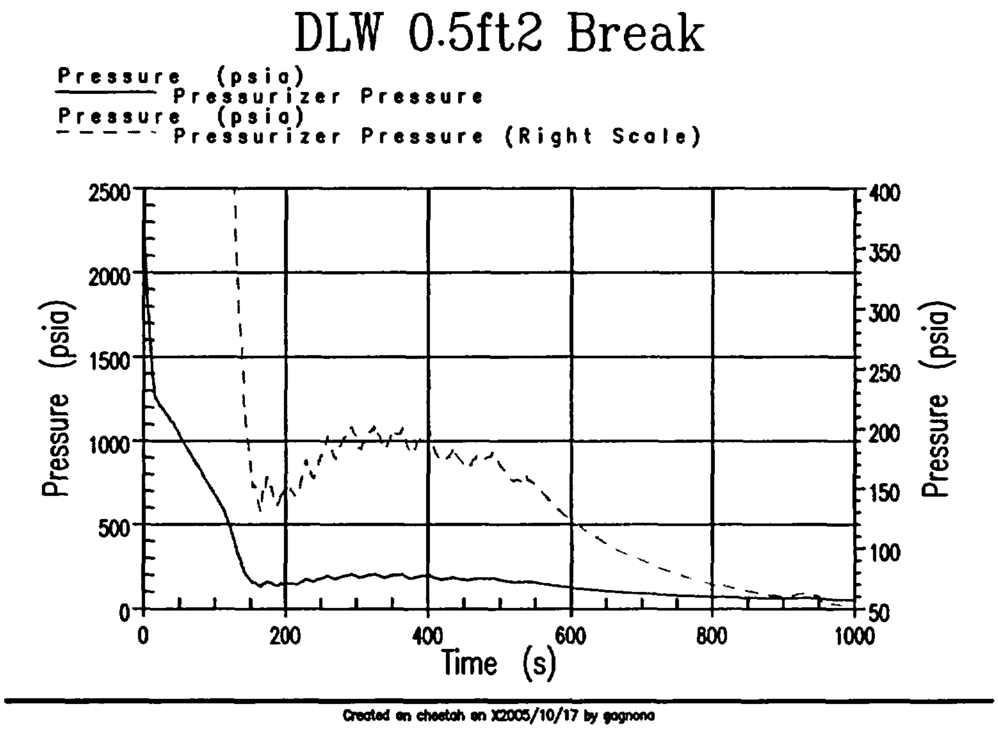

Figure E.11-1 BVPS-1 0.5-ft² Break, RCS Pressure Response

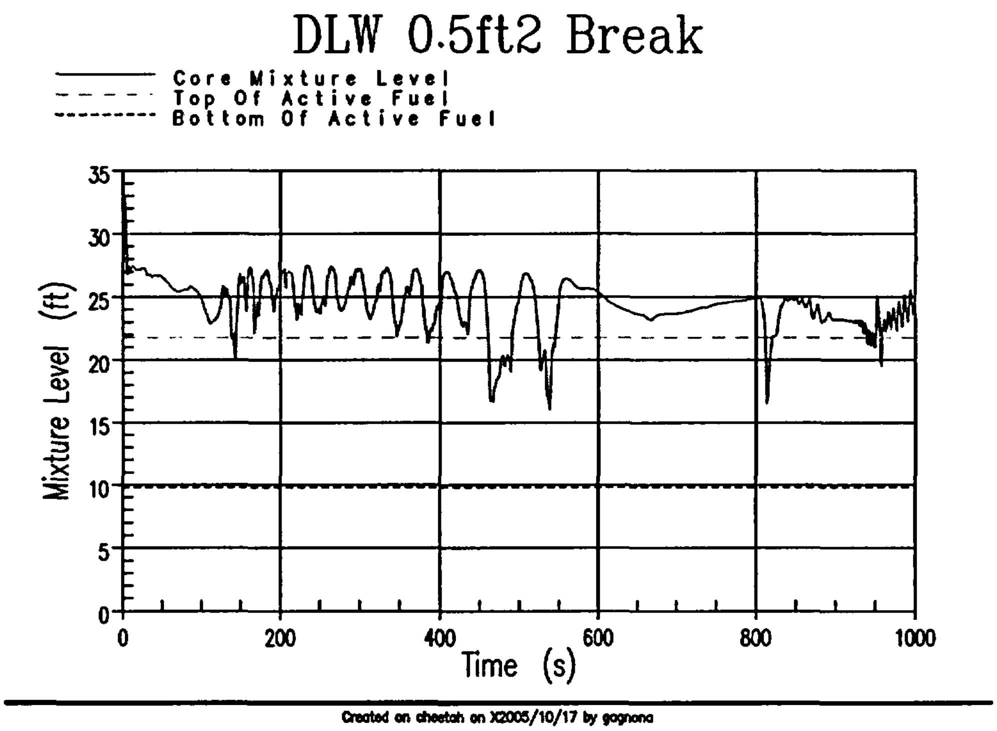

Figure E.11-2 BVPS-1 0.5-ft^2 Break, Core/Upper Plenum Mixture Level

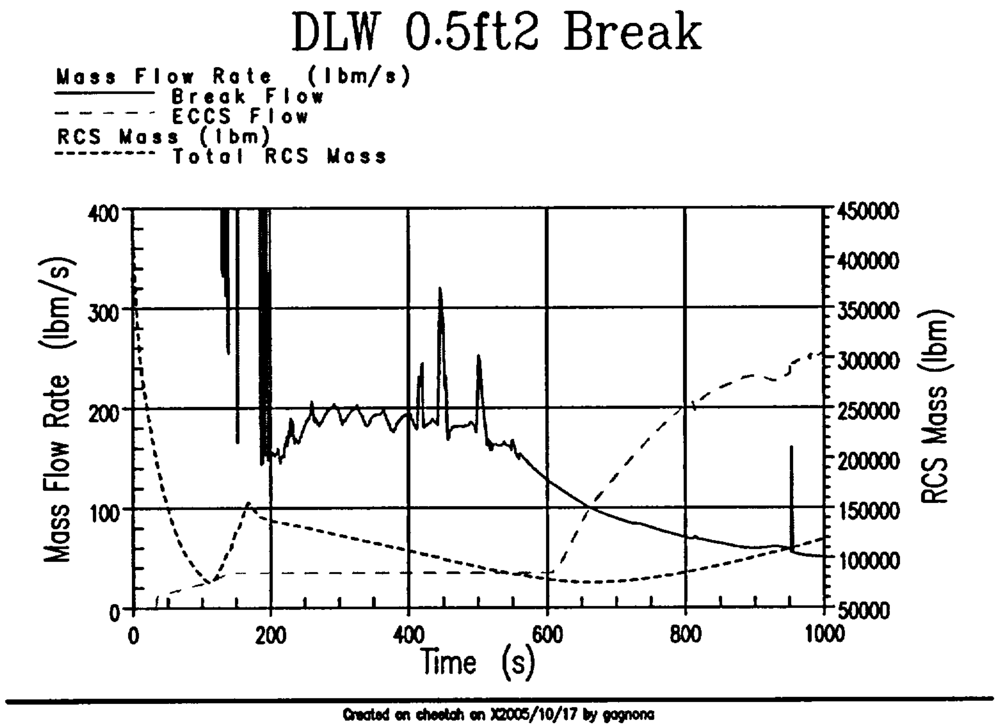

Figure E.11-3 **BVPS-1 0.5-ft² Break, Safety Injection Flow vs. Break Flow**

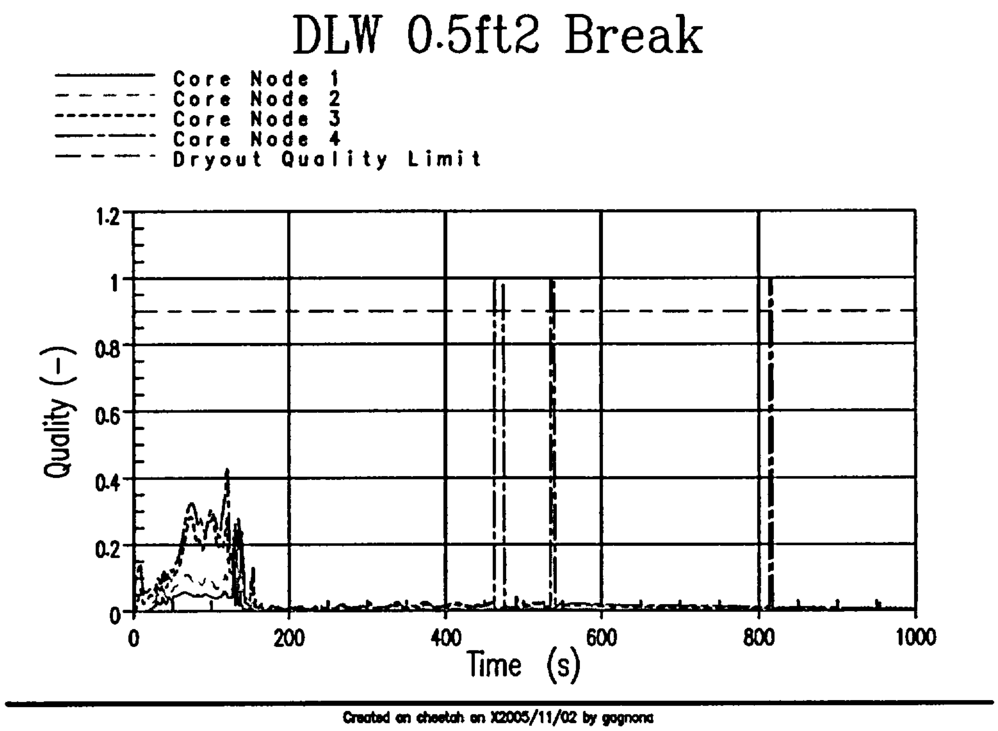

Figure E.11-4 BVPS-1 0.5-ft^2 Break, Dryout Quality Check

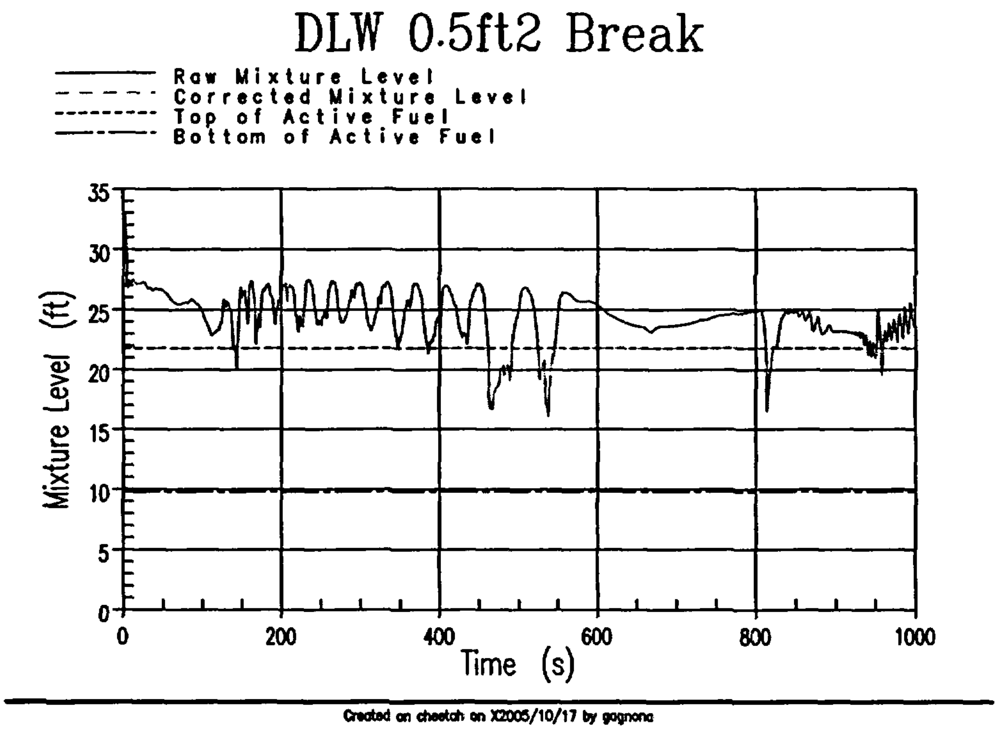

Figure E.11-5 BVPS-1 0.5-ft^2 Break, Adjusted Core Level

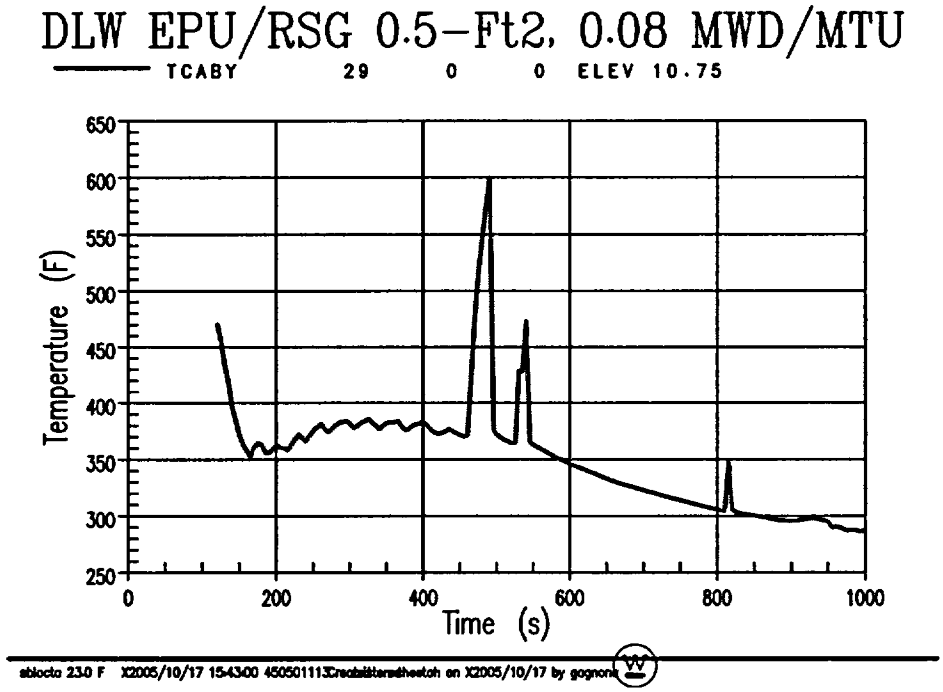

Figure E.11-6 BVPS-1 0.5-ft^2 Break, Peak Cladding Temperature At 10.75 ft

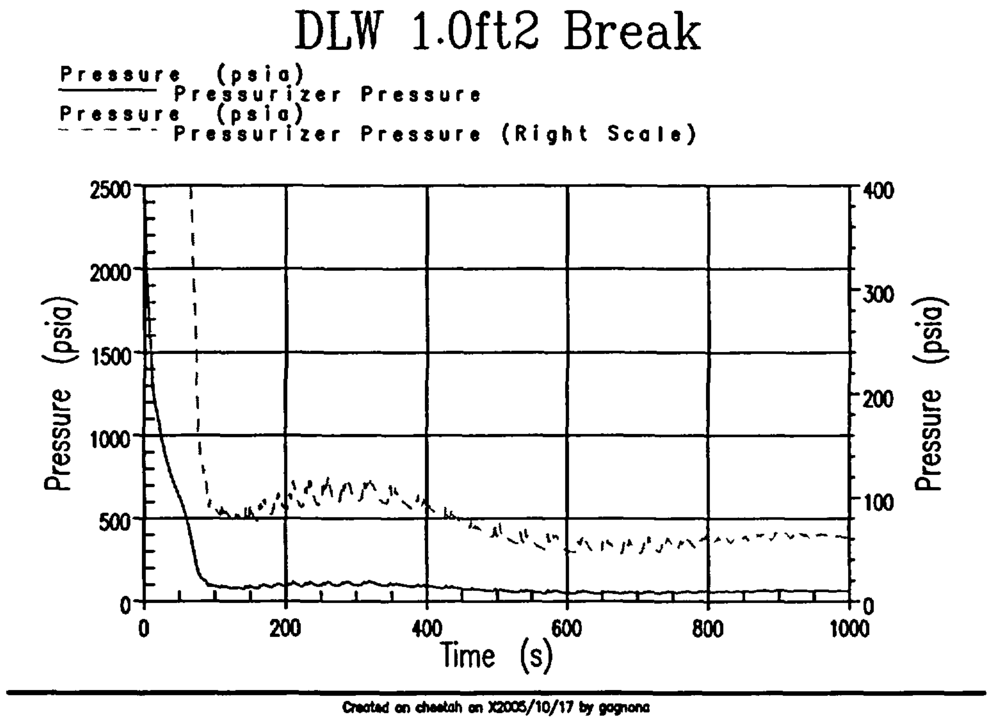

Figure E.11-7 BVPS-1 1.0-ft^2 Break, RCS Pressure Response

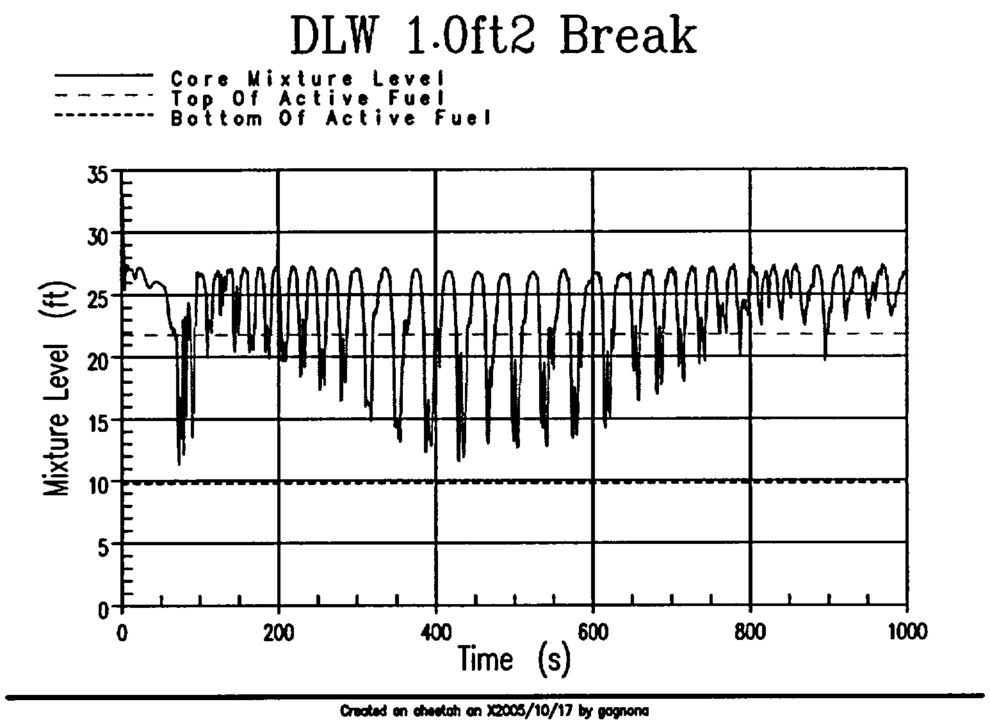

Figure E.11-8 BVPS-1 1.0-ft^2 Break, Core/Upper Plenum Mixture Level

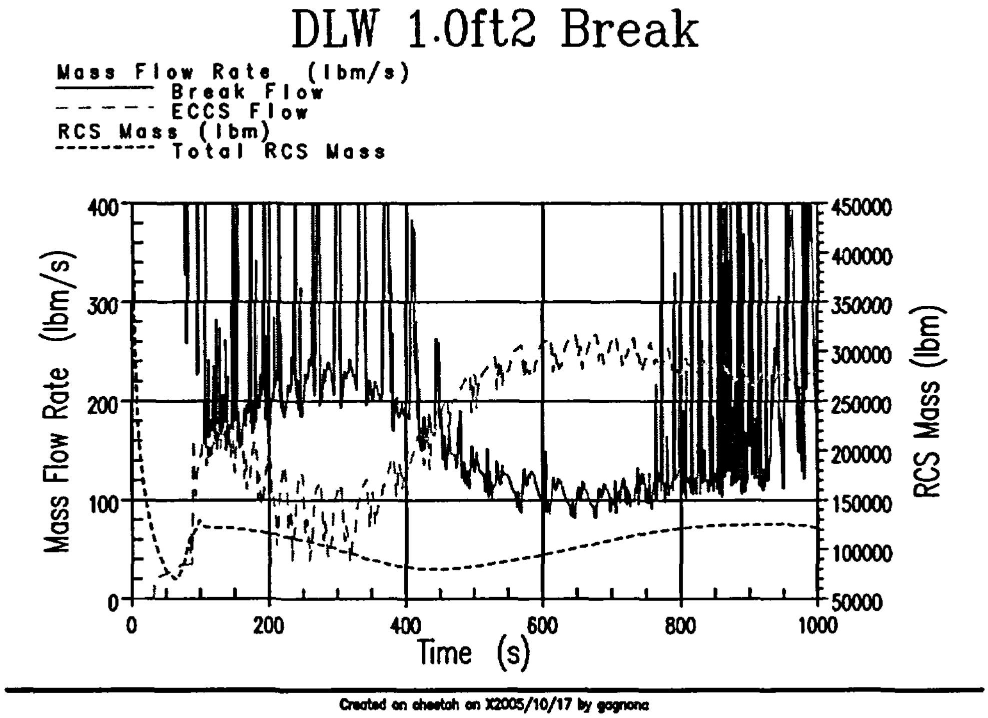

Figure E.11-9 BVPS-1 1.0-ft^2 Break, Safety Injection Flow vs. Break Flow

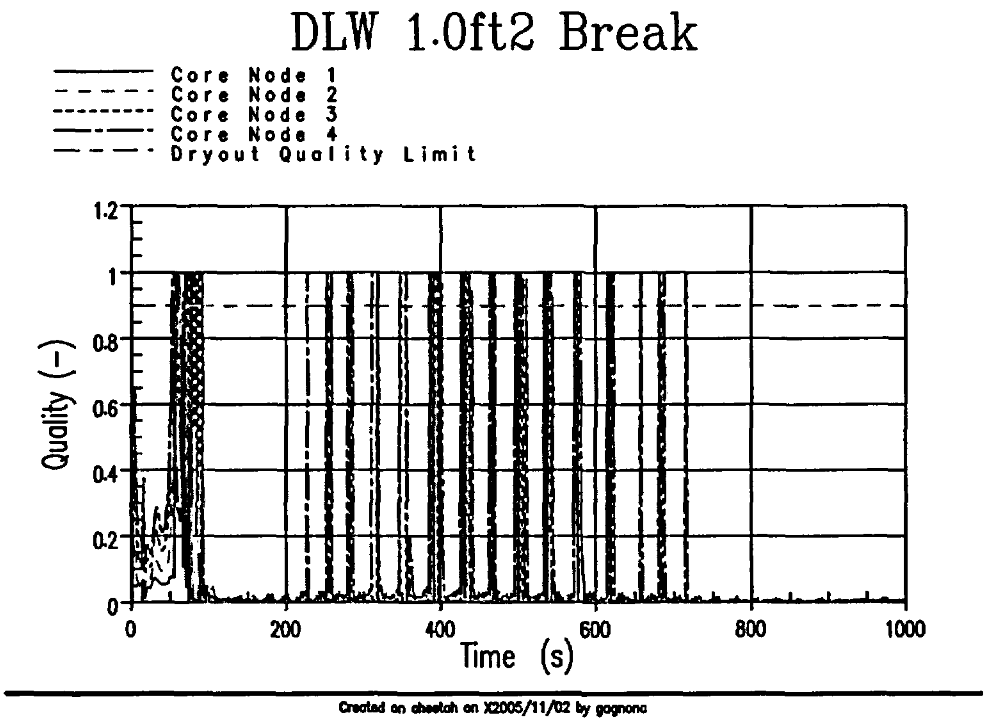

Figure E.11-10 BVPS-1 1.0-ft^2 Break, Dryout Quality Check

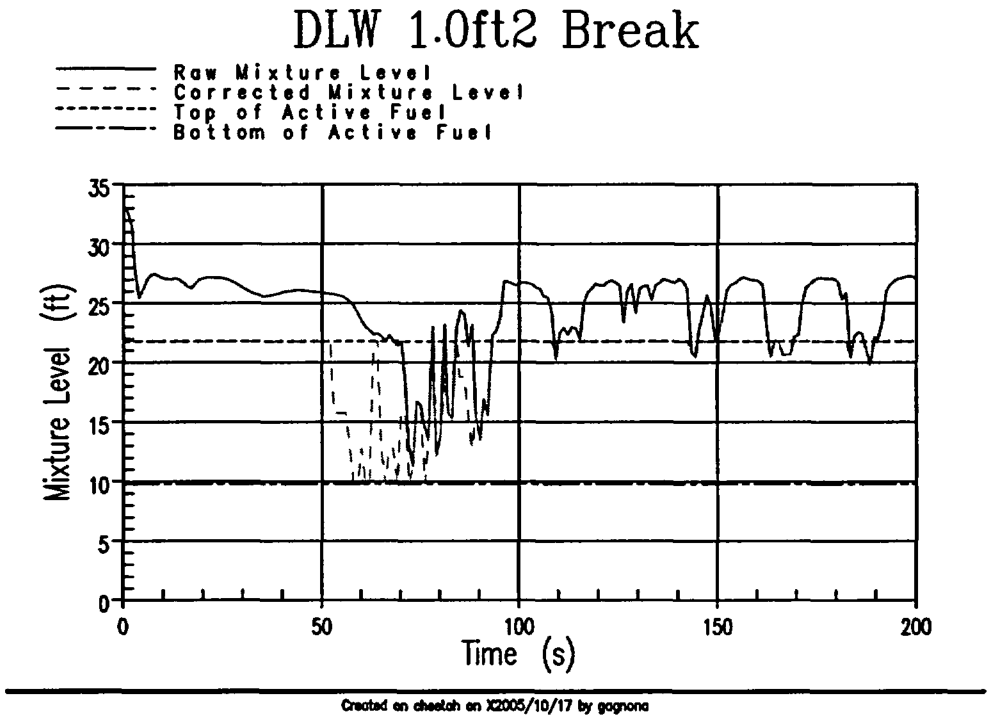

Figure E.11-11 BVPS-1 1.0-ft^2 Break, Adjusted Core Level

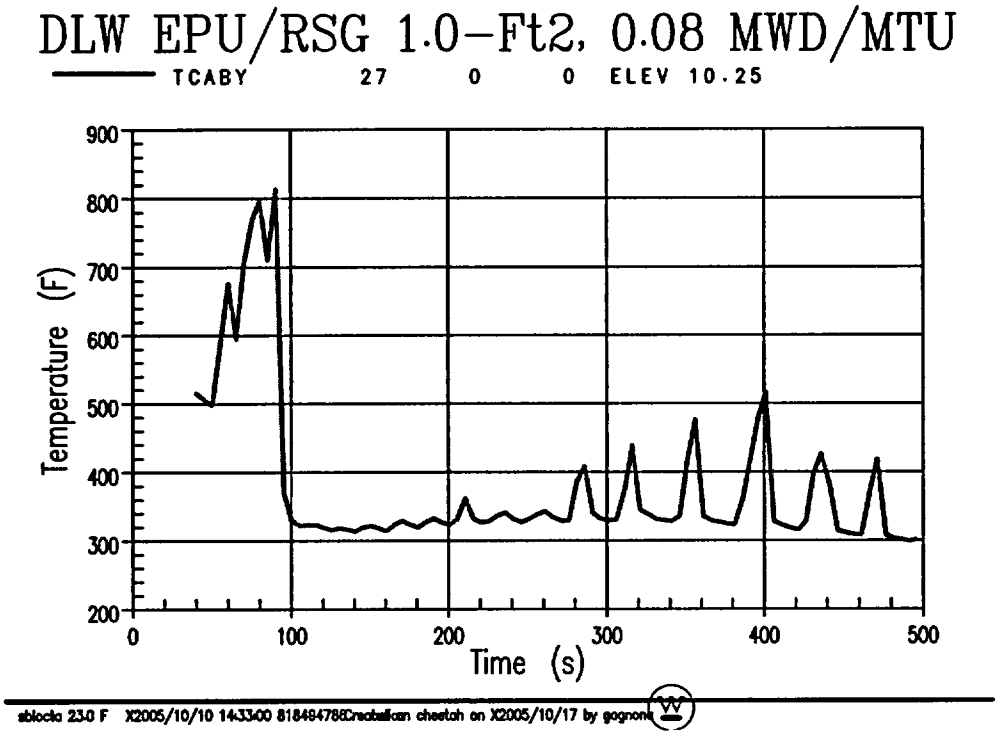

Figure E.11-12 BVPS-1 1.0-ft^2 Break, Peak Cladding Temperature At 10.25 ft

References

E.11-1 WCAP-13071, "An Evaluation of the Revised Transfer to Cold Leg Recirculation Procedure, H. B. Robinson Unit 2," M. P. Kachmar, September 1991.

E.11-2 Zuber, et. al., "The Hydrodynamic Crisis in Pool Boiling of Saturated and Subcooled Liquids," Part II, No. 27, International Developments in Heat Transfer, 1961.

E.11-3. Griffith, et. al., "PWR Blowdown Heat Transfer," Thermal and Hydraulic Aspects of Nuclear Reactor Safety, ASME, New York, Volume I, 1977.

Question E.13 (Applicable to RSG & EPU)

Fig. E13.1 shows the steaming rate of 50 lbs/sec at 5000 sec for the 3-inch break, yet the injection into the intact loops is only about 38 lb/sec at this time shown in Fig. 5.2.2-11A. As such, the two-phase level should decrease with time until the core uncovers to the lower steaming rate of 38 lb/sec. Please explain why the core is not uncovering more and how the two-phase level is recovered at EPU conditions for this break.

Likewise, for the 2-inch break, the steam rate is about 60 lb/sec at 6000 sec, while the injection into the intact loop is about 33 lbs/sec. Explain why the core level does not uncover to a lower two-phase level where the steaming rate is about 33 lb/sec or that equal to the injection.

Please also provide the core inlet mass flow rate and inlet temperature vs. time for all breaks.

Response:

Note: The Beaver Valley Power Station Unit Nos. 1 and 2 SBLOCA Transients have been re-performed; however, since the revised transients exhibit the same characteristics, the question continues to apply. The discussion presented herein relates to the original License Amendment Report (LAR) transient simulations (Reference E.13-1).

While the core steaming rate may be higher than the calculated safety injection from the intact loops, this term does not constitute whether or not inventory recovery is occurring. The NOTRUMP Evaluation Model (EM) models safety injection to both the intact and broken loops, as such, RCS inventory recovery is a function of total safety injection flow (including both accumulators) vs. total break flow. Figure E.13-1 presents a comparison of the BVPS-1 3-inch break total break flow vs. total safety injection flows. As can be seen, total safety injection flow is greater than or equal to total break flow with the accumulators providing additional recovery during this time period (Figure E.13-2). The same is true for the 2-inch break (Figure E.13-3).

The differential between the core steaming rates and the total pumped safety injection minus liquid break discharge can be seen in Figure E.13-4 and Figure E.13-5 for the 2-inch and 3-inch breaks respectively. As can be seen, on a pure inventory basis, the core is generating more steam than is available from pumped ECCS alone; however, the effects of pumped safety injection condensation and the steam generators heat sinks must also be considered.

Accounting for only the effects of condensation via ECCS interaction in the cold legs (which is an NRC approved model in NOTRUMP), the differential between core steaming and pumped safety injection can essentially be eliminated without relying on the steam generators (Figure E.13-6 and Figure E.13-8 for the 2- and 3-inch breaks respectively). Figure E.13-7 and Figure E.13-8 present an alternate view of this same behavior by presenting the intact loop cold leg liquid flow to the downcomer versus the core exit vapor generation rate and core/upper plenum mixture level. Figure E.13-10 and Figure E.13-11 demonstrate that the steam generators continue to remove a significant amount of heat over the time period of interest.

As requested, Figure E.13-12 through Figure E.13-25 present the core inlet flow and temperature plots for both BVPS-1 (Figure E.13-12 through Figure E.13-19) and BVPS-2 (Figure E.13-20 through Figure E.13-25).

Reference:

E.13-1 FENOC Letter L-04-125, "Beaver Valley Power Station, Unit No. 1 and 2, License Amendment Request Nos. 302 and 173," October 4, 2004.

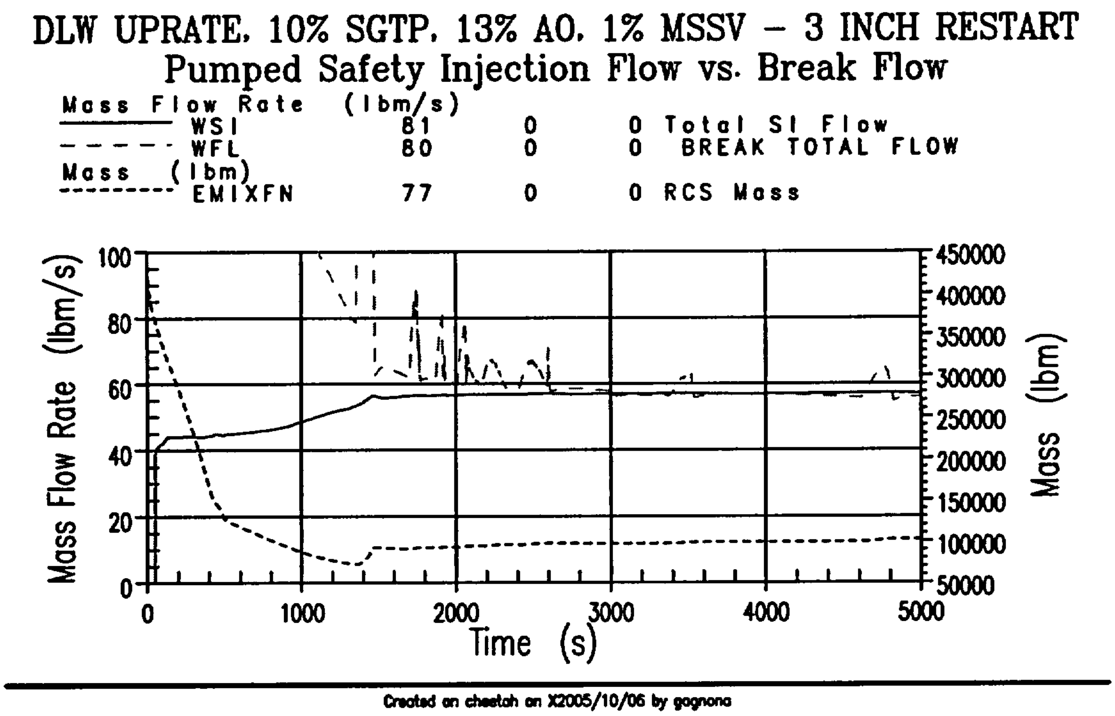

Figure E.13-1 BVPS-1 3-Inch Break, Total Safety Injection vs. Total Break Flow

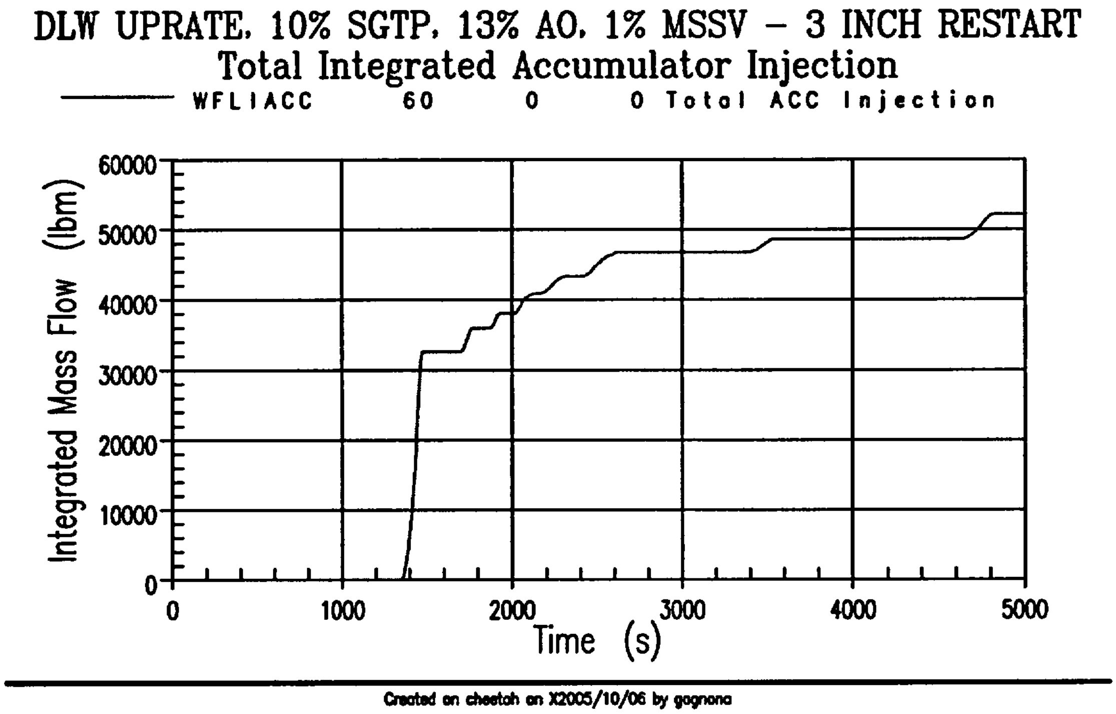

Figure E.13-2 BVPS-1 3-Inch Break, Integrated Accumulator Injection Flow

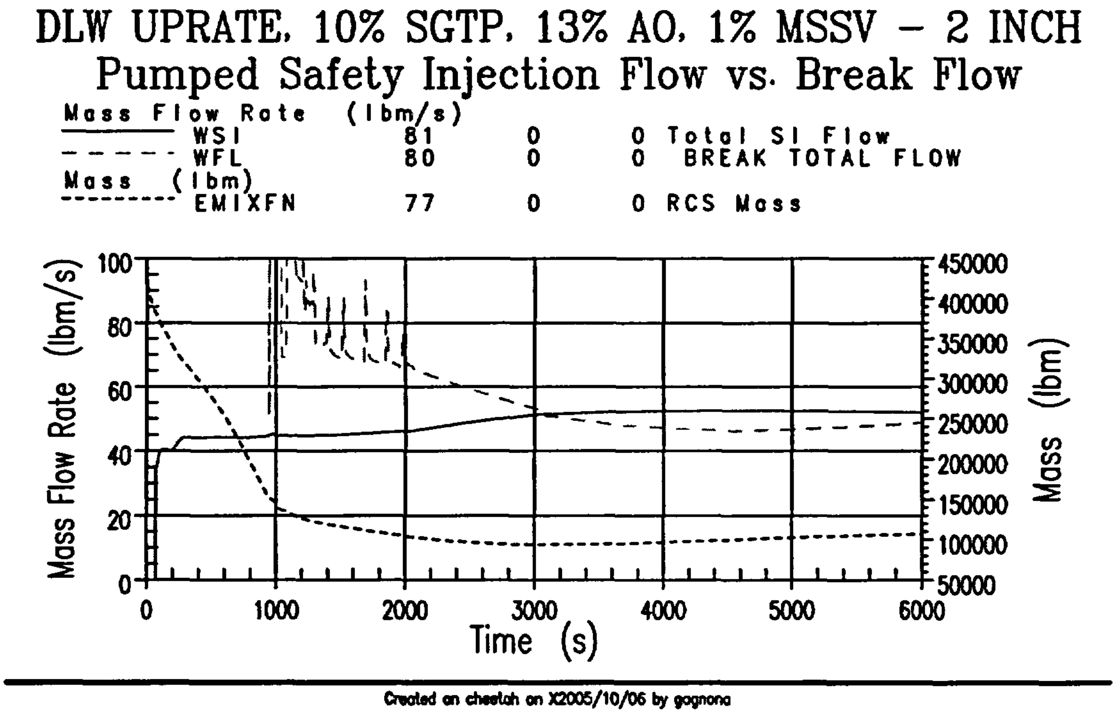

Figure E.13-3 BVPS-1 2-Inch Break, Total Safety Injection vs. Total Break Flow

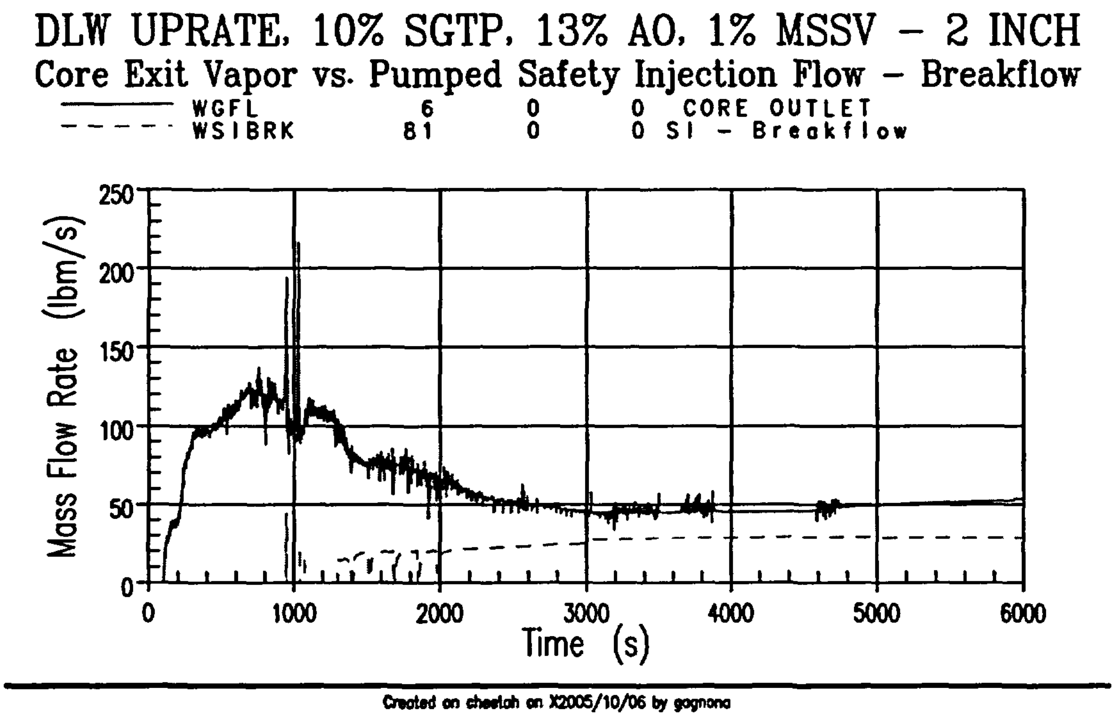

Figure E.13-4 BVPS-1 2-Inch Break, Total Safety Injection vs. Core Exit Steam Flow

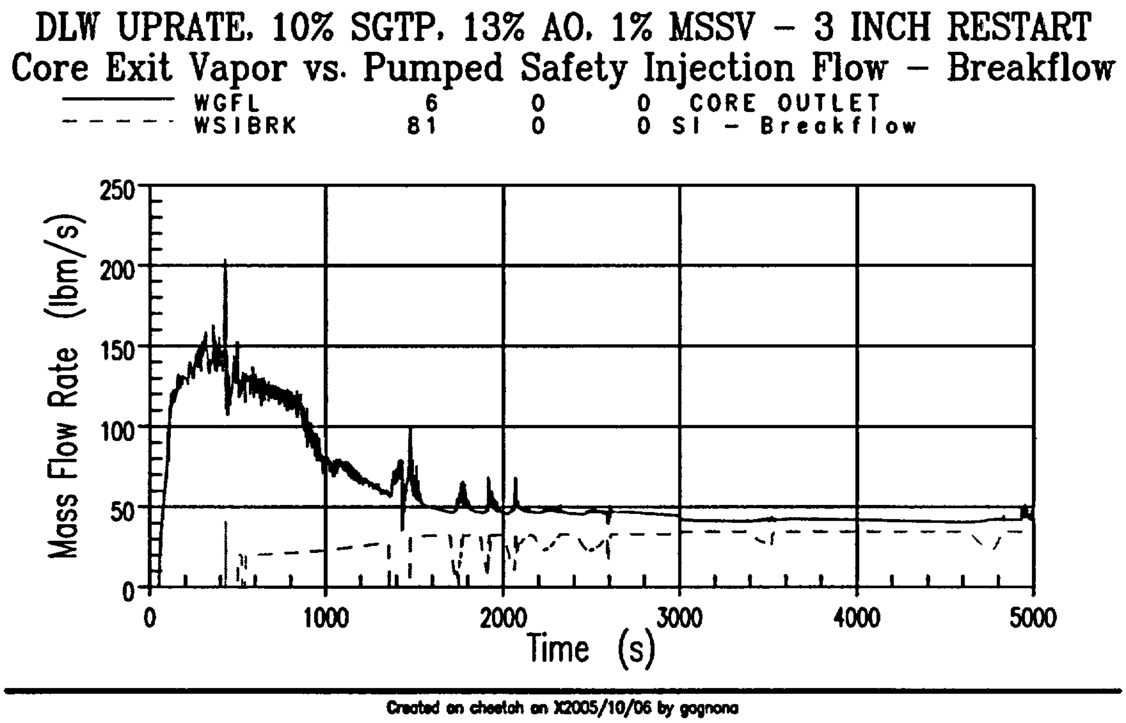

Figure E.13-5 BVPS-1 3-Inch Break, Total Safety Injection vs. Core Exit Steam Flow

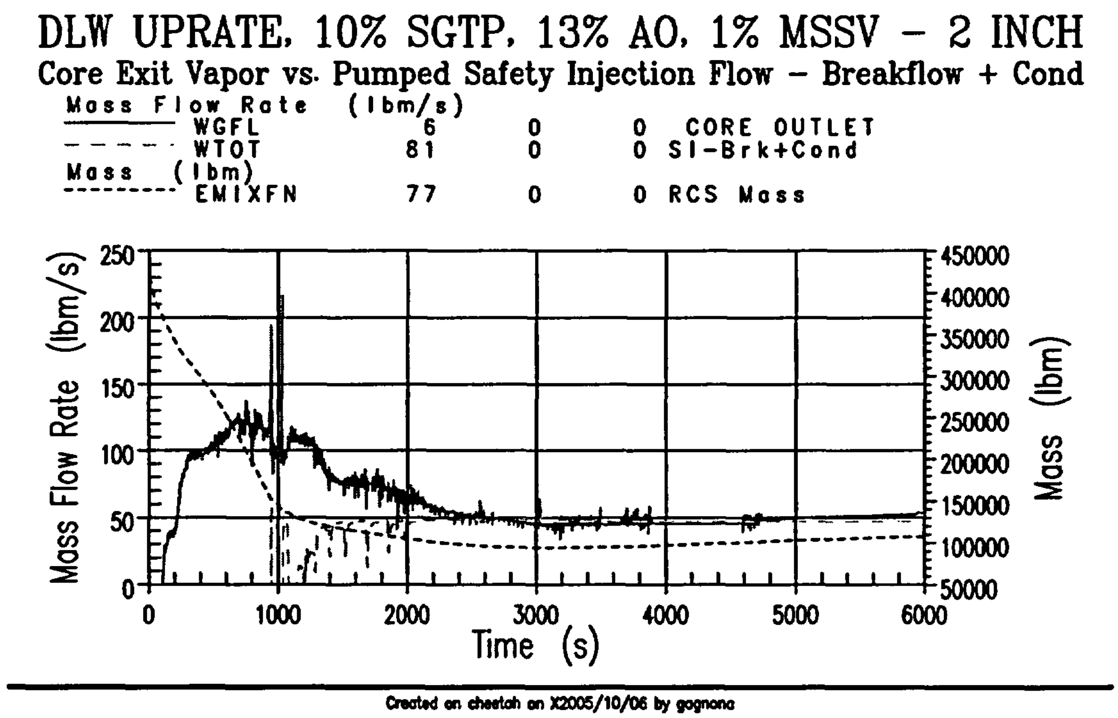

Figure E.13-6 BVPS-1 2-Inch Break, Total Safety Injection + Condensation vs. Core Exit Steam Flow

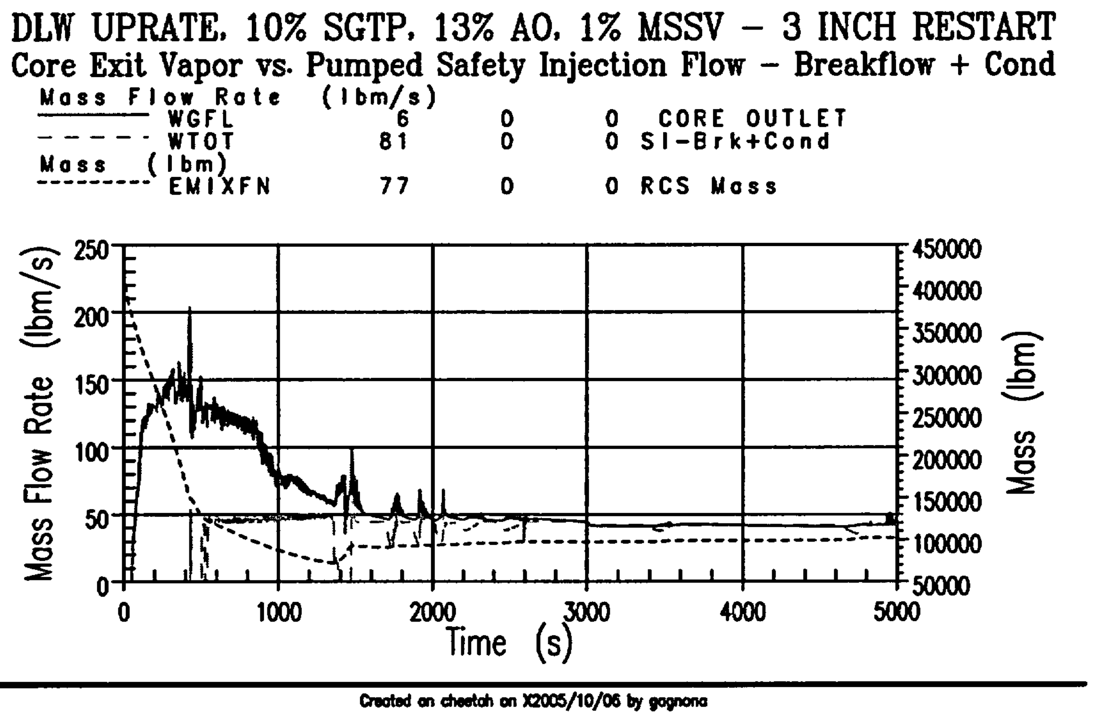

Figure E.13.7 **BVPS-1 3-Inch Break, Total Safety Injection vs. + Condensation vs. Core Exit Steam Flow**

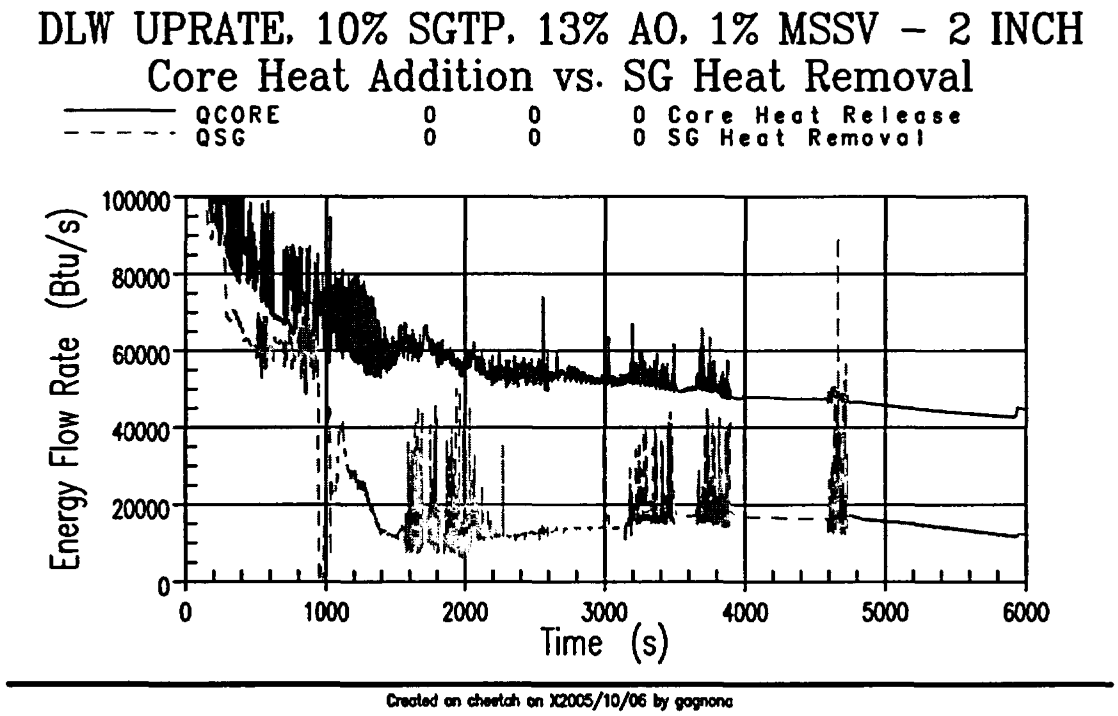

Figure E.13-8 BVPS-1 2-Inch Core Heat Generation vs. SG Heat Removal

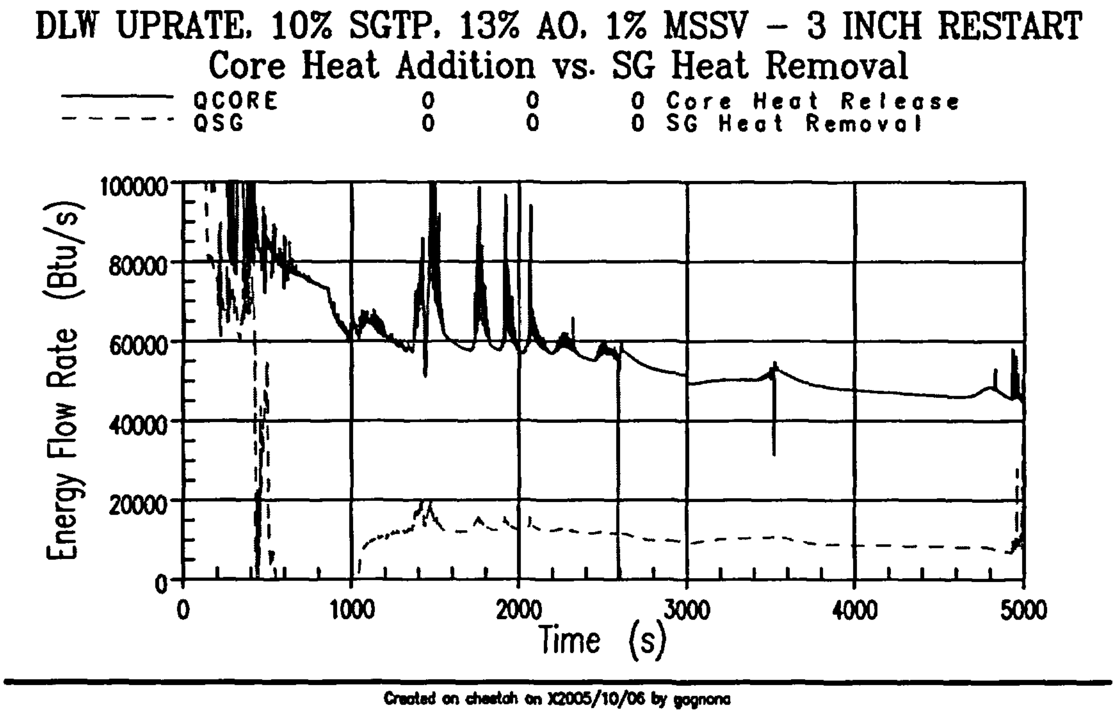

Figure E.13-9 BVPS-1 3-Inch Core Heat Generation vs. SG Heat Removal

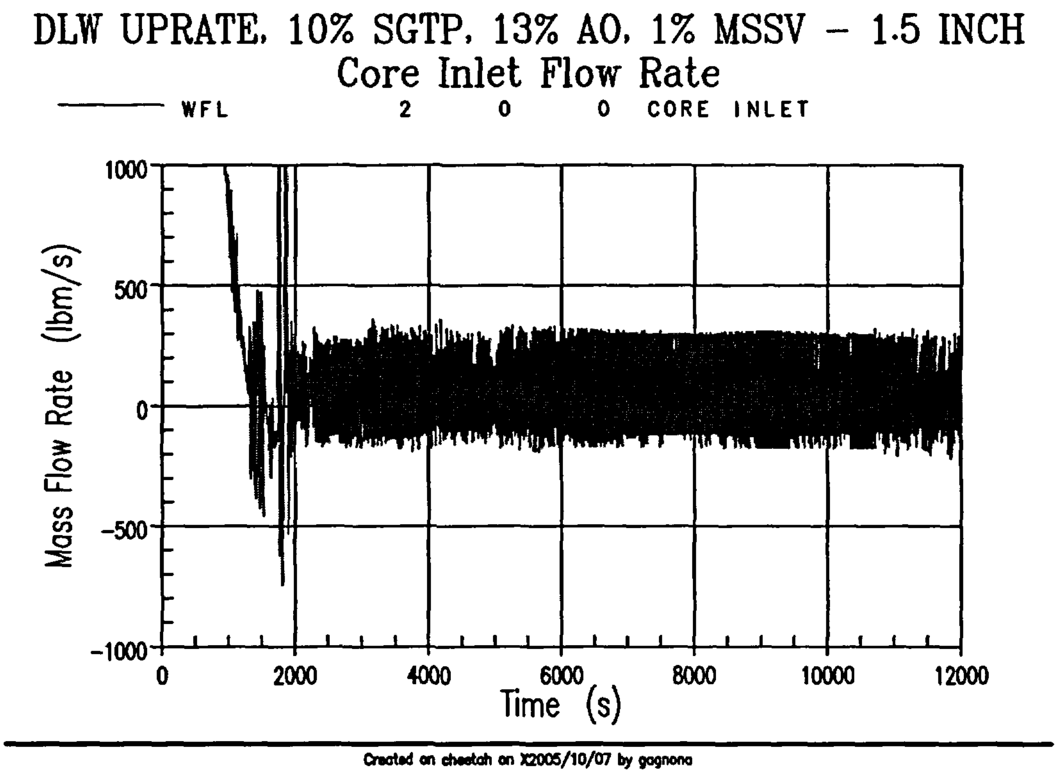

Figure E.13-10 BVPS-1 1.5-Inch Break, Core Inlet Flow

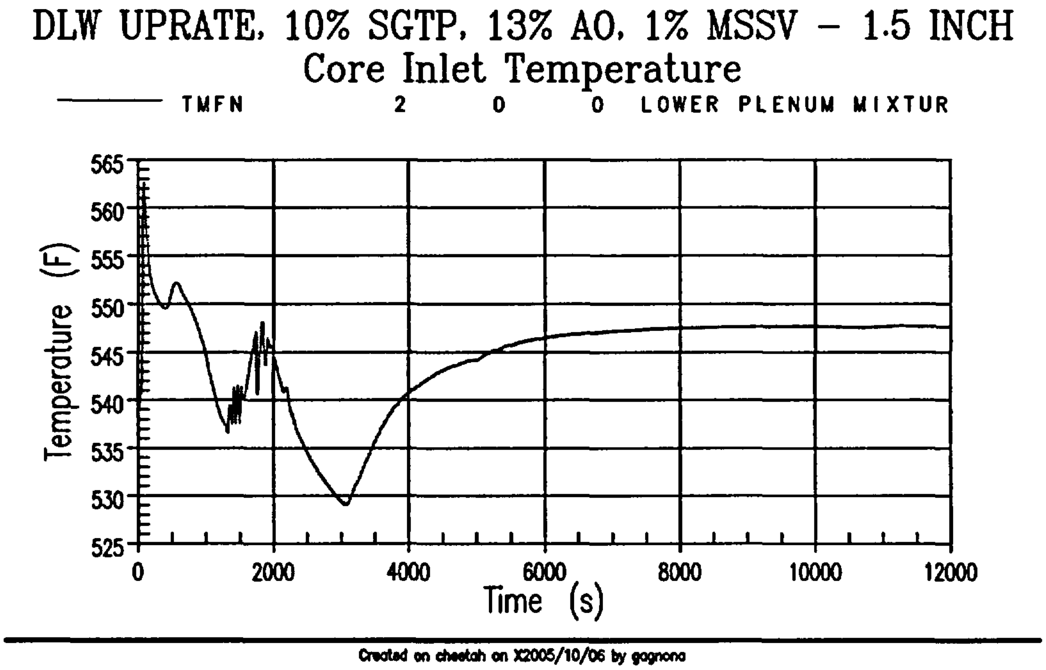

Figure E.13-11 BVPS-1 1.5-Inch Break, Core Inlet Temperature

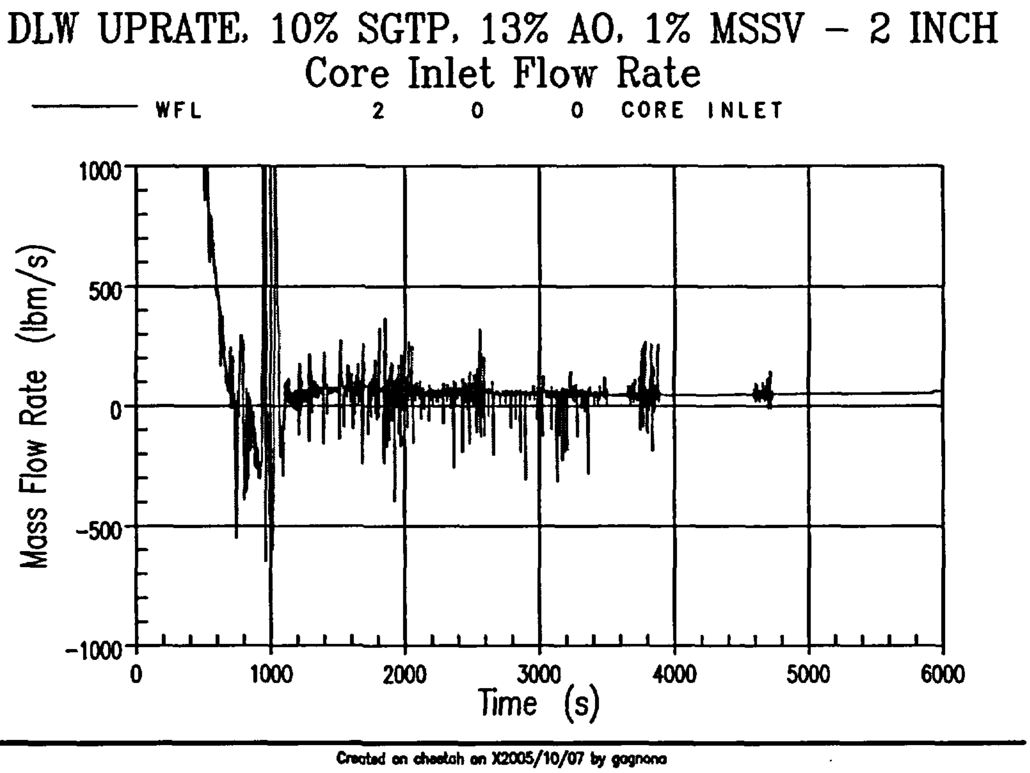

Figure E.13-12 BVPS-1 2-Inch Break, Core Inlet Flow

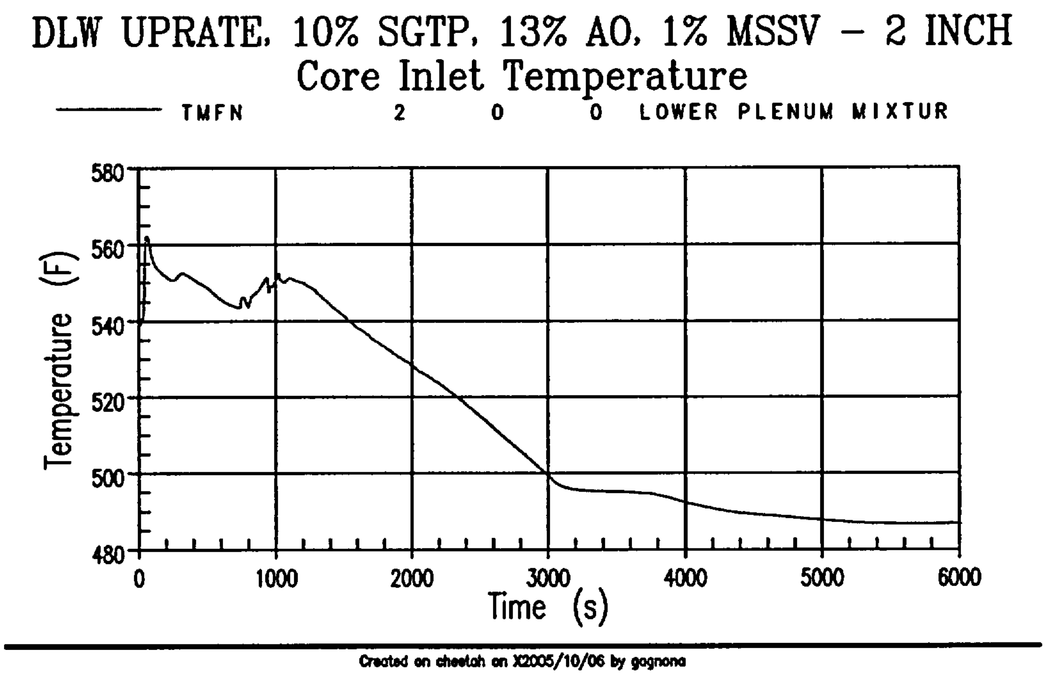

Figure E.13-13 BVPS-1 2-Inch Break, Core Inlet Temperature

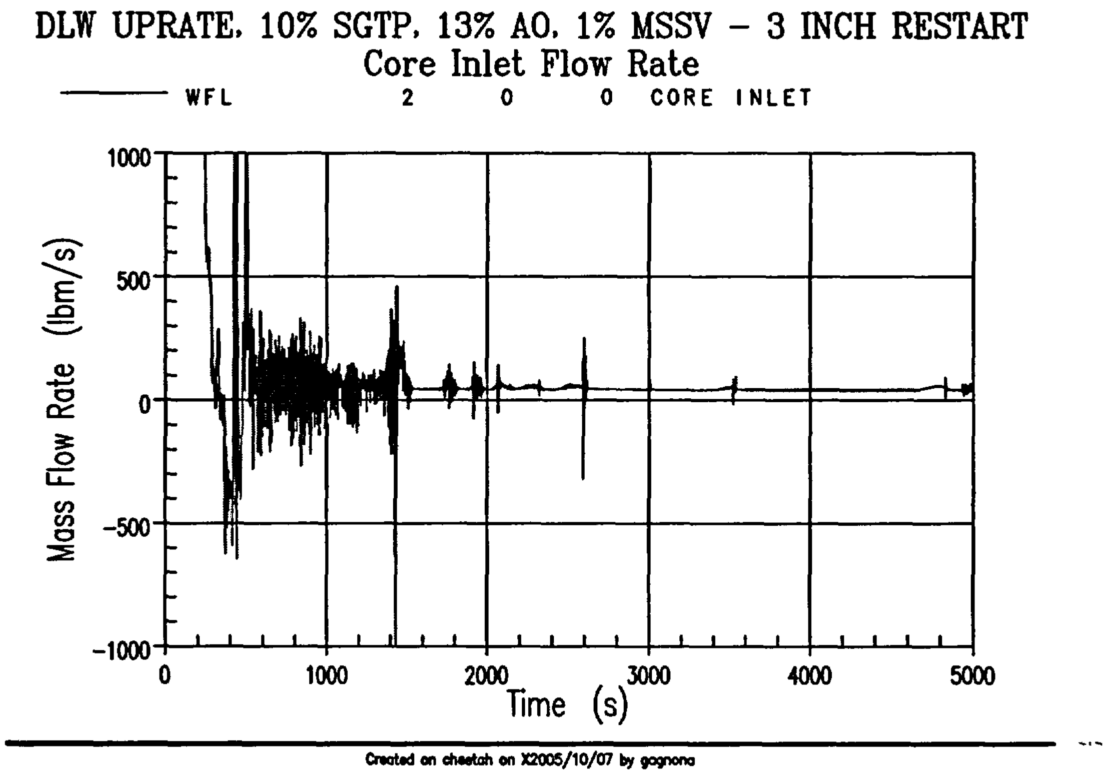

Figure E.13-14 BVPS-1 3-Inch Break, Core Inlet Flow

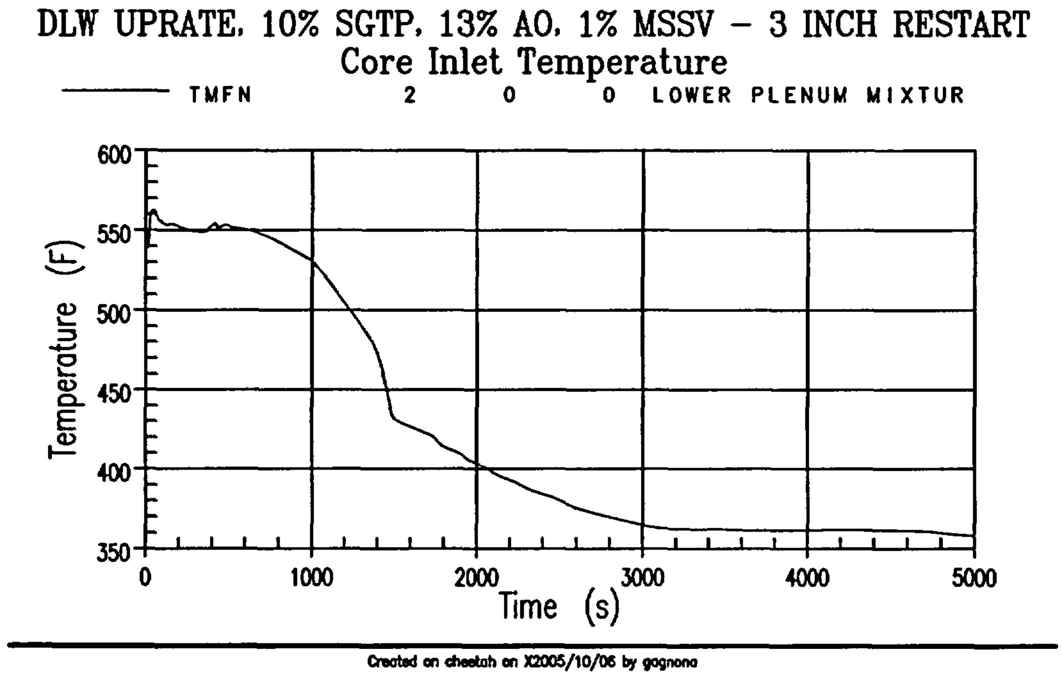

Figure E.13-15 BVPS-1 3-Inch Break, Core Inlet Temperature

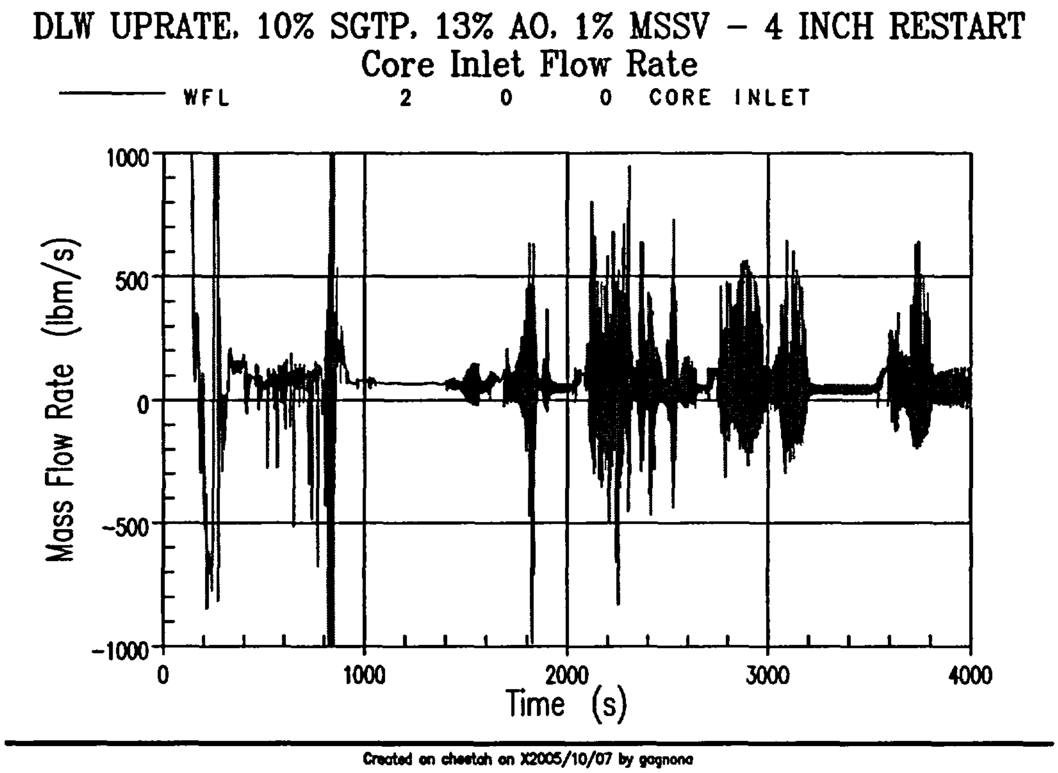

Figure E.13-16 BVPS-1 4-Inch Break, Core Inlet Flow

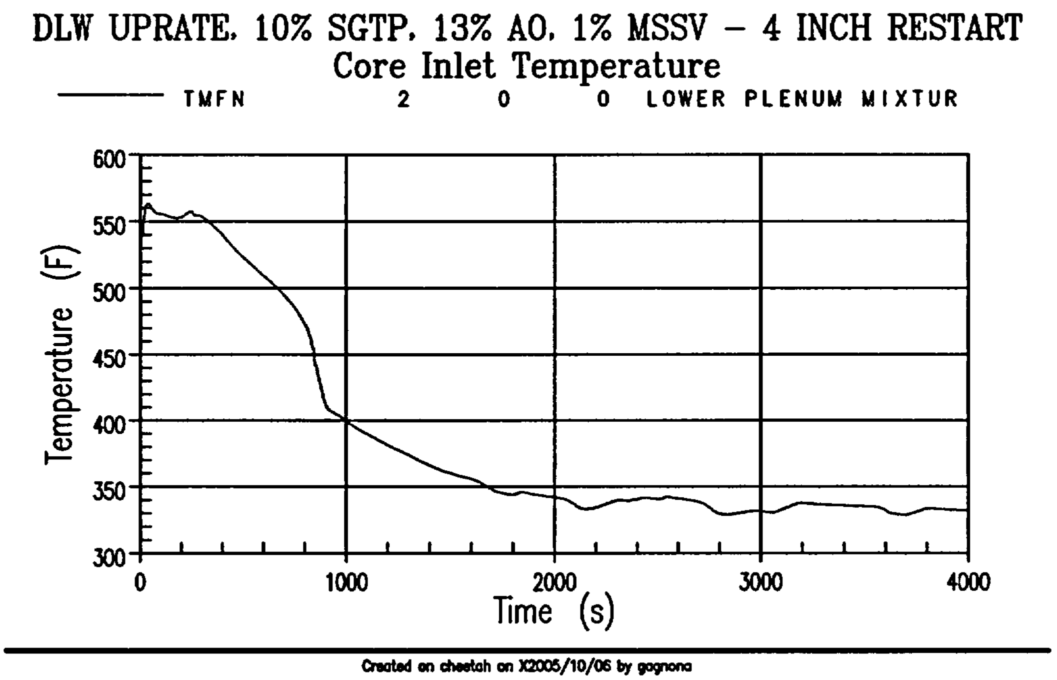

Figure E.13-17 BVPS-1 4-Inch Break, Core Inlet Temperature

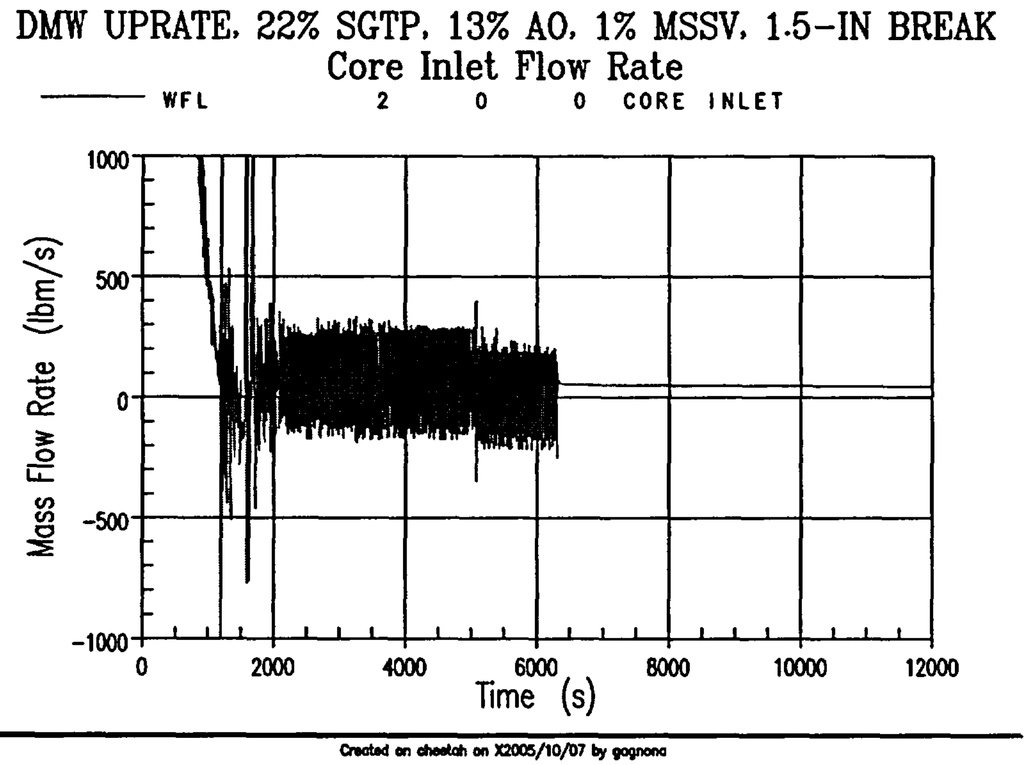

Figure E.13-18 BVPS-2 1.5-Inch Break, Core Inlet Flow

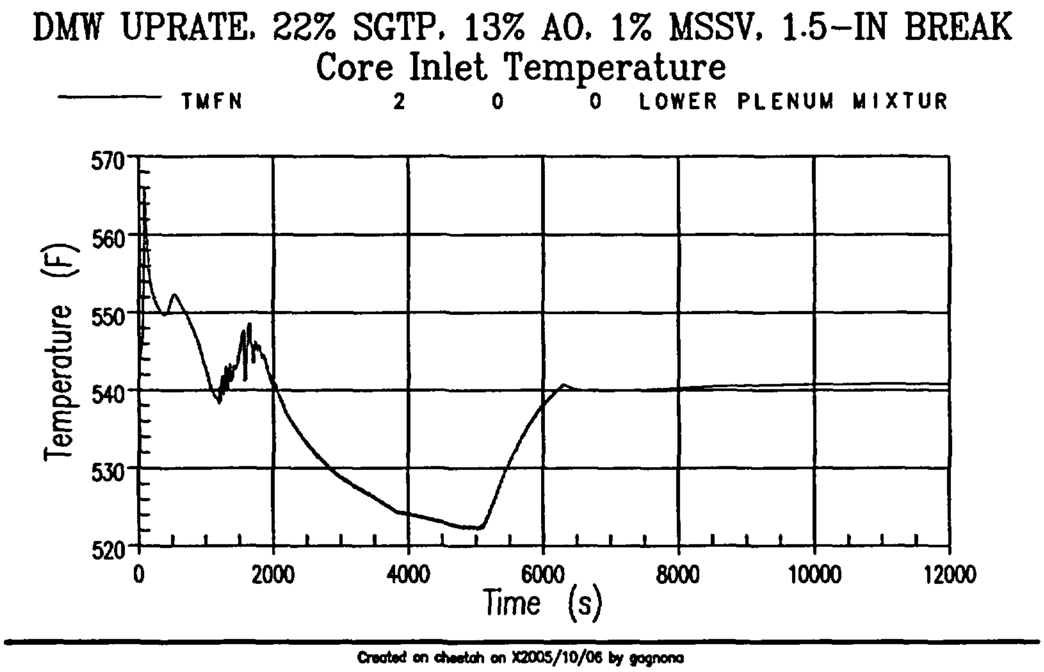

Figure E.13-19 BVPS-2 1.5-Inch Break, Core Inlet Temperature

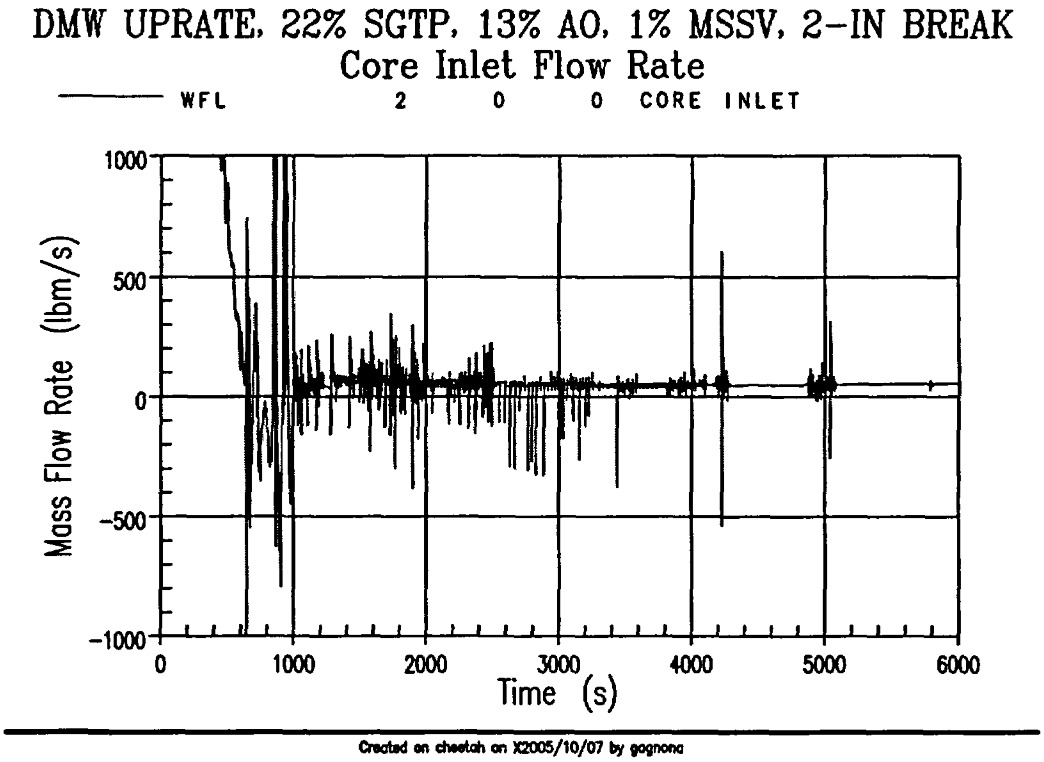

Figure E.13-20 BVPS-2 2-Inch Break, Core Inlet Flow

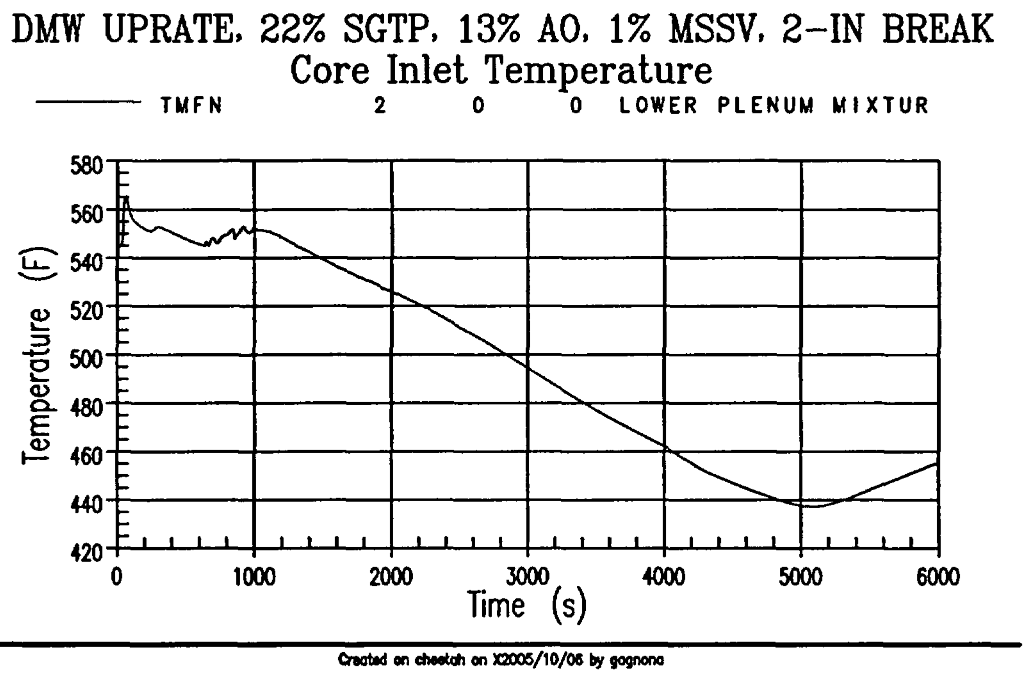

Figure E.13-21 BVPS-2 2-Inch Break, Core Inlet Temperature

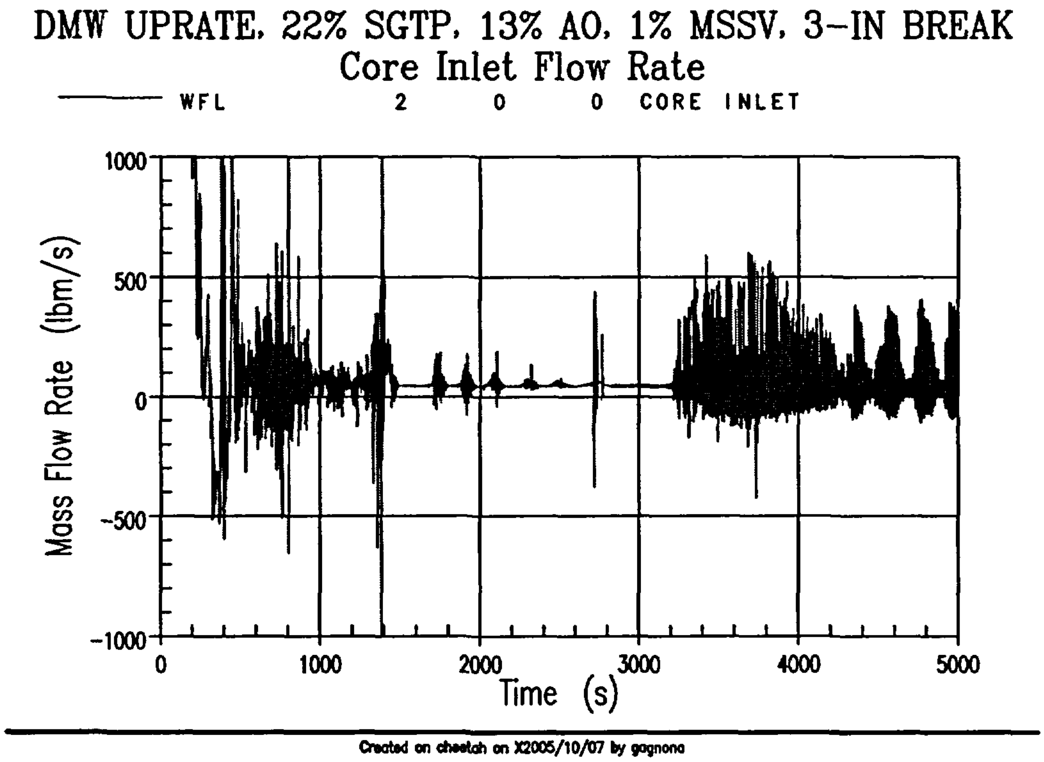

Figure E..13-22 BVPS-2 3-Inch Break, Core Inlet Flow

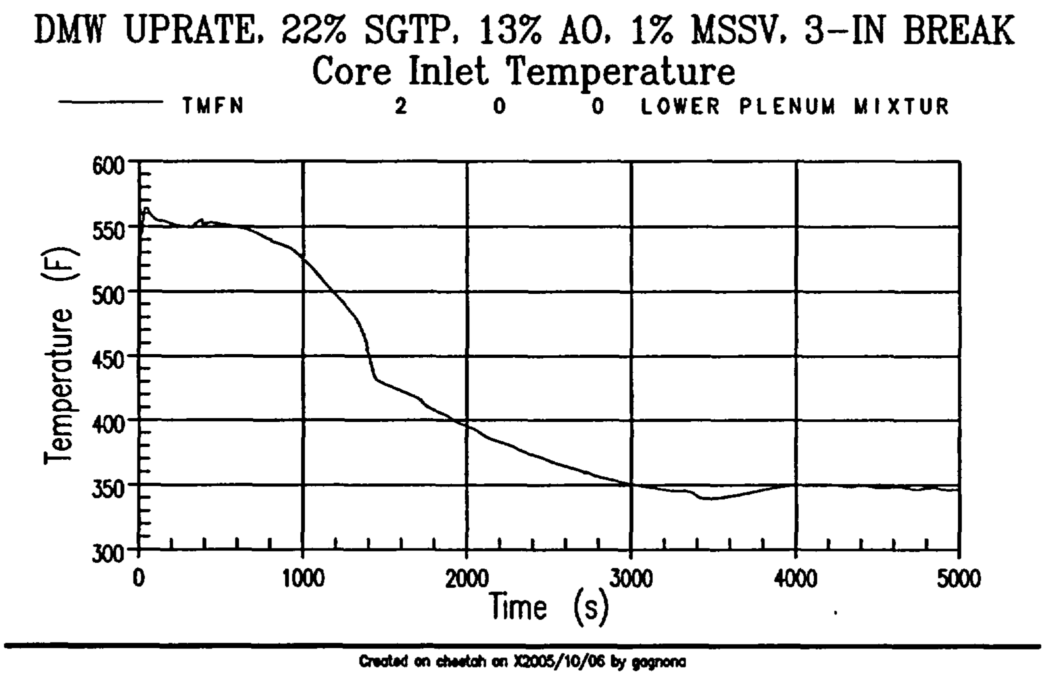

Figure E.13-23 BVPS-2 3-Inch Break, Core Inlet Temperature

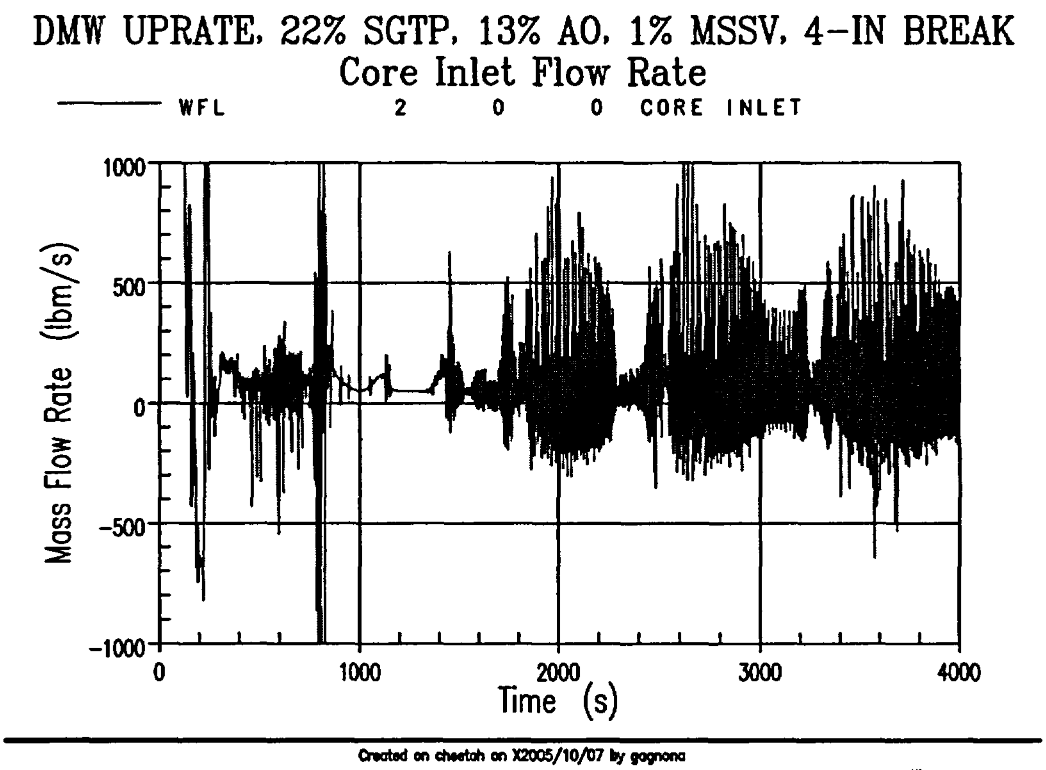

Figure E.13-24 BVPS-2 4-Inch Break, Core Inlet Flow

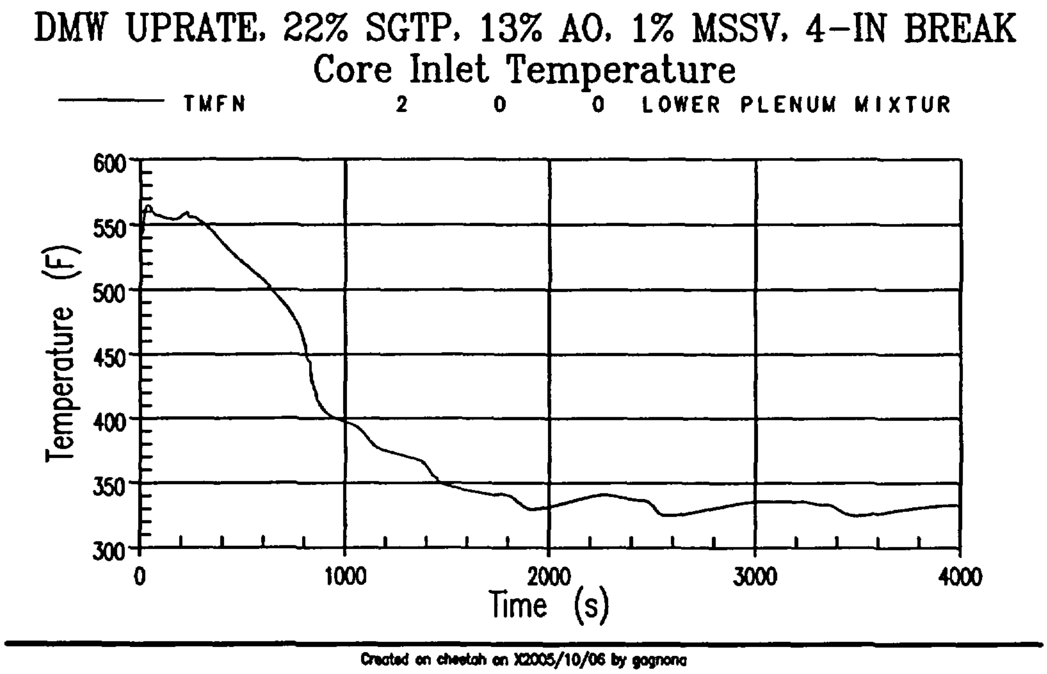

Figure E.13-25 BVPS-2 4-Inch Break, Core Inlet Temperature

Question E.15 (Applicable to RSG & EPU)

No hot assembly is shown in Fig. E .15-1. Please explain how the sink temperature is computed in the hot channel for PCT determinations. How is the core bypass modeled? Also, with one cell in the uphill side of the pump suction leg, explain how loop seal clearing occurs as predicted by the NOTRUMP code.

Response:

[ ] a,c

Loop seal clearing is explained in the response to Question E.27. Note that the simplified loop seal model used in NOTRUMP was demonstrated to be conservative. The NRC agreed with this conclusion as discussed on Page 41 of the NOTRUMP SER, Reference E.15-1.

Reference:

E.15-1 Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10054(P), Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," May 21, 1985

Question E.16 (Applicable to RSG & EPU)

Please explain why there is condensation in the broken loop. It is the NRC staff's understanding that no ECC water can accumulate in the broken loop since the critical flow rate for these breaks is far greater than the injection rate. If the break is on the bottom of the discharge leg below the injection point, all of the injection will pass immediately out the break. Condensation by the flow stream is expected to be small since highly superheated steam should be entering the break through the broken loop. Please explain how the condensation is calculated in the intact and broken loops.

Response:

While no significant amount of water can accumulate in the broken loop since critical flow rates are such that broken loop ECCS injection inventory is lost, the nature of the COSI condensation model is such that the majority of the condensation occurs in the vicinity of the injection cone (References E.16- 1 and E.16-2). As such, it is not necessary to store fluid in the cold leg to achieve steam condensation from the Safety Injection (SI) flow stream even in the broken loop.

It is noted that even though superheating of the steam may be occurring as a result of core uncovery, the steam generators provide a significant source of de-superheating.

The NOTRUMP Evaluation Model (EM) cold leg model, including the effects of SI condensation, are discussed in Reference E.16-2. Specifically Sections 3.2.5 and 3.3.1 describe the cold leg and non-equilibrium safety injection model. [

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References

E.16-1 WCAP-10054-P-A Addendum 2 Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," C. M. Thompson, et. al., July 1997.

E.16-2 WCAP-10054-P-A "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," N. Lee, et. al., August 1985.

E.16-3 WCAP-11767, "COSI SI/STEAM Condensation Experiment Analysis," D. J. Shimeck, March 1988.

Question E.19 (Applicable to RSG & EPU)

Please explain the meaning of the statement: "The EPU analysis was performed with the latest version of NOTRUMP EM and, thus, all 10 CFR 50.46 changes that were to be implemented on a forward-fit basis are included in the analyses." Were additional changes, beyond those described in the response to this question, made to the approved method included in the analyses? If so, please describe them and provide an analysis to justify the changes.

Please explain how the new ECCS delivery curves were generated. Are these curves based on surveillance test data and do they take into account the two minimum-measured injection delivery rates adjusted for instrument error? Was an analysis of NPSH performed to show adequate margins at the higher injection flow rates? Please explain.

Please explain why a reduced hot assembly average power peaking factor is conservative. Use of the appropriate/higher hot assembly power should increase the hot bundle steam temperatures during uncovery, which should increase the hot rod/hot channel sink temperatures and the PCT. Please justify the use of the lower peaking factor.

Please refer to the concerns with the loop seal model as described in E.7 above.

Response:

As discussed at the meeting with the NRC staff on October 13, 2005 the EPU analysis was performed with Version 39 of NOTRUMP. The statement is clarifying that no changes to the NOTRUMP EM have been made to the code other than those for error corrections which have been reported through the 10 CFR 50.46 process. The only change not reported per the 50.46 process was the implementation of the COSI condensation model which was approved by the NRC in 1996. No other changes were made to the approved EM to execute the analysis.

The ECCS delivery curves are generated using a network flow analysis based on the performance requirements for the HHSI pumps and the system characteristics as specified in the system throttling and balancing procedure. The analysis develops the maximum system flow allowed during low pressure conditions to prevent pump run out conditions by using the maximum pump performance. The analysis accounts for potential increases in operating speed due to emergency diesel generator frequency variations and uncertainties associated with the flow and pressure instrumentation used during system testing. The minimum injection flow is then calculated based on biasing of these uncertainties in the conservative direction for minimum flow, conservatively accounting for injection path balancing as described in the response to Question E.2, and assuming minimum allowable pump performance. The minimum available Net Positive Suction Head (NPSH) under the increased flow conditions was analyzed and the results show that the required NPSH will be met under all flow conditions.

The reference to the hot assembly peaking factor was citing differences between the current analysis and the EPU analysis. The EPU analysis does use a lower hot assembly peaking factor, however, EPU core designs will have to abide by this and accommodate this new limit.

Question E.20 (Applicable to RSG & EPU)

The current SBLOCA analyses for BVPS-1 and 2 fail to justify identification of the worst small-break case. Since the integer break spectrum is too coarse, analysis of break sizes between the integer sizes could produce PCTs much higher than those reported in the recent analysis. Please demonstrate that the worst SBLOCA has been identified in the current pre-EPU analyses. Please also note that for any changes made to the SBLOCA analyses that result in PCT increases of more than 50°F, a re-analysis of the break spectrum and/or limiting break must be reported to the staff for review.

Response:

As a result of identification of this issue during the RAI process for EPU, a condition report was entered into the FENOC corrective action process. This condition report identified the potential for higher PCTs resulting from consideration of non-integer break sizes in SBLOCA analyses. Based on the concerns identified, FENOC management requested that a Basis for Continued Operation (BCO) be prepared in accordance with Generic Letter 91-18. In order to provide a basis for continued operation, Westinghouse was requested to perform an analysis for the current plant conditions, which considered non-integer break sizes and removal of credit for loop seal clearing in non-faulted loops. The analysis was performed in a manner which was bounding for both BVPS-1 and -2. [

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Two additional input parameters were also changed in this analysis to recover available operating margins. The first parameter was the average hot assembly power peaking factor. A value for this input parameter was derived by examining the core reload analyses and choosing a value which was lower than the current design limit but was bounding for the current operating cycles for both BVPS-1 and -2. The other input parameter modified was the minimum accumulator nitrogen gas cover pressure. In the current analysis of record, a minimum value more conservative than the current Technical Specification limits had been used in anticipation of changes associated with EPU conditions. A new value consistent with the current Technical Specification limits and bounding for both BVPS-1 and -2 was used in the revised analysis. All other input parameters were consistent with the bounding value for either BVPS-1 or -2 for this new analysis.

The results of this analysis showed that the 10 CFR 50.46 PCT and oxidation limits continued to be met. The PCT results showed a maximum value of 1902 °F. This represents a change of greater than 50 °F from the current analysis results of 1849 °F for BVPS-1 and 2105 °F for BVPS-2. The BVPS-1 results show a higher PCT due to the use of more conservative BVPS-2 ECCS flows in the bounding analysis. The results of this analysis were incorporated into the BCO, which was reviewed and approved by station management.

Additional corrective actions were initiated to address reportability under both 10 CFR 50.72 and 10CFR 50.46. The issue was determined to not be reportable under 10 CFR 50.72 following a review of the previous three cycles which determined that the new analysis remained bounding based on past reload designs. The issue has been determined to be reportable under 10 CFR 50.46 based on the change in PCT exceeding the 50 °F threshold. Since FENOC has already committed to re-analysis of SBLOCA as part of the EPU and RSG programs, reaffirmation of these commitments is made in lieu of submittal of the revised analysis performed to support the Basis for Continued Operation (BCO).

Question E.24 (Applicable to RSG & EPU)

Please show that the boric acid concentration does not exceed 30 wt% for all small breaks where boiling occurs for extended periods of time. Also, steam exiting the two-phase surface in the core contains boric acid which can plate out on the spacer grids and the fuel alignment plate or any structures in the upper plenum, thus increasing the resistance to steam exiting the core. Please justify that boric acid plate-out does not increase the steam resistance in the loop and depress the two-phase level into the core. During condensation of the primary steam, can plate-out of boric acid block the steam generator tubes or increase the tube resistance? What happens to the boric acid after the steam condenses on the tube walls or any other structures in the loop? Please explain.

Response:

Small break LOCA scenarios were discussed in the response to Question F.6, and the core region boric acid concentration for a 120 psia small break LOCA scenario are presented in Figures F-6 and F-7. As seen in Figures F-6 and F-7, the boric acid concentration does not exceed 30 wt% for all small breaks even if cooldown is not considered. In reality, small break LOCAs would create significant less core boiloff than calculated since procedures to start reactor cooldown would be expected to occur within 1 hour after the accident. Such cooldown actions, if modeled, would introduce considerable margin in the calculation.

There are no known industry studies of the effect of boric acid plating out on spacer grids, fuel alignment plate, or any structures in the upper plenum prior to the boric acid solubility limit being reached. Once residual heat is removed from the non-heat-source structures (such as spacer grids and fuel nozzles), there would be insufficient surface boil-off to create the large amounts of deposition prior to hot leg switchover time. Concerning the possibility of plating out boric acid on the fuel rods, any plating on the fuel rods prior to fuel quench would likely be a boron compound other than boric acid since the melting point of boric acid is 340°F. After core quench, boron compounds that come out of solution might be expected to return to solution quickly since locations within mixture would be exposed to a dilute core region solution. Boron solutes that return to solution and flow back into the vessel would be consistent with the boric acid assumptions used in the calculations (i.e. boric acid contained in core boil-off remains in the core region).

Boric acid plate-out in the SG tubes could occur only during the period where there is significant liquid entrainment around the loop and only when the SGs act as a heat source. Under these conditions, the rate at which boric acid accumulates in the core would be greatly reduced for two reasons; the liquid entrainment passing through the SGs that is not vaporized would remove boric acid from the core region, and the boric acid plate-out in on the tubes would be removing boric acid from the core region. Once the SGs act as a heat sink, condensation on the inside of the tubes would return the boric acid to solution. Similarly, boric acid plate-out on other structures would return to solution when the residual heat in the structure is removed and the plated surfaces are exposed to a low-quality 2-phase mixture or liquid dilute solution. In response to this question, a review of the BVPS-2 boric acid concentration cases were made in order to estimate the volume of boric acid that might be deposited in the SGs during the period entrainment around the loop would be significant. For the case presented in Figure F-2, a total of about 4000 lbm of boric acid was left behind in the core as the result of boiloff. If it is assumed that 5% of that boric acid is entrained around the loops and left in the steam generators, the total boric acid per steam generator is 4000 x 0.05 / 3 = 67 lbm. Assuming a density of boric acid of 50-100 lbm/ft^3, the resulting volume per steam generator is less that 1 cubic foot. For BVPS-1, if the 1 cubic foot of boric acid was deposited over a 10 ft length of steam generator tubing, the thickness would be 1 ft^3 / (3592 tubes x 0.775 in ID / 12 in/ft x π) x 10 ft = 0.0016 inches thick. This is not a sufficient deposition to cause a significant increase in loop resistance.

It is noted that the mechanisms for potential boric acid plate-out and subsequent dissolution are not directly dependent on reactor power or SG performance and therefore can be characterized as generic and not directly related to the RSG or EPU submittals.

Question E.25 (Applicable to RSG & EPU)

Can a failure in the ability to cool the RWST affect the SBLOCA or LBLOCA analyses? What is the worst single failure associated with the RWST cooling system? Please explain.

Response:

A failure in the ability to cool the Refueling Water Storage Tank (RWST) can not affect the SBLOCA or LBLOCA analysis. Technical Specification 4.1.2.8.b requires that the RWST temperature be maintained within the analytical temperature range. Presently, the upper limit is 55°F for BVPS-1 and 50°F for BVPS-2. The EPU LAR proposes to raise the upper limit temperature limit to 65°F for both Units. The cooling system for the RWST is used to maintain the RWST such that the analysis initial conditions are satisfied. If the cooling system were to fail, and the allowable temperature limits could not be maintained within the Technical Specification limits, the plant must shut down within 6 hours.

Question E.27 (Applicable to RSG & EPU)

Plant short-term analyses (in the first 3 hrs) show that with a break placed on the top of the discharge leg, the initial core uncovery can be increased as the steam pressure builds in the upper plenum forcing vapor into the suction leg. With the deep loop seals, and the break on the top of the pipe, the loop seals do not clear of liquid and during the first 2 hrs the clad temperatures can become very high and then remain above 1500°F. Preliminary NRC staff's calculations show that the PCT exceeds 2200°F during the initial loop seal blowout for the 0.04 ft^2 current licensing basis (CLB) and remains above 1500°F until about 3 hrs into the event. Please show that breaks on the top of the pipe are not more limiting than those on the bottom of the pipe using the NRC-approved version of NOTRUMP for the SBLOCA EM.

Plant-specific analyses show that small breaks can uncover during the long term (2-3 hrs) due to loop seal refilling. If plant cooldowns are initiated no later than 1 hr post-LOCA, the effects of loop seal refilling for small breaks may be averted. Please describe the EOP instructions and timing for initiating a plant cooldown following an SBLOCA. What is the latest time post-LOCA the operators are instructed to initiate a cooldown to shutdown cooling?

For large breaks, plants with deep loop seals can uncover during the long term. While temperatures are not expected to exceed 10 CFR 50.46 limits, oxidation limits can become excessive with a slowly increasing core two-phase level with time. Since large breaks on the top of the discharge leg (large enough so that the ECC cannot refill the RCS and re-establish single phase natural circulation) can produce a long-term uncovery of the top of the core, operator actions such as those mentioned to cool down the RCS will be ineffective for large breaks since the system is already depressurized. Please explain how the post-LOCA recovery guidance addresses large breaks with long-term uncovery. Also, how do the uncertainties related to break type in best-estimate LOCA (BELOCA) analyses, as mentioned in the response, cover loop seal refilling and the attendant thermal hydraulic effects? Please explain.

Response:

In order to demonstrate the effects of break orientation with the NOTRUMP Evaluation Model (EM), the revised limiting PCT transient for BVPS-1 EPU analysis was re-performed with the break oriented at the top of the RCS cold leg. Comparison plots between those obtained with the traditional [ ]^{a,c} break orientation and that of the top break orientation can be observed in Figures E.27-1 through E.27-3. As can be seen, the response observed is as previously described in the NOTRUMP documentation, most notably Reference E.27-1 which states that the [ ]^{a,c} break orientation is more limiting than observed for a top break orientation.

The general phenomena associated with the initial loop seal clearing is unaffected by the break location in the NOTRUMP Evaluation Model (EM). Since the NOTRUMP EM is comprised of a series of stratified fluid nodes with inter-connecting flow link, the initial loop seal clearing process is controlled by the formation of a stratified level in the broken RCS cold leg. The NOTRUMP code does not allow the formation of a stratified level until the region reaches saturation conditions. As such, even if vapor is flowing through the loop seal piping, it can not readily vent out the break and allow for pressure equalization between the vessel downcomer and core/upper plenum region until stratification occurs in the broken loop cold leg. In other words, the liquid trapped in the RCP suction cross-over leg uphill side must saturate and void before vapor flow can be established through this piping. Physical mechanisms such as entrainment and evaporation can also play a role in this. Obviously the larger the break size, the more significant these become. Another possible method of core/upper plenum pressure equalization with the downcomer is via the stratification of the RCS downcomer region through the upper head spray nozzle flow path. The upper head cooling spray nozzles are a design bypass flow path between the vessel downcomer and upper head fluid region. Vapor venting through this flow path typically occurs following the drainage of the upper head region. While this flow path may not be sufficient by itself to relieve core steam generation, it does provide a means of relieving core steam pressure once the head has drained. Depending on the break location, orientation and transient time, this may result in a fraction of the core steam being bypassed to the break.

Small breaks may indeed undergo brief periods of uncovery as a result of loop seal re-plugging as a result of RCS system refill; however, this is highly dependent on the break orientation assumed as well as ECCS performance characteristics utilized in the analysis. For a typical NOTRUMP EM Appendix-K analysis with the break oriented on the [ ]^{a,c} of the RCS cold leg, ECCS backflow into the loop seal piping can not occur until a significant refill of the RCS occurs due to the nature of the critical flow models utilized to simulate the break. That being essentially all ECCS flow introduced to the broken loop RCS piping is discharged out the break. As such, backflow due to broken loop ECCS injection can only be expected to occur for breaks oriented at locations other than the [ ]^{a,c} of the RCS cold leg. As seen in the figures associated with the top break orientation, loop seal re-plugging was indeed observed (Figure E.27-3) although the predicted uncovery associated with this type of phenomena was tenuous (Figure E.27-2) and the core exit vapor temperatures observed were much less severe than predicted with a [ ]^{a,c} break orientation (Figure E.27-1). The time at which this loop seal plugging/clearing behavior occurs after 1 hour, or the time after which operator action to cooldown and depressurize the plant would have occurred. This action will minimize this observed behavior. See response to Question E.9 for additional details regarding the effects of RCS depressurization on SBLOCA transient behavior.

The issue described with respect to large break LOCA recovery and potential loop seal re-plugging is not specific to the BVPS units and should be addressed outside of the scope of the EPU. Once again, only large breaks which are oriented at the top of the RCS cold leg piping would be susceptible to loop seal re-plugging since a significant amount of water build-up in the cold leg piping is required prior to water (either from the broken loop or intact loop ECCS) being able to backflow into the loop seal piping. In a simplified sense, the RCS is acting as a manometer at this point. However, in order to establish this in a long term sense, a delicate set of conditions with regard to vent paths in the RCS which rely on lower steaming rates would need to occur. In the short term (i.e. prior to hot leg switch over) this would be extremely unlikely because of the high boil-off rates resultant from near term decay heat. As such the loop seal plugging and purging would be an ongoing process that would not lead to any long term uncovery periods. This has been shown to be the case in facility tests such as ROSA.

Finally, the uncertainties in break type do not directly account for the loop seal re-plugging and it's attendant thermal-hydraulic effects as these scenarios are not simulated for a sufficient length of time to demonstrate this behavior. It is noted that the short term effects of break type are encompassed in the Best Estimate LOCA methodology utilized for the BVPS EPU analyses.

See response to Question E.9 for discussion on operator actions and plant cooldown.

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Figure E.27-1 BVPS-1 2.75-Inch Break, Core Exit Vapor Temperature Comparison

a,c

Figure E.27-2 BVPS-1 2.75-Inch Break, Core Mixture Level Comparison

a,c

Figure E.27-3 BVPS-1 2.75-Inch Break, Broken Loop Seal Vapor Flow Comparison

Reference:

E.27-1 WCAP-10054-P-A Addendum 2 Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

Question F.1 (Applicable to EPU)

Does the calculated mixture volume include the effect of the loop resistance? The loop resistance will depress the mixture volume in the inner vessel, especially in the first 1-2 hours when the decay heat steaming rate is high. Please show the effect of the loop resistance on the mixing volume and boric acid concentration vs time. SG heat addition will superheat the steam in the cold legs. Was this included in the loop resistance? Please explain. What is the effect of liquid in the loop seals on mixing volume and boric acid content? Does the analysis consider injection from the boric acid storage tank and how were they included in the analysis?

Response:

The boric acid concentration analysis uses a simplified vessel model, not a system response model. As such, it does not include loop pressure drop effects due to loop resistance, steam superheat, or loop seal liquid content. While the simplified vessel model does not include the loop resistance effects, the core region liquid volume used in the calculations was successfully benchmarked against the EPU licensing basis large break LOCA WCOBRA/TRAC transient used to demonstrate stable and sustained quench. This licensing basis large break LOCA transient includes the effect of SG heat addition and cold leg steam superheat. The benchmark comparison showed that the liquid mixing volume used in the boric acid concentration calculations was conservative (FENOC Letter L-05-112, F.1 RAI response Figures F.1-1 and F.1-2). As compared to the original equipment SGs, the replacement SGs will decrease loop hydraulic resistance which will improve core reflooding and increase available liquid mixing volume. The use of a simplified vessel model has been the current industry standard for boric acid precipitation calculations and this methodology supports the current licensing basis for nearly all US PWRs. As such, the use of a simplified vessel model in lieu of a system model can be characterized as generic and not directly related to the RSG or EPU submittals.

Increased loop resistance would have two effects on the calculations; the core region pressure would increase and the core mixture level would decrease. A higher core region pressure has two significant effects on the calculation; core voiding is less and core boil-off is increased. Reduced core voiding increases the liquid volume available for the boric acid and water solution. Core boil-off is increased because the heat of vaporization (h_{fg}) decreases as pressure increases if no subcooling is credited. Increased core boil-off increases the rate at which boric acid is being concentrated in the core region. In response to this question, additional calculations were performed for BVPS-2 to demonstrate the effect of higher core region pressures on the rate of boric acid buildup in the core. The results of the these additional calculations, summarized in Figure F-1, show that if higher core region pressures are considered, the effects of reduced voiding and lower h_{fg} approximately offset each other. Additional discussion of loop resistance is presented in the response to Question F.2.

Loop seal refilling would temporarily increase the loop pressure drop and would depress the mixture level in the core. The expected increase in loop pressure is discussed in the response to Question F.2. Loop seal refilling would be significant to the calculations only if [ ] a,c. Neither LOCA ECCS Evaluation Models (EMs) nor observations during the ROSA tests (discussed in Reference 1 in F.2 RAI response of FENOC Letter L-05-112) predict [ ] a,c. Cyclic loop seal refilling/clearing would [ ] a,c. Addition discussion on loop seal refilling/clearing is presented in the response to Question F.2.

For Westinghouse 3-loop PWR plant designs, the boric acid storage tank is part of the Chemical and Volume Control System (CVCS) and is not part of the ECCS.

Question F.2 (Applicable to EPU)

Please show the effect of the loop resistance on the mixing volume vs. time and the effect of loop seal refilling. The loop seal is not expected to remain clear for the 7-8 hour time period prior to the switch to simultaneous injection. Downcomer boiling will also reduce the size of the mixing volume. Please provide an analysis of the effect of downcomer boiling on the mixing volume vs. time.

Response:

System effects such as loop resistance versus time, downcomer boiling, and loop seal refilling are not part of the simplified vessel model. In response to this question, a study was performed on BVPS-2 to compare the expected loop pressure drop (from the top of the core region to the top of the downcomer) to the supportable loop pressure drop based on the difference between the hydrostatic heads of the downcomer and the calculated core collapsed liquid level. Figure F-2 shows downcomer elevation, the top-of-active fuel elevation, and the core collapsed liquid level that corresponds to the liquid mixing volume used in the boric acid concentration calculations. At 1 hour after a large break LOCA, it is assumed that the core would be quenched and the mixture level in the vessel would be at least to the top elevation of the core. The expected loop pressure drop can be calculated using a form of the Darcy formula by considering the steam flow in a single loop;

$$\text{Expected Loop } \Delta P = C * \nu * (\text{mdot} / 3)^2 * (7.48052)^2 * (60)^2 / 144 * g/g_c$$

where:

ΔP = pressure drop, psi
C = Hydraulic loss coefficient, ft/gpm^2
ν = specific volume, ft^3/lbm
mdot = core steaming rate = 50.51 lbm/sec
7.48052 = volume conversion, gal/ft^3
60 = time conversion, sec/min
144 = area conversion, in^2/ft^2
g/g_c = force/mass conversion, lbf/lbm

This formulation makes the simplifying assumptions that by 1 hour after the break, the entrainment flow is low and the steam generator secondary side has cooled to the point that the steam flow is not significantly superheated by the steam generator.

The value for mdot, 50.51 lbm/sec, is the calculated core steaming rate at 1 hour using BVPS-1 and BVPS-2 uprated power, 0.6% calorimetric uncertainty, Appendix K decay heat, saturated atmospheric pressure conditions, and no SI subcooling. The loop hydraulic loss coefficient is [ ] a,c (including RCP Locked Rotor) based on the loop loss coefficients supplied in F.10 RAI response (FENOC Letter L-05-112).

Expected Loop ΔP = [ ] a,c

This pressure drop is well within the loop pressure drop supported by the assumed mixing volume used in the calculations. For example, for BVPS-2 at 1 hour, the collapsed liquid level used in the boric acid calculations is approximately 6.60 feet and the downcomer hydrostatic head is 15.51 feet. Assuming saturated conditions in the vessel at atmospheric pressure, the available hydrostatic head supports a loop pressure drop calculated as follows;

$$\text{Supportable Loop } \Delta P = \Delta EL_{DC\text{-}CLL} * \rho / 144 = 3.70 \text{ psi}$$

where:

$\Delta EL_{DC\text{-}CLL}$ = 15.51 - 6.60 = 8.91 ft
ρ = density of liquid = 59.8 lbm/ft^3 (assume saturated conditions at atmospheric conditions)
144 = area conversion, in^2/ft^2

This compares favorably to the expected loop pressure drop of 0.69 psi. In reality, downcomer and vessel density differences would affect the hydrostatic balance exercise. For saturated liquid conditions, the liquid in the vessel would be denser due to the higher concentration of boric acid. From Figure F.1-10 from F.1 RAI response (FENOC Letter L-05-112,), the density increase of a boric acid and water solution (near the solubility limit) can be calculated as follows;

$\Delta\rho$ = (29.27 - 1.00) wt% * 0.0034 $\Delta\rho/\Delta$wt% = 9.6%

Assuming the full density difference, the Supportable Loop ΔP would be;

Supportable Loop ΔP = 3.70 psi / 1.096 = 3.38 psi

Boiling in the downcomer would reduce the hydrostatic head in the downcomer and therefore would reduce the core region mixture level. Boiling in the downcomer as a result of stored energy release in the vessel metal mass would be minimal later in the transient when boric acid precipitation is a concern. If the downcomer were to be 5% voided when the boric acid concentration in the core region becomes significant, the effect on the hydrostatic balance would be;

Supportable Loop ΔP = 3.70 psi * 0.95 = 3.52 psi

A comparison of the expected loop pressure drop (ignoring loop seal refilling) to the loop pressure drop supported by the assumed mixing volume shows that the assumed mixing volume to be conservative. Effects due to downcomer/vessel density differences or boiling in the downcomer are not large enough to affect this conclusion.

Concerning the effect of loop seal refilling, the expected increase in loop pressure drop can be estimated using the difference in elevation of the bottom of the cold leg to the top of the crossover pipe in the loop seal. For BVPS-1 and BVPS-2, this elevation difference is 7.87 feet. The resulting increase in loop pressure drop would be;

Loop Seal ΔP = ΔEL * ρ / 144 * g/g_c = 3.27 psi

where:
ΔEL = loop seal elevation = 7.87 ft
ρ = density of liquid = 59.8 lbm/ft^3 (assume saturated conditions at atmospheric conditions)
144 = area conversion, in^2/ft^2
g/g_c = force/mass conversion, lbf/lbm

Loop seal refilling could increase the loop pressure drop to a point where the mixture level in the core is depressed and perhaps to a point where the resulting collapsed liquid level may be less than that assumed in the calculations. As discussed in the response to Question F.1, it is unrealistic to expect complete loop seal closure to be sustained. Based on test/analysis experience, it is expected that cyclic loop seal refilling/clearing will occur. Cyclic loop seal refilling/clearing would promote mixing in the lower plenum and downcomer, a benefit in the calculations.

In summary, the simplified vessel model conservatively calculates the core region boric acid concentration even though system effects are not specifically modeled. Loop seal refilling/clearing would be cyclic and should it occur, it would promote mixing in the vessel. Downcomer boiling would have a slightly negative effect on the mixing volume assumptions, but not so much as to exceed the conservative mixing volume used in the calculations.

Question F.3 (Applicable to EPU)

The use of saturated temperature at the core inlet is not justified. Since the ECC pumped injection contains 65°F water, please show that the temperature of the water entering the core during the injection phase does not produce precipitation prior to the switch to hot-side injection. At 6.5 hrs, the switch to hot-side injection will flush high concentrate boric acid into the sump. Please show that there is no local precipitation when this switch occurs.

Response:

Even if the temperature of the injected SI were 65°F it is not realistic that the water entering the core region via the lower plenum would be at this temperature. In addition to the stored energy in the metal structures in the downcomer (vessel wall, barrel, thermal shields or neutron pads), there are two other mechanisms to heat the injected SI water as it travels to the core region. These mechanisms were recognized by the NRC staff several years ago in the Byron/Braidwood Uprate Safety Evaluation Report (SER). They are as follows;

(a) Heat transfer from the core and upper plenum regions into the downcomer water through the core-former structure or through the wall separating the upper downcomer from the upper plenum.

(b) Steam flowing in the path from the upper plenum to the break will heat incoming water. This would apply to water injected into either the hot legs or cold legs.

These mechanisms provide a means of heating incoming SI water making it highly unlikely that water in the lower plenum would be significantly below saturation temperature.

The response to F.4 RAI (FENOC Letter L-05-112) discussed the competing effects of SI; thermal cooling and boric acid solution dilution. The response (Question F.4 discusses the possibility of boric acid and thermal gradients and the adverse effects of diffusion (thermal or molecular) without density-driven convection mixing. If incoming SI at 65°F were to mixing thoroughly in mixing volume (i.e. no significant temperature or boric acid gradients) core boil-off would be reduced due to the sub-cooling effect. Figure F-3 shows the results of a comparative study of the core region boric acid concentration with and without a 65°F SI sub-cooling assumption. The comparative study used the BVPS-2 EPU licensing basis case as a basis. The only difference between the two cases in Figure F-3 is the effect of 65°F SI sub-cooling. The reduction in boric acid concentration with 65°F sub-cooling is about 3.2 wt%.

The response to the following RAI provides additional discussion regarding the potential for localized boric acid precipitation when hot leg injection is initialized either at the top of the core where the SI enters, or at the bottom of the core (where the displaced core volume flows to the lower plenum).

Question F.4 (Applicable to EPU)

The simple calculation ignores diffusion. With a subcooled region in the lower portion of the core, boric acid will diffuse into the colder region as shown by the test data once the density difference across the core increases sufficiently. Thus, at the switch time to simultaneous injection, please show that the boric acid concentration is below the precipitation limit at the minimum injection temperature or minimum temperature of the subcooled region in the bottom portion of the core.

Response:

The calculations presented in the previous RAI responses (FENOC Letter L-05-112) assumed effective mixing with no differentiation between different mixing mechanisms such as diffusion (thermal or molecular) and density-driven convection within the vessel. While it is possible to hypothesize thermal and boric acid concentration gradients where thermal and molecular diffusion compete with each other, it is unlikely that such temperature or boric acid gradients would exist. The MHI tests indicate substantial temperature gradients in the lower plenum only prior to boric acid concentration buildup and density-driven mixing. Mixing due to density-driven circulation cells would likely dominate over diffusion mixing. If boric acid diffusion is dominant in the vessel, the rate at which boric acid accumulates in the core would be reduced since a greater portion of the vessel liquid inventory would be available for the boric acid solute.

The precise temperature of the injected water (as it enters the core region) or the temperature of the water in the lower plenum (just below the core) during the post-LOCA recovery period is difficult to predict. Water injected into the hot legs would interact with the steam generated in the core.

Condensation in the hot legs would not only heat the injected water, but would reduce the core boil-off by returning condensed water to the core region. Water injected into the cold legs would be heated by the interaction with steam in the cold legs and by thermal diffusion and convection mixing in the downcomer. This heating of the injected water as it travels to the core region and the expected density-driven mixing mechanisms in the vessel would make it unlikely that significant temperature or boric acid gradients would exist. Even though in-depth questions concerning mixing in the vessel cannot be answered with the tools currently available, it is relevant to cite margin that realistically can be expected to exist. Realistic calculations performed to demonstrate margin are discussed in the response to Question F.5.

Question F.5 (Applicable to EPU)

It is understood that the hot leg switchover time is not related to the switch to recirculation. The timing for switch to recirculation affects the injection temperature, the fluid balance in the vessel and the potential for boric acid precipitation at the bottom of the core. Fig F.1-4 shows that the boron concentration at 6.5 hrs is about 28 wt%. This concentration requires the temperature in the bottom of the core to remain above approximately 212°F. Please provide the analysis to show that the temperature at the core inlet up to 6.5 hrs does not exceed 212°F.

Response:

This question concerns the possibility of colder temperatures in the lower core and the effect of these colder temperatures on potential boric acid precipitation. The specific concern is that the temperature of the lower core might be colder than 212°F. The simplified vessel model assumes that there is complete mixing within the mixing volume. The effects of temperature or boric acid concentration variations within the mixing volume can only be addressed by other conservatisms in the calculations. Many of these conservatisms were discussed in F.1 RAI response (FENOC Letter L-05-112), the most significant being as follows:

- Appendix K decay heat
- No mixing outside the mixing volume
- Worst case assumptions for sump boron/water sources
- No credit for early entrainment around the loop
- No credit for SI sub-cooling
- No credit for boron presence in steam

In response to this question and Questions F.3 and F.4, BVPS-1 and BVPS-2 boric acid concentration calculations were performed using realistic assumptions for decay heat, lower plenum mixing and containment sump boron/water sources (the first three items on the above list). The assumptions for the realistic case are as follows:

- 1979 ANS Decay Heat with no uncertainty
- Best estimate lower plenum volume (75%)
- Nominal volumes and boron concentrations for containment sump volume constituents

The results, shown in Figures F-4 and F-5, indicate that a realistic boric acid concentration at hot leg switchover time for both BVPS units would be 18.4 wt% and 18.0 wt% respectively. These realistic calculations indicate that boric acid precipitation would not occur even if the temperature of the lower core region was significantly lower than 212°F.

In summary, margin demonstrated by realistic calculations is available to offset concerns about specific vessel mixing behavior and/or system effects. While specific vessel mixing behavior may be affected by reactor power or SG design details, fundamental vessel mixing mechanisms would not change. Questions concerning fundamental vessel mixing mechanisms can be characterized as generic and not directly related to the RSG or EPU submittals.

Question F.6 (Applicable to EPU)

This response provides insufficient information to address the question (for an example of the type of information needed by the NRC staff, please see the Westinghouse-CE Topical Report CENPD-254). What specific operator actions are required to assure the boric acid is controlled or dispersed before a late/rapid depressurization using PORVs, for example? What is the minimum RCS pressure following opening of both PORVs between 2 and 7 hours and what is the RCS concentration and temperature of the subcooled water in the core bottom? What is the longest duration boiling time and maximum boric acid concentration calculated for SBLOCAs that would be allowed by the guidance and directions given to the operators in the EOPs? Could the concentration be high enough that a late depressurization with sprays or PORVs would cause precipitation?

Response:

CENPD-254-P-A has been reviewed and compared to the methodology used in re-analyses performed in response to the previous RAIs (FENOC Letter L-05-112). The comparison can be summarized as follows;

<u>Large Break LOCA</u>

For large break LOCA, the CENPD-254-P-A (as modified by the Waterford-3 EPU LAR RAI responses) methodology and the methodology used in the re-analyses performed in response to the previous RAIs (FENOC Letter L-05-112) are essentially the same. The differences include the decay heat assumptions and the method used for assigning a reduction in the mixing volume to address core voiding. Also, whereas the CENPD-254-P-A methodology calculates a core dilution rate transient for large break LOCA once simultaneous injection occurs, such core dilution rate transients were not provided nor calculated in the BVPS EPU LAR or the previous RAI responses (FENOC Letter L-05-112). However, in response to this question and Question F.8, BVPS EPU core dilution rate transients (once hot leg injection begins) were calculated. These dilution calculations are discussed in the response to Question F.8.

<u>Small Break LOCA</u>

For small break LOCA, CENPD-254-P-A methodology uses the CE CELDA code to produce a series of break cases to address decay heat removal, specifically the possible depletion of the condensate storage tank and resulting loss of emergency feed water. BVPS-1 and BVPS-2 have no limitation on auxiliary feed water capacity and therefore the decay heat aspect of CENPD-254-P-A is not relevant. However, the CENPD-254-P-A methodology also uses the CELDA break cases to calculate the boric acid concentration in the core for small break LOCAs. The calculated boric acid concentration is then compared to the boric acid solution solubility limit at the calculated RCS pressure throughout the transient. The EPU LAR, as well as the re-analyses discussed in the previous RAI responses (FENOC Letter L-05-112), was based on small break LOCA boric acid precipitation evaluations that assumed the following:

1) Small break LOCA core region boric acid concentration is conservatively represented by large break LOCA calculations for the period before reactor operators would initiate reactor cooldown and depressurization actions.
2) Once cooldown and depressurization procedures begin, boric acid precipitation is not a concern because;
   a) Core boiloff will be reduced due to steam generator heat removal and condensed steam returning to the core

b) Cooldown and depressurization actions will tend to increase core mixture level

c) Cooldown and depressurization actions will allow the SI flow to flush the core region before the boric acid solubility limit is reached

In regard to 1), it is anticipated that operators would begin reactor cooldown within 1 hour of the accident. This is consistent with the CENPD-254-P-A methodology. In regard to 2), a specific small break LOCA boric acid concentration was not provided nor calculated for the BVPS EPU LAR or the previous RAI responses (FENOC Letter L-05-112). However, in response to this question, additional calculations were performed to conservatively predict the core region boric acid concentration for a small break LOCA scenario. An RCS pressure of 120 psia was chosen since that represents a saturation temperature above which water and boric acid become miscible and the boric acid solubility is effectively 100%.

For each BVPS unit, a 120 psia RCS pressure boric acid buildup calculation was made. The assumptions used in these calculations are as follows:

- Constant 120 psia RCS pressure
- No SI subcooling (decay heat removed by h_{fg} at 120 psia)
- Mixing volume based on collapsed liquid level from NOTRUMP break case
- No credit for operator actions (SG cooldown)
- Core dilution calculated for SI flows at 120 psia backpressure

As listed above, the mixing volume for these small break calculations were taken from unit-specific NOTRUMP runs for a 6-inch break. Since these runs indicated an RCS pressure below 100 psia at 6 hours, the core and upper plenum collapsed liquid volume taken from these runs is conservative for deriving a liquid mixing volume. Mixture volume above the bottom of the hot leg was not included. The core region boric acid concentrations for these cases are shown in Figures F-6 and F-7. For these cases, the SI was assumed to be at the sump concentration since for this size break, sump recirculation prior to hot leg switchover would be expected to occur. Regardless, the difference between the boric acid concentration of the sump and the boric acid concentration of the RWST is small since the RWST is the dominant contributor to the sump liquid volume. Figures F-6 and F-7 can be compared to Figures F-8 and F-9. As expected, the benefit of the reduced core voiding and greater vessel liquid mass is offset by the lower heat of vaporization (h_{fg}) used to calculate boil-off. Boric acid concentration calculations for realistic small break LOCA scenarios would see lower net boil-off due to lower decay heat and SI sub-cooling. Heat removal by the SGs would create condensation that would flow back into the reactor vessel (reflux condensation), reducing net boil-off. Figures F-6 and F-7 show that for a small break LOCA scenario with no cooldown actions, the core region boric acid concentration is below the atmospheric solubility limit at hot leg switchover time. So long as the RCS is depressurized to 120 psia by hot leg switchover time, hot leg dilution flow will prevent boric acid precipitation in the core. The ability to depressurize smaller small break LOCAs is demonstrated in the response to the small break LOCA RAI E.9.

This question also requests information concerning the possibility of a late/rapid depressurization using the pressurizer PORVs. Late/rapid depressurization creates a potential for boric acid precipitation when the boric acid concentration in the core is above the solubility limit of post-depressurized RCS. In other words, depressurization could cause boric acid to precipitate out of solution if the resulting cooldown causes the temperature of the boric acid solution to pass through the solubility limit. This concern is relevant only to situations where the boric acid concentration is well above the atmospheric pressure solubility limit and where the RCS depressurizes to below 120 psia. Even at pressures as low as 50 psig, the boric acid solubility limit would be twice what it is as atmospheric conditions. It is not anticipated that reactor operators would depressurize the RCS to below 50 psig using the pressurizer PORVs. Regardless, in any depressurization scenario, boric acid precipitation will not occur if core dilution flow is established before the atmospheric pressure boric acid solubility limit is reached.

In summary, the large break LOCA boric acid analysis performed for the BVPS EPU submittal is consistent with the CENPD-254-P-A methodology (as modified by the Waterford-3 EPU LAR RAI responses). A small break LOCA boric acid analysis was performed to demonstrate that there is no potential for boric acid precipitation prior to realignment to hot leg recirculation. Once aligned for hot leg recirculation, the hot leg SI flow will provide sufficient core dilution flow. Rapid complete depressurization of the RCS prior to hot leg recirculation, while not expected to occur, would not cause boric acid precipitation since the boric concentration would be below the atmospheric solubility limit.

Question F.8 (Applicable to EPU)

What is the maximum containment pressure at 6.5 and 6 hrs? Is there sufficient flow in excess of the core boil-off rate to flush the core? Please show the boron concentration vs time for a minimum flushing flow of 5 gpm initiated at 6.5 hours?

Response:

The maximum containment pressure at hot leg switchover time for both BVPS units is 16.6 psia.

In the BVPS EPU LAR calculations and in subsequent re-analyses, checks were made to confirm sufficient SI flow to dilute the core after hot leg switchover. Specific core dilution plots were not provided. In response to this question, calculations were performed for both BVPS units to show the core dilution rate at hot leg switchover. The results are summarized in the Figures F-8 and F-9. The boric acid concentration and the core dilution were calculated assuming a RCS and SI backpressure of 14.7 psia. The difference in SI flow rates for a containment pressure of 16.6 psia would be insignificant. The reference to a flushing flow of 5 gpm is interpreted as 5 gpm above the flow needed to replace core boil-off. As shown in Figures F-8 and F-9, a flushing flow of 5 gpm is insufficient to promote core dilution until somewhat after hot leg switchover time.

Question F.9 (Applicable to EPU)

How much debris can accumulate in BVPS-1 and 2 during the 6.5-hr time period leading up to the switch to simultaneous injection? What is the effect of debris particles on the precipitation characteristics? Please provide the results of studies specific to BVPS-1 and 2 and show the impact on the boron concentration vs time.

Response:

Two separate effects of containment sump debris are addressed in this question. The first, the amount of calculated settled debris accumulation in the lower plenum, is a follow-up to the previous RAI response (FENOC Letter L-05-112) on this subject. That question response reported that [ ] a,c of the lower plenum volume is not credited in the calculation mixing volume and as such this volume would be available for debris accumulation. BVPS plant-specific calculations of sump debris generation/characterization, pass-through, and RCS debris ingestion are being performed to respond to NRC Generic Letter 2004-02. Once these sump debris calculations have been completed, the accumulation of debris in the lower plenum can be evaluated with respect to the hot leg switchover calculations performed for the EPU submittal. It is noted that the sump debris calculations are not directly dependent on reactor power or SG design details, and therefore this issue can be characterized as generic and not directly related to the RSG or EPU submittals.

The second effect of containment sump debris addressed in this question is the effect of sump debris on boric acid precipitation characteristics. While there are no known comprehensive industry studies of this effect, some relevant observations were made in the Fauske solubility tests discussed previously in the previous F.1 RAI response (FENOC Letter L-05-112). The Fauske test summary report indicated that powdered impurities introduced into a saturated boric acid water solution did not cause boric acid to precipitate out of solution.

The evaluation of sump debris effects (either debris accumulation or boric acid precipitation effects) is not directly dependent on reactor power or SG design details and therefore can be characterized as generic and not directly related to the RSG or EPU submittals.

General Question:

1. Does BVPS-1 and 2 have bottom-mounted instrument tubes (penetrating the RV lower head)? What is the tube diameter?

Response:

The BVPS-1 & 2 reactor vessel lower heads have bottom-mounted instrument tubes (BMI) that have an outer diameter of 1.5 inches. A description of the RPV lower head penetration inspection program for BVPS-1 & 2 was provided as part of the BVPS response to Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity" on September 19, 2003 (L-03-138).

Generic Concerns:
The NRC staff understands that the licensee is pursuing resolution of the 2 below listed questions with Westinghouse on a schedule to be determined. Please provide a schedule for the responses to these issues:

- E.12 Since the NOTRUMP code models the core as four cells, any subcooled water entering the core will be heated in the first large cell. As such, the tendency will be to over-swell the two-phase level. Four cells is insufficient to accurately predict the void distribution in a heated core, especially with a skewed power shape. Please show the sensitivity of the 3 inch break to 12 and 24 core nodes. RELAP5/MOD3 with 24 core nodes shows the PCT for the 3 inch break of about 2100 °F and a PCT greater than 2200 °F for the 0.04 ft^2 break. This may be due in part to the more detailed core nodalization and the fact that only the broken cold leg loop seal clears in the RELAP5 analysis. All other pertinent RELAP5 assumptions are consistent with the EPU conditions i.e. top peaked axial power distribution, HHSI injection curve, 1.2 x ANS 1971 decay heat standard, etc.

- F.9 More specific information and justification for the model and analysis is required. Some of the finer debris particles will permeate the core region and possibly adhere to the boric acid. Rising steam bubbles will mix the finer debris particles in the core and could cause them to concentrate.

Response (E.12):

On 11-07 thru 11-09-2005, the Staff attended an audit meeting at Westinghouse. During this meeting NOTRUMP level swell sensitivities were presented to the Staff which showed the model is insensitive to variations in core noding. In addition, several stand-alone cases were created from the Beaver Valley EM core model which again showed that core noding was not a sensitive aspect. In addition, boundary conditions were provided to the Staff which in turn were used to generate level swell simulations in an independent model that was developed by a Staff member. At that time, it was acknowledged that the predicted mixture level between the two models agreed almost exactly (9.75 feet). It was noted that the void fraction profile was different between the two models but the predicted mixture level was approximately the same.

Subsequent to this, an additional supplement was provided to Westinghouse on the details of the Staff void fraction calculations that are predicted by the independent model. In this provision, it was stated that further simulations performed by the Staff showed the models with regard to mixture level did not agree. The Staff's more recent calculations showed that mixture level was predicted to be 9 feet in lieu of the 9.75 feet predicted by Westinghouse. However, it appears that the Staff's calculations were based on a make-up flow rate of 75% of the boil-off rate. This is not consistent with the conditions that were provided originally to the Staff which was 80.6% (68.0 lbm/s). Westinghouse has re-run the simulation with the reduced flow rate (75%) and notes that the predicted mixture level is 9 feet.

It is noted that the demonstration cases used a flat power profile. However the core noding sensitivities performed in the Beaver Valley cases were top skewed. These cases showed that the power shape had no significant effect here either. Thus it is Westinghouse's position the mixture level calculations in NOTRUMP are appropriate and the model for Beaver Valley is adequate for demonstrating 10 CFR 50.46 compliance per Appendix K requirements.

Westinghouse is currently in the process of resolving the void profile issues and will forward the results to the Staff when this study is complete.

Response (F.9):

BVPS plant-specific calculations of sump debris generation/characterization, pass-through, and RCS debris ingestion are being performed to respond to NRC Generic Letter 2004-02.

The Westinghouse Owners Group (WOG) has recently approved a program that includes a review the post-LOCA boric acid calculation methodologies employed by the 3 major PWR safety analysis vendors. The objective of the program is to develop generic and plant-type specific boric acid precipitation analysis methodologies that would be submitted for NRC approval.

The initial phase of this program will characterize and rank phenomena that are important to adequately address the potential for post-LOCA boric acid precipitation. The ranking process would include generic items as the effect of debris particles on boric acid precipitation. The NRC has expressed a willingness to work with the WOG in identifying generic issues and formulating appropriate action plans. It is anticipated that the initial meeting with the NRC will be in January or February of 2006.

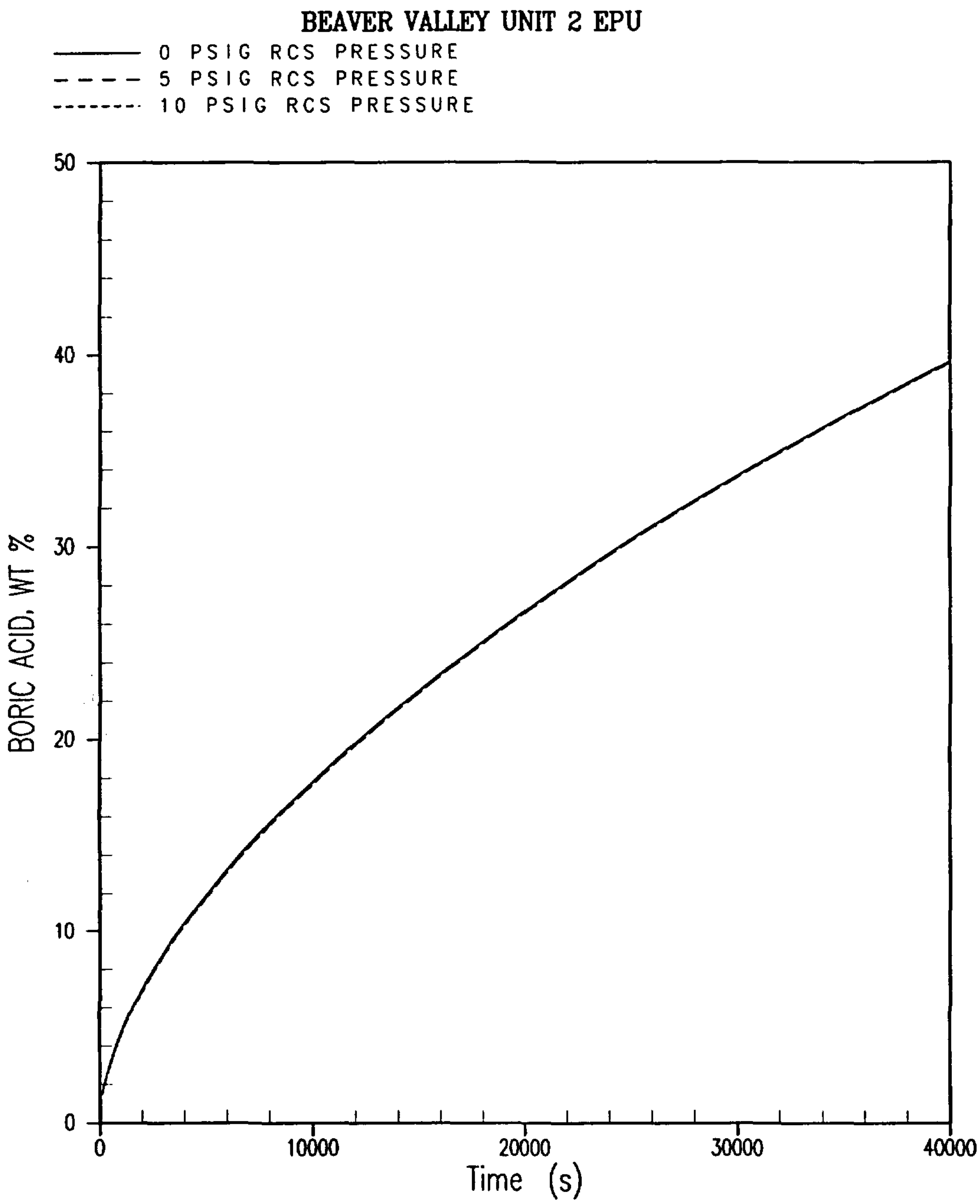

Figure F-1 - Effect of Assumed Pressure on Boric Acid Concentration (BVPS-2)

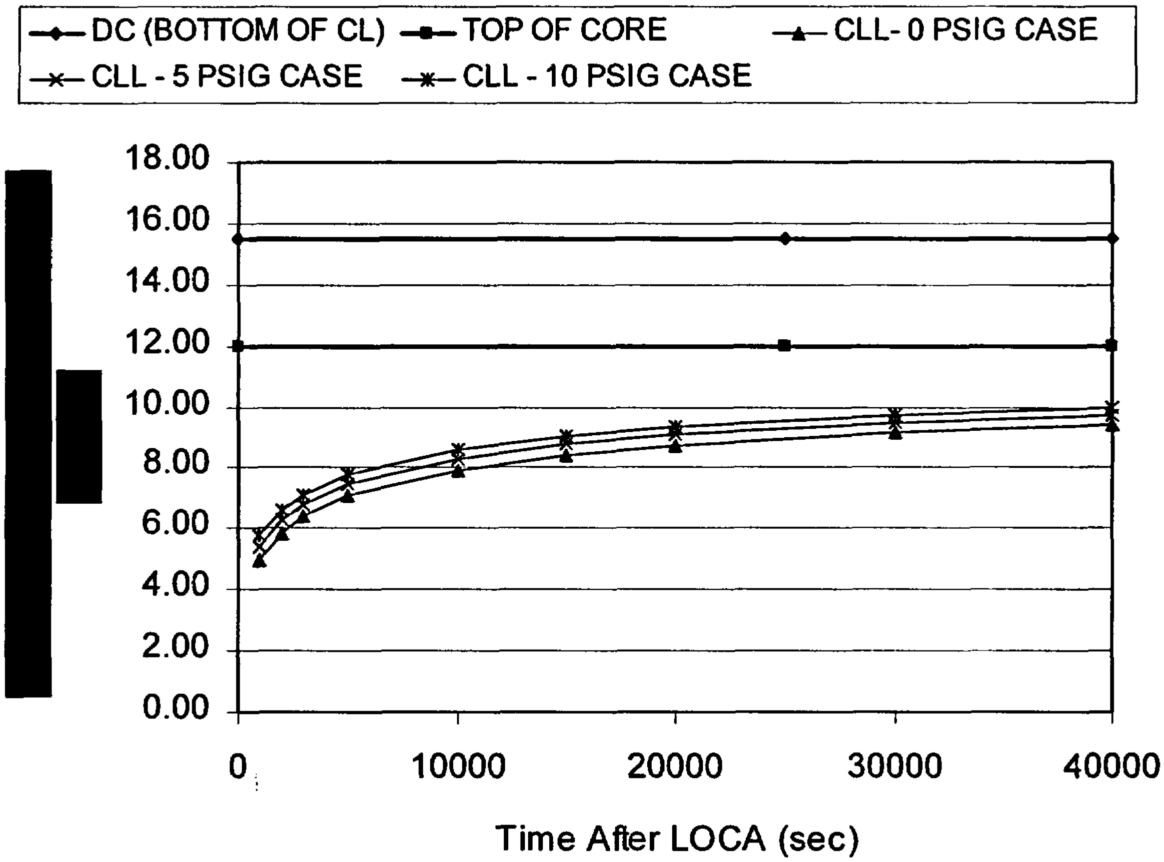

Figure F-2 - Comparison of Collapsed Liquid Level versus Downcomer Level (BVPS_2)

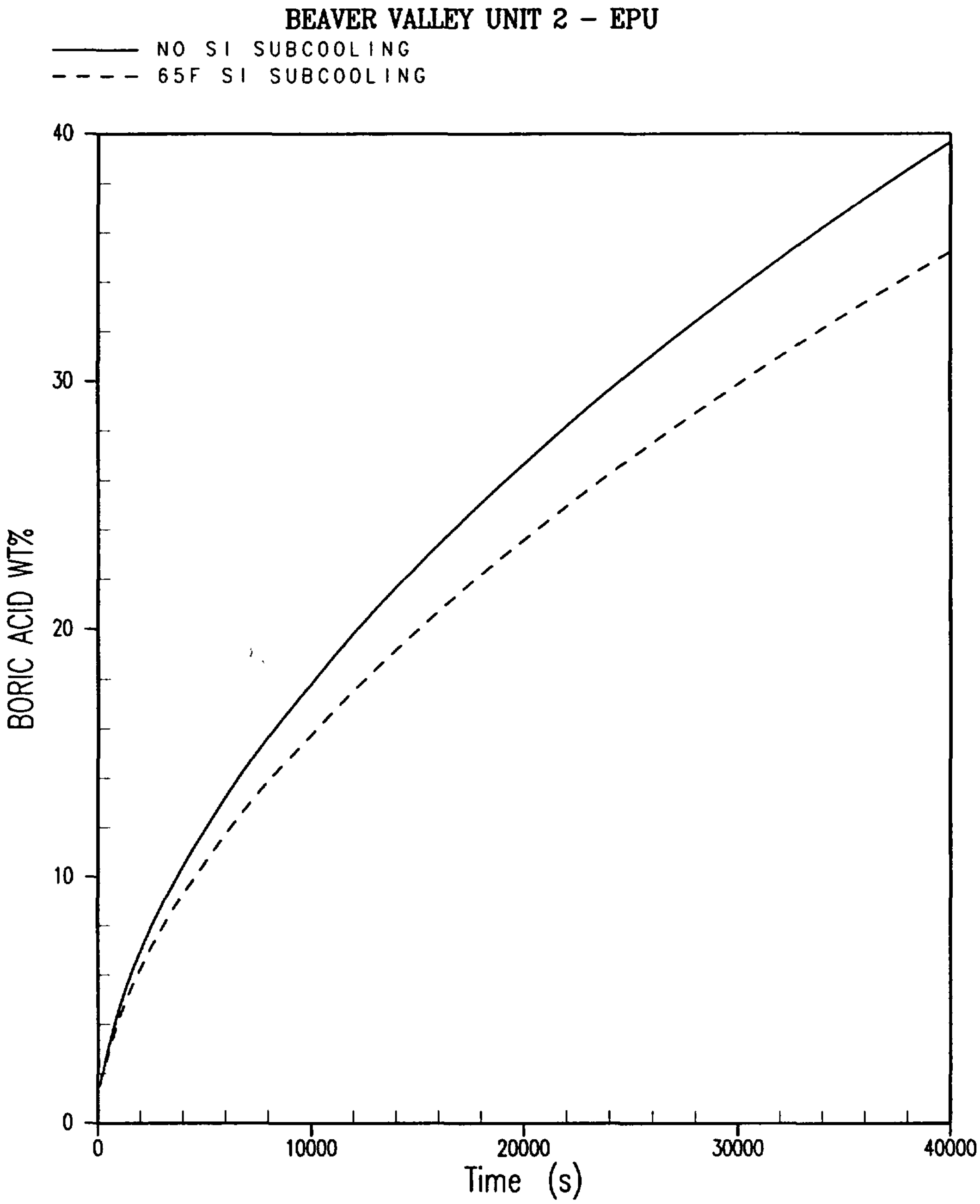

Figure F-3 - Effect of 65°F Subcooling on Boric Acid Concentration (BVPS-2)

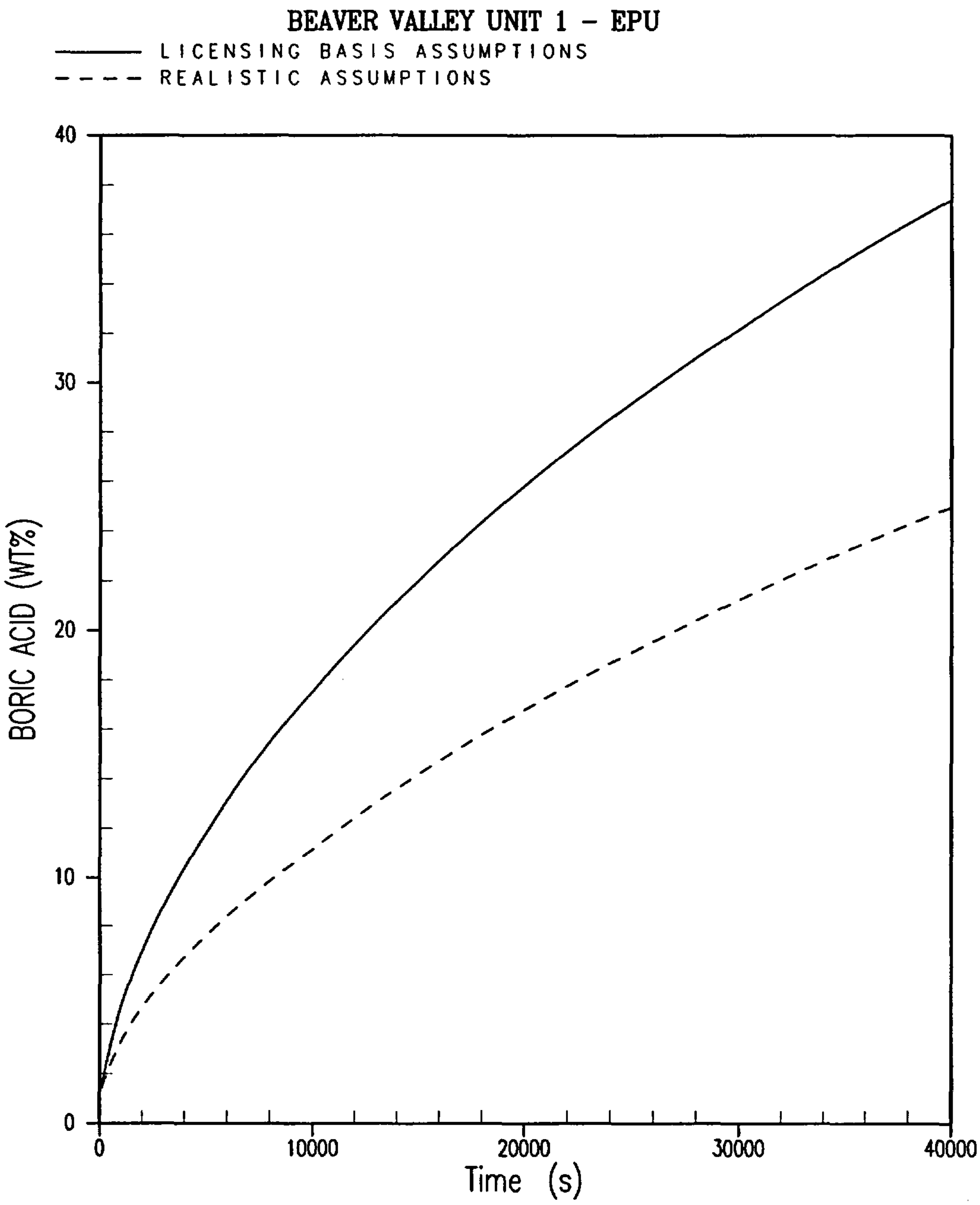

Figure F-4 - Realistic Calculation of Vessel Boric Acid Concentration (BVPS-1)

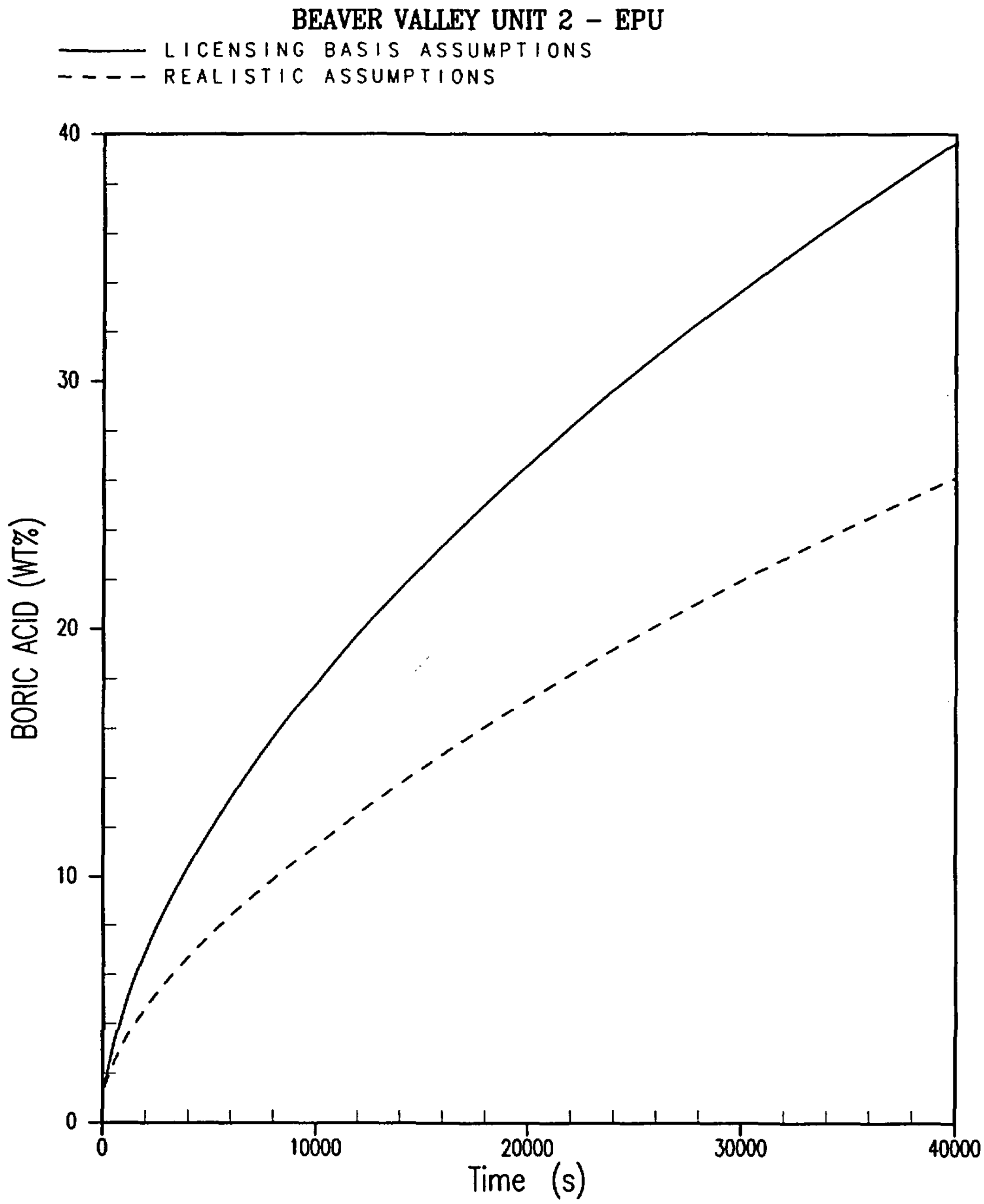

Figure F-5 - Realistic Calculation of Vessel Boric Acid Concentration (BVPS-2)

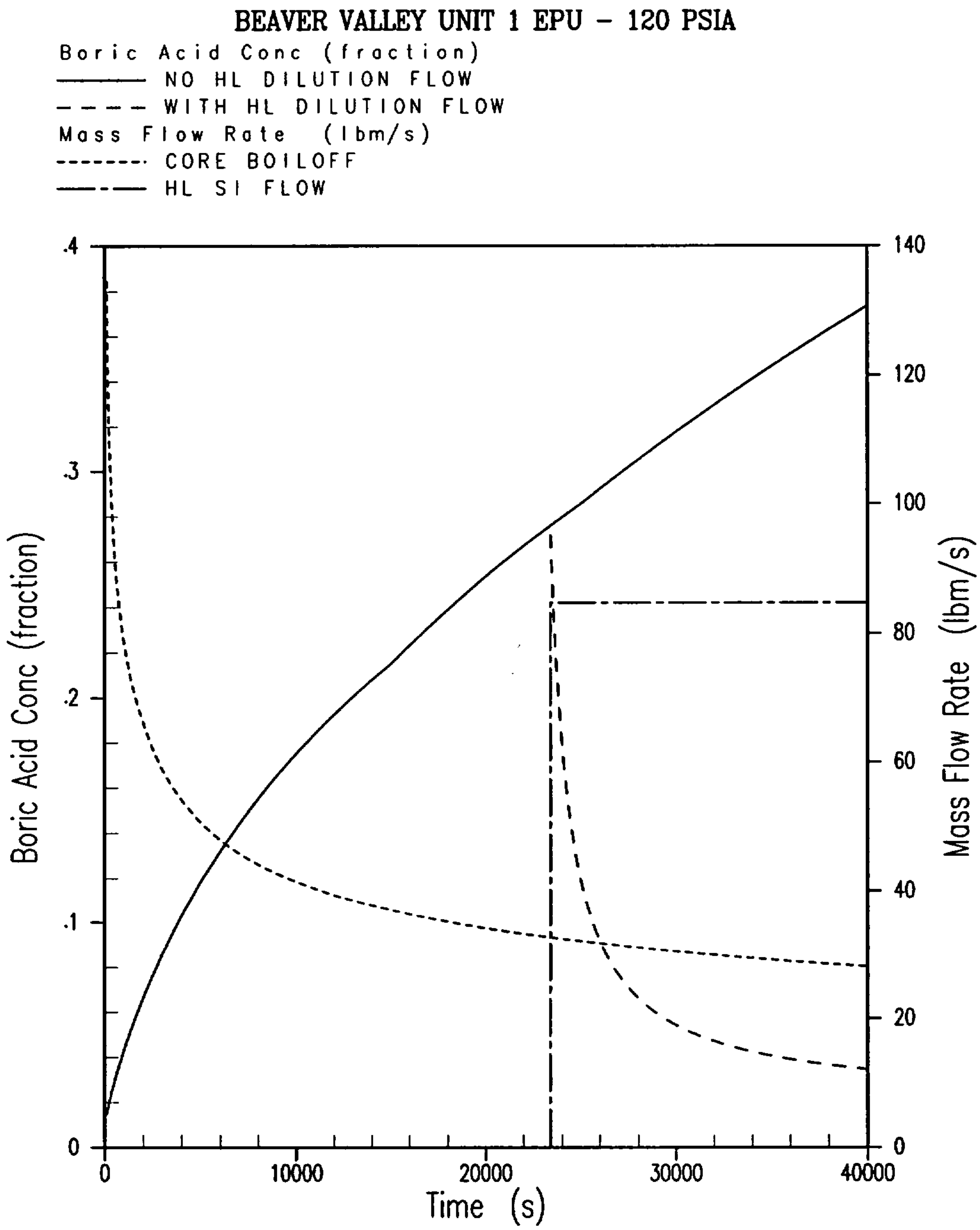

Figure F-6 - Vessel Boric Acid Concentration for 120 psia RCS Pressure (BVPS-1)

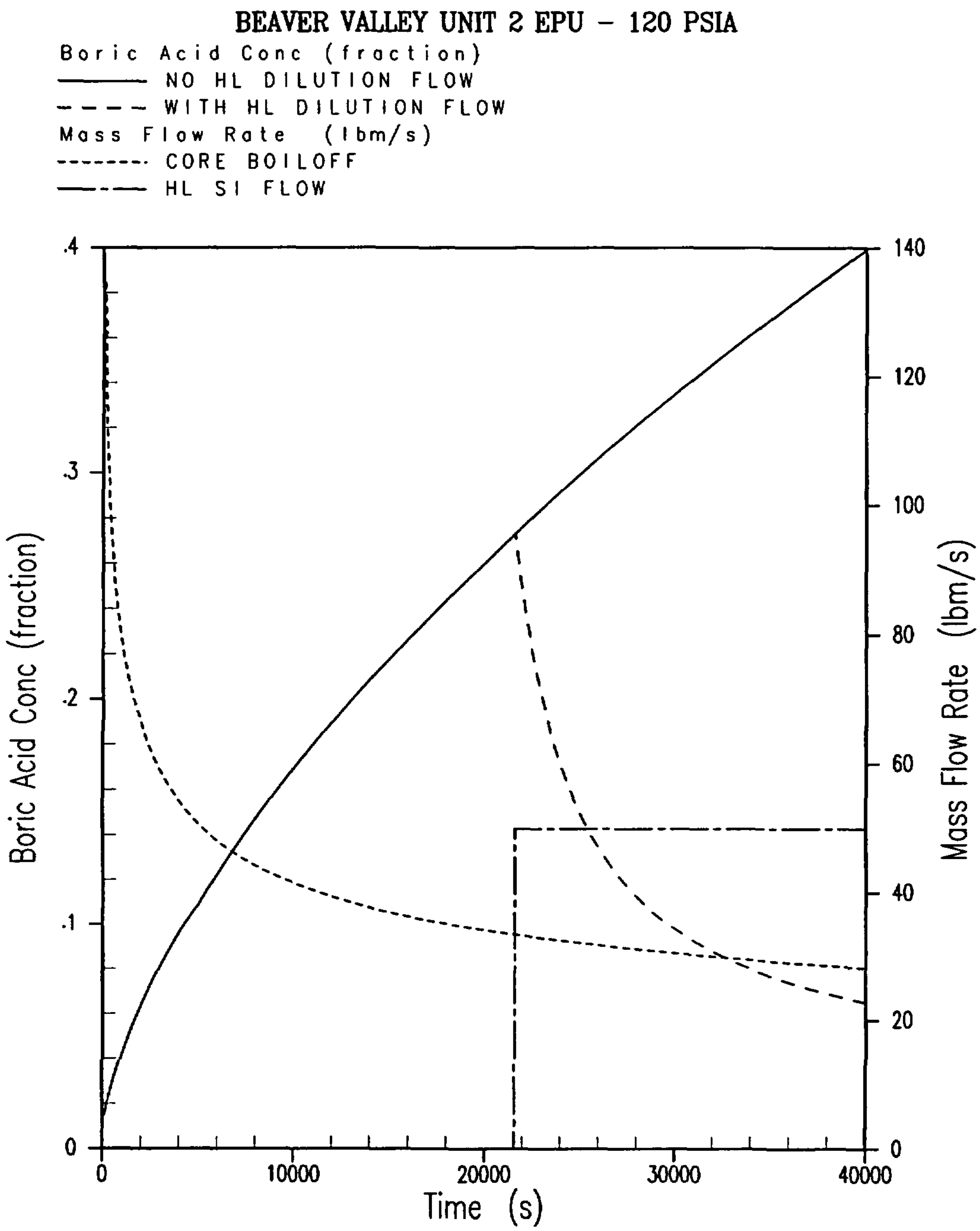

Figure F-7 - Vessel Boric Acid Concentration for 120 psia RCS Pressure (BVPS-2)

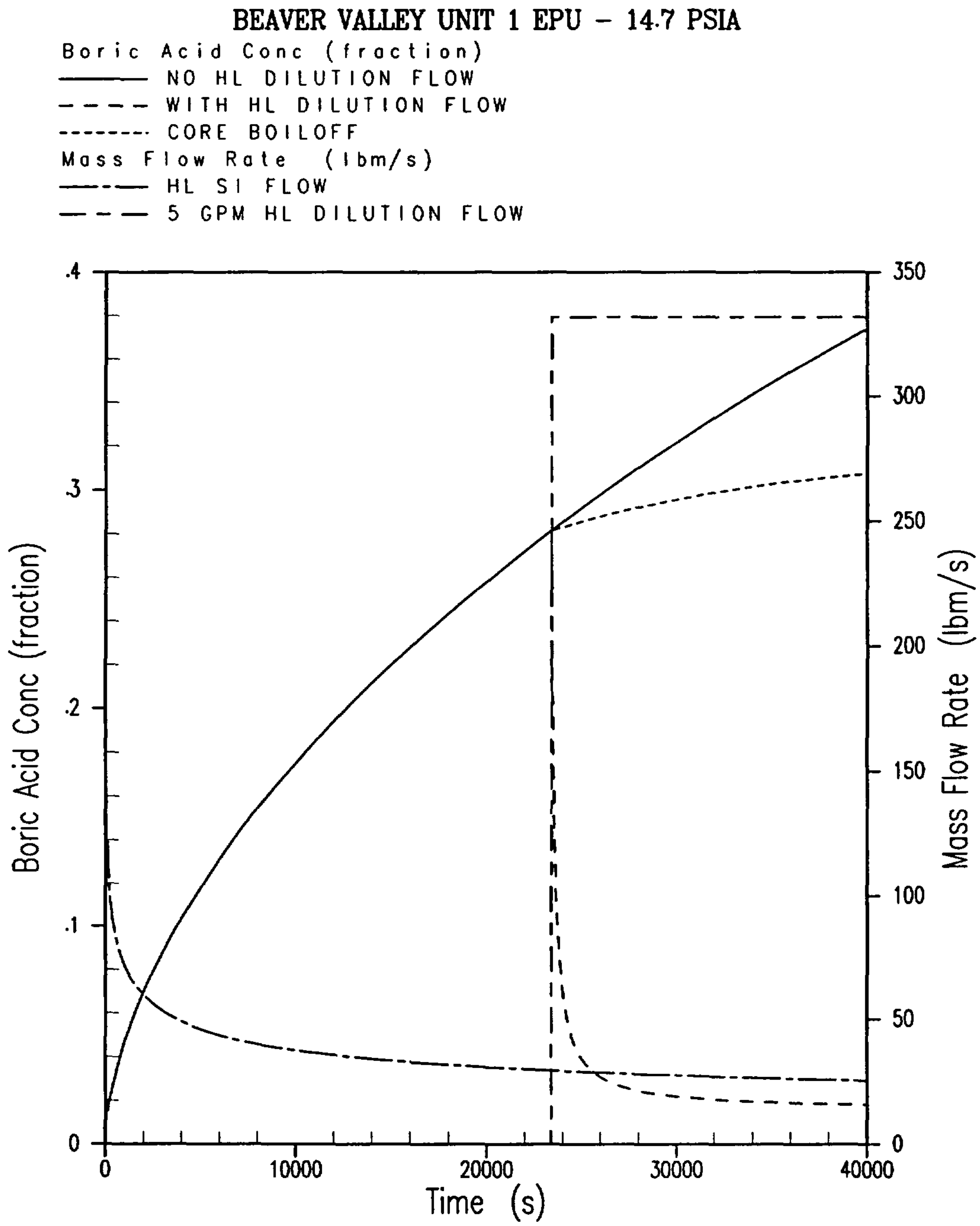

Figure F-8 – Boil-off, SI, and Core Dilution Rate at Hot Leg Switchover Time (BVPS-1)

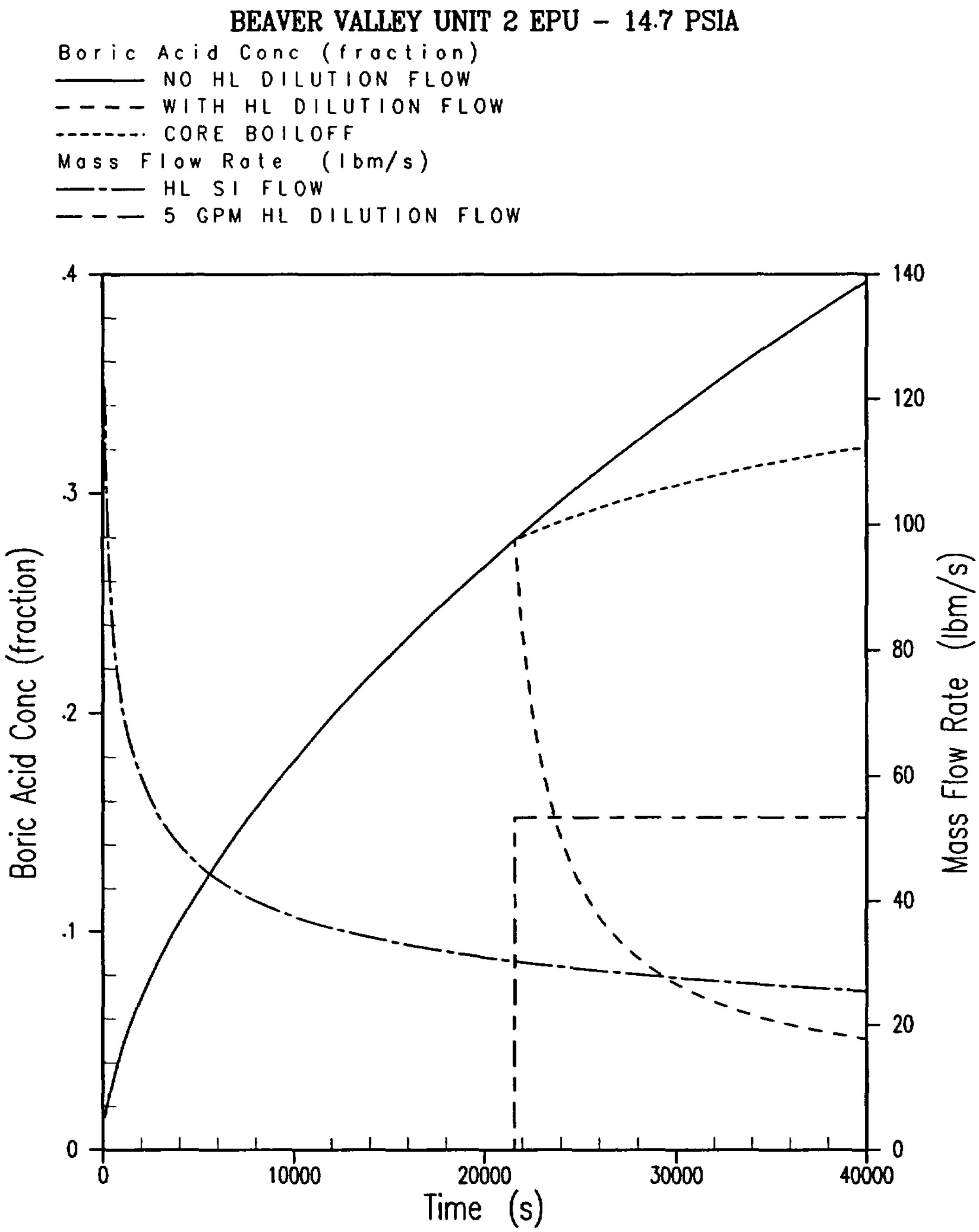

Figure F-9 – Boil-off, SI, and Core Dilution Rate at Hot Leg Switchover Time (BVPS-2)

ENCLOSURE 2 (Proprietary)

Affidavit and Proprietary Responses to RAIs dated September 30, 2005

ENCLOSURE 2

Westinghouse Affidavit

Westinghouse

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Our ref: CAW-05-2071

November 18, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-MPG-05-134 P-Attachment, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Request for Additional Information (RAI) – Extended Power Uprate (EPU) (TAC Nos. MC4645 and MC4646) [EPU RAI #5 – SBLOCA/HLSO] (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-05-2071 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by FirstEnergy Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2071, and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney
L. Feizollahi

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer , who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

BF Maurer

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 18th day
of November , 2005

Sharon L. Fiori

Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "brackets" in LTR-MPG-05-134 P-Attachment, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Request for Additional Information (RAI) – Extended Power Uprate (EPU) (TAC Nos. MC4645 and MC4646) [EPU RAI #5 – SBLOCA/HLSO]" (Proprietary) dated November 18, 2005, for Beaver Valley Power Station, being transmitted by the FirstEnergy Nuclear Operating Company FENOC letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Beaver Valley Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of RAI responses.

This information is part of that which will enable Westinghouse to:

(a) Assist the customer in obtaining NRC approval by responding to NRC.

Further, this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for plant-specific applications for other customers.

(b) Its use by a competitor would improve his competitive position in the design and licensing of a similar product.

(c) The information requested to be withheld reveals specific aspects of safety analysis inputs which were developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).