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Your ref: Docket No. 52-006
Our ref: DCP/NRC1650

November 17, 2003

SUBJECT: Transmittal of Revised Responses to AP1000 DSER Open Items

This letter transmits Westinghouse revised responses to Open Items in the AP1000 Design Safety Evaluation Report (DSER). A list of the revised DSER Open Item responses transmitted with this letter is Attachment 1. The non-proprietary responses are transmitted as Attachment 2.

Please contact me at 412-374-4728 if you have any questions concerning this submittal.

Very truly yours,

R. P. Vijuk

R. P. Vijuk, Manager
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/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1650
2. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated November 17, 2003

D063

DCP/NRC1650
Docket No. 52-006

November 17, 2003

Attachment 1

List of

Non-Proprietary Responses

Table 1	
"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1650"	
3.8.2.1-1 Revision 2 14.3 Meeting Item 19A.2-8 Revision 2 19.1.10.1-5 Revision 2	

DCP/NRC1650
Docket No. 52-006

November 17, 2003

Attachment 2

AP1000 Design Certification Review
Draft Safety Evaluation Report Open Item Non-Proprietary Responses

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.2.1-1 Revision 2

Original RAI Number(s): **None** (April 3, 2003, *meeting summary)*

Summary of Issue:

The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel. During the April 2-5, 2003 audit at Westinghouse, the applicant informed the staff that the final detailed analyses, to be documented in the ASME Design Report, are not available and will be the responsibility of the COL applicant. The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process for AP1000. To complete the staff evaluation of the AP1000 steel containment design, the staff will need to audit the final detailed analyses. This is Open Item 3.8.2.1-1.

<u>Additional NRC Comments in meeting of October 6-9, 2003</u>

The evaluation of the containment vessel should be revised to incorporate the seismic loads described in the latest DCD. These loads were revised following the revised assumptions of shear wall stiffness (see DSER Open Item 3.7.2.3-1). Additional justification should be provided that any of the specified load combinations not evaluated are bounded by those evaluated.

The DCD should be revised to specify critical dimensions as Tier 2*. In particular, the spacing between stiffeners should be specified as Tier 2* since there is little margin in the design calculation for external pressure.

Westinghouse Response (Completely revised in Revision 2):

The detailed design calculations provided for review during the meeting on October 6-9 were initiated before the change in seismic analyses. A separate reconciliation of the new loads was prepared by Westinghouse. The revised loads have now been included in a revision to the Containment Vessel Design Specification. The detail design calculations for the containment vessel have been revised based on the updated specification. This revision also describes the selection of the load combinations and justifies why those not evaluated are less critical. These documents are available for audit.

The maximum vertical spacing of the horizontal stiffeners is added below and identified as Tier 2*.

Design Control Document (DCD) Revision:

Revise fifth paragraph of subsection 3.8.2.1.1 as follows:

The containment vessel includes the shell, hoop stiffeners and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachments. **The design for external pressure is dependent on the spacing of the hoop stiffeners and crane girder which are shown on Figure 3.8.2-1.** ***[The spacing between each pair of ring supports (the bottom flange of the crane girder, the hoop stiffeners, and the concrete floor at elevation 100' 0") is less than 50' 6".]****

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3 Meeting Item

Original RAI Number(s): None

Summary of Issue:

<u>NRC comment from the NRC / WEC meeting on October 30, 2003</u>

Westinghouse should evaluate the need for an ITAAC related to as-built condition of power cables.

Westinghouse Response:

The cable pulling process will be governed by the construction procedures for the plant. Regarding cable pulling tension, proper calculations and procedures will be used to ensure that the pulling tension does not exceed the cable manufacturer's specification for maximum pulling tension. The Quality Assurance program for procurement, fabrication, installation, construction, and testing of structures, systems, and components in the facility will cover the cable pulling process, as well as other installation processes. As stated in DCD section 17.5, this QA program is the responsibility of the Combined License applicant. ITAAC's were not provided for the cable pulling process for any of the three licensed designs. No additional ITAAC are needed for this purpose.

Furthermore, the AP1000 is a passive plant. AC power is not required for pumps, fans, and other motors to keep the plant safe. The Class 1E cables are therefore relatively small, not sized to carry high power, and are simpler to install than larger, heavier power cables.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19.1.10.1-5 (Response Revision 2)**

Original RAI Number(s): **720.009, 720.012, 720.013, 720.014, 720.017, 720.021, 720.024, 720.025**

NRC Follow-on Comments:

A teleconference was held on 8/25/2003 to discuss the Westinghouse response to 19.1.10.1-5. The following provides a summary of the NRC issues that were discussed.

Staff comments on items related to Open Issue 19.1.10.1-5

(a) Additional justification is needed for long-term cooling analyses for which the initial and boundary conditions were obtained from analyses using MAAP4 for input into WCOBRA/TRAC (RAI 720.013):

This issue remains open. In the revised response to RAI 720.013, Westinghouse performed long term cooling analyses for bounding conditions in the PRA. (Case F DEDVI, 1 CMT, 1 recirc line, 3/4 ADS4 and CI) and (Case G DEDVI, 1 CMT, 1 recirc line, 4/4 ADS4 and CI failure). The WCOBRA/TRAC code was used for LTC calculations with input conditions derived from MAAP4 analyses. As discussed in the DSER the staff has not reviewed MAAP4 except for its use in screening studies. These are analyses using minimum equipment sets as discussed in the DSER. The staff believes that only a methodology the staff has reviewed should be utilized. In addition, as a result of staff and ACRS questions, the WCOBRA/TRAC long term cooling model has been changed. The staff believes that the revised model should be used in these bounding calculations.

(b) Additional justification should be provided that a large break LOCA can be mitigated if one of the two CMTs fail (RAI 720.012-2):

This issue remains open. In the revised response to RAI 720.012, Figure 2-1, Westinghouse listed large break LOCA sequences as success sequences (OK7 sequences). Westinghouse should verify these conclusions by using a methodology that the staff has reviewed.

(c) Additional justification should be provided that adequate water can be maintained within the containment to provide for long term core cooling if containment isolation fails (RAIs 720.021 and 720.024):

This response is acceptable based on Westinghouse arguments on the relative elevations between the postulated RCS break and the postulated failed containment penetration and the tortuous path that would be involved.

(d) Additional justification should be provided that one of the two startup feedwater pumps can deliver adequate water to the two steam generators following an ATWS event (RAI 720.024):

This response is acceptable based on new analyses to be added to Appendix A of the PRA.

(e) Additional justification should be provided that evaluations made for AP600 are appropriate to be used in the AP1000 PRA Table 6-1 and in the response to RAI 720.025 where Westinghouse assumes that 30 minutes of core cooling is available following a small break LOCA, steam generator tube rupture or transient with no accumulator injection (RAIs 720.024 and 720.025):

References to AP600 have been removed and acceptable arguments applying to AP1000 have been added. This response is acceptable.

An analysis using MAAP4 was performed to demonstrate that a 30 minute delay in CMT injection is acceptable following a SBLOCA and multiple failures. The consequences were determined to be bounded by a MAAP4 analysis with automatic actuation. Westinghouse asserted that since this is not a limiting case a NOTRUMP analysis is not required. For the manual CMT case there is no ECCS for 2000 seconds after the break. For the automatic actuation case, CMT injection occurs about 200 seconds after the break. How can the automatic CMT actuation injection case be worse than the manual CMT actuation case which has a longer delay time?

(f) Additional justification should be provided that sequences which assume failure of one of the four ADS stage #4 valves and also assume failure of containment isolation, will end in successful core cooling (RAIs 720.012, 720.009 and 720.017):

This issue is unresolved. In the revised response to RAI 720.09, Westinghouse presented the results of an analysis using WCOBRA/TRAC with inputs determined from a MAAP4 analysis. The staff has the same issue with this analysis as is stated under Item a. In the revised response to RAI 720.17, Westinghouse argued that this case is not risk significant and therefore it is not necessary to perform a T&H uncertainty analysis. This argument is not valid since OK6 (See Figure 2-4 of RAI 720.012), OK2 and OK4 sequences (on Figure 2-5 of RAI 720.012) fall in this category. Are these OK sequences considered to be low risk?

(g) Additional Issue - Use of MAAP4 for MSGTR Calculation:

In its response to the staff RAI 440.043 regarding the AP1000 design features that mitigate or prevent steam generator safety valves challenges during an event of rupture of multiple steam generator tubes (MSGTR), Westinghouse provided a beyond-design-basis analysis of MSGTR using MAAP4. Two cases were analyzed: a passive system mitigation case with PRHR heat exchanger operation; and a minimum PRHR heat removal case with the assumption of steam generator safety valve (SGSV) failed open. Based on the MAAP4 analysis, Westinghouse concluded that for the MSGTR, the core remains covered and

cooled, and thus no significant fission product release occurs. In DSER Section 5.4.2.3.2, the staff stated that the staff's evaluation of the use of MAAP4 for the AP1000 PRA evaluation is discussed in Chapter 19 of DSER. In DSER Chapter 19 the staff gave conditions for the use of MAAP4 as described in the above excerpt.

In light of Open Item 19.1.10.1-5 and the concern described in the DSER that MAAP4 does not provide a rigorous solution of reactor system conditions during transients and accidents, the staff requests that Westinghouse confirm the beyond-design-basis MSGTR results of no core uncovery described in response to RAI 440.043 with a methodology reviewed by the staff.

NRC Additional Comment (from telecon):

Westinghouse should quantify the effect of lower containment backpressure as indicated in the response to OI 15.2.7-1 Item 7 on long term cooling performance.

Westinghouse Response (Revision 2):

Revision 2 of this response adds the response to the NRC additional comment, starting on page 62.

The following provides Westinghouse responses to the NRC follow-on comments:

(a) The long term cooling success criteria analysis case and the thermal hydraulic uncertainity cases F and G have now been performed using the revised WCOBRA/TRAC model and using input conditions from WGOTHIC. The revised cases replace the previous cases as shown below in the revision to PRA Appendix A.

(b) In addition to our previous justification, it is noted the AP1000 large-break LOCA analysis performed for Chapter 15 using WCOBRA-TRAC includes a sensitivity study that determines the PCT without credit for operation of the CMTs. Results of these calculations show that the maximum PCT without operation of the CMTs is within the regulatory limits. This sensitivity study is identified in AP1000 DCD subsection 15.6.5.4A. This study provides additional justification that the PRA success criteria is acceptable.

(c) As noted in the NRC follow-on summary, the previous Westinghouse response is acceptable.

(d) As noted in the NRC follow-on summary, the previous Westinghouse response is acceptable.

(e) The case in question (2 inch break, 1 CMT, 0 ACC, 0 ADS1-3, 0 PRHR, ¾ ADS4) has now been performed using NOTRUMP, for automatic CMT actuation and for manual CMT actuation. Comparison plots of these two NOTRUMP cases are shown in Figures 19.1.10.1-5e-1 through 19.1.10.1-5e-6. For the automatic CMT case the CMT draindown starts earlier and leads to earlier actuation of ADS4. For the automatic case the CMT

recirculates from the early part of the event and its water inventory becomes heated before draindown begins, whereas for the manual case CMT draindown occurs shortly after CMT actuation and therefore is injecting subcooled liquid during its draindown. When ADS4 actuation occurs in the automatic case the vessel inventory and CMT injection are near saturation and the rapid depressurization results in sufficient voiding to cause the vessel mixture level to fall into the core region before it recovers as a result of IRWST injection. In the manual case the CMT injection just prior to and following ADS4 actuation is still highly subcooled so the depressurization causes less voiding and the vessel mixture remains in the upper plenum region during the ADS4 to IRWST transition.

These NOTRUMP results confirm the behavior seen in the previous MAAP4 analyses.

(f) The long term cooling success criteria analysis in the revised response to RAI 720.09 (3 of 4 ADS4, containment isolation failure) has now been performed using the revised WCOBRA/TRAC model and using input conditions from WGOTHIC. The revised analysis replaces the previous one as shown below in the revision to PRA Appendix A. This confirms the success criteria for this case. With respect to the statement we made in RAI 720.017 rev 1 about a T&H uncertainty case with 3 ADS-4 valves and failure of containment isolation, we have the following response. Such a sequence is not risk important with the current expanded event trees. The question regarding sequences that we identified as OK sequences with 3 ADS-4 valves and CI failure is a new / different question. These sequences are less severe than the success criteria case in that they have some ADS 2/3 valves opening and have 1 CMT (in addition to 1 Accumulator). The addition of the CMT will compensate for the water lost out of the containment through a failed CI. In addition, these OK sequences have probabilities of ~ E-10 / yr, or less. The highest UC sequence with 3 ADS-4 and failed CI is UC8/sad27, see PRA Appendix A Table A5.1-2. This sequence has a CDF probability of 2 E-10 and is not risk important.

(g) As discussed with the NRC in the teleconference, the use of the MAAP4 code for AP1000 is the same as was approved for AP600. Westinghouse uses MAAP4 to perform analyses of accident sequences from the PRA. For those cases with large margin (i.e. little or no core uncovery), the MAAP4 results are used to validate the PRA success criteria. For low margin success sequences that are also high risk, Westinghouse performs additional analyses with the approved design basis analysis computer codes to demonstrate success.

For both AP600 and AP1000, Westinghouse has submitted results of the analysis of a multiple steam generator tube rupture. For both AP600 and AP1000, there is a large margin to core uncovery, and the results have been accepted for the purpose of demonstrating the plant response to this beyond design basis accident. The previous analysis results provided for AP1000 did not identify the top of the active fuel, and the staff was unable to assess the large margin in the analysis results. The attached figures 19.1.10.1-5g-1 through 4 show the results of the multiple steam generator tube rupture for AP1000. The MAAP4 code is applied appropriately for this large margin case.

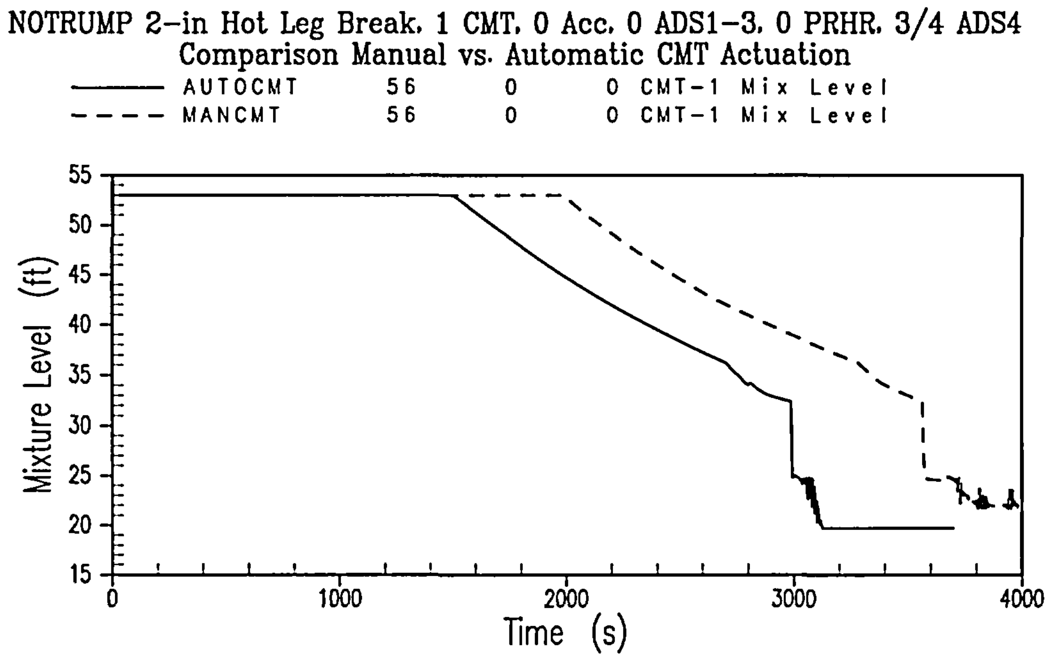

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Figure 19.1.10.1-5c-6- 2: CMT Level for 2-inch Hot Break Leg PRA Case - NOTRUMP

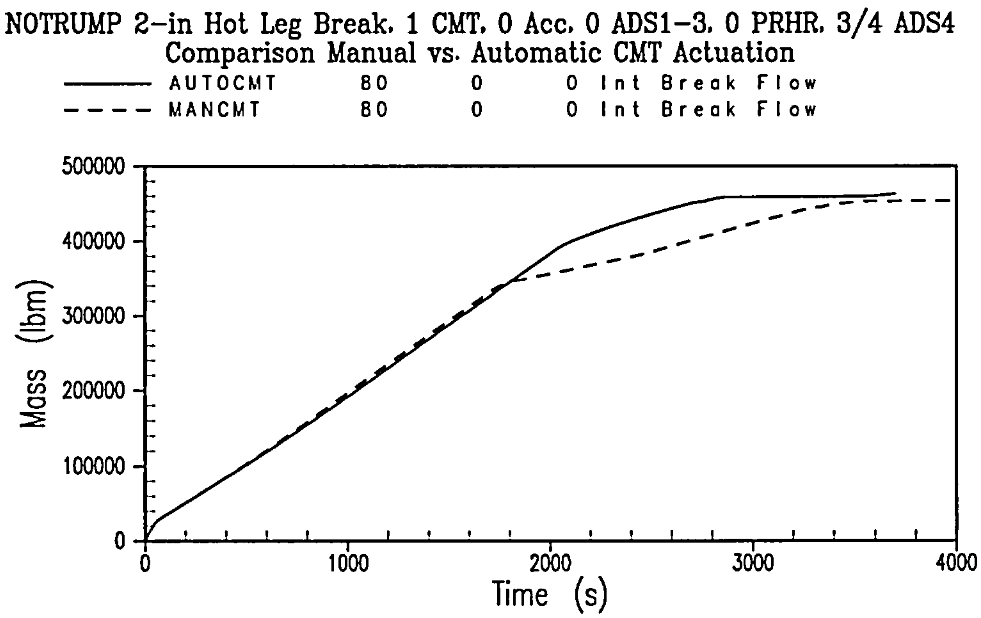


Figure 19.1.10.1-5c-6- 3: Integrated Break Flow for 2-inch Hot Leg Break PRA Case - NOTRUMP

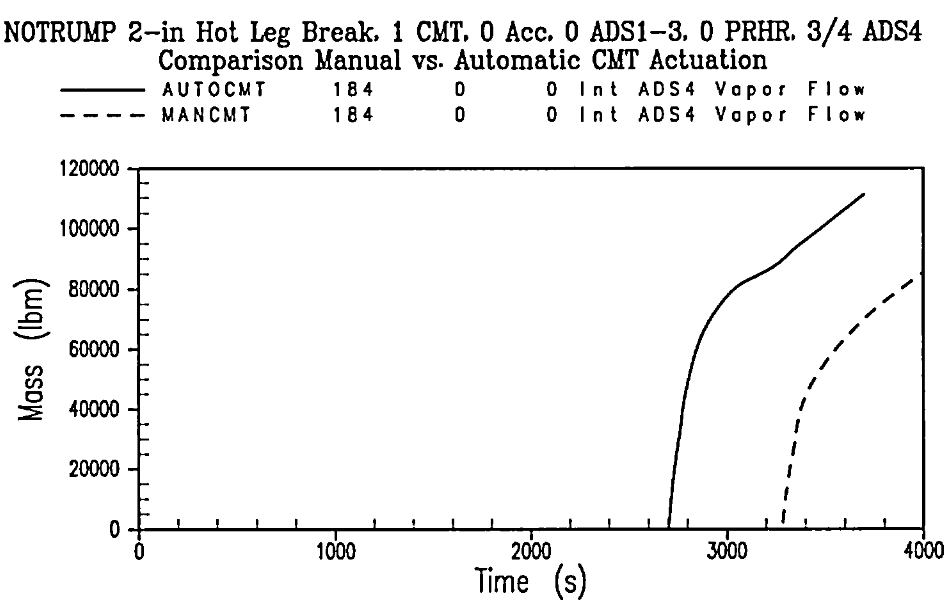


Figure 19.1.10.1-5c-6- 4: Integrated ADS4 Vapor Flow for 2-inch Hot Leg Break PRA Case - NOTRUMP

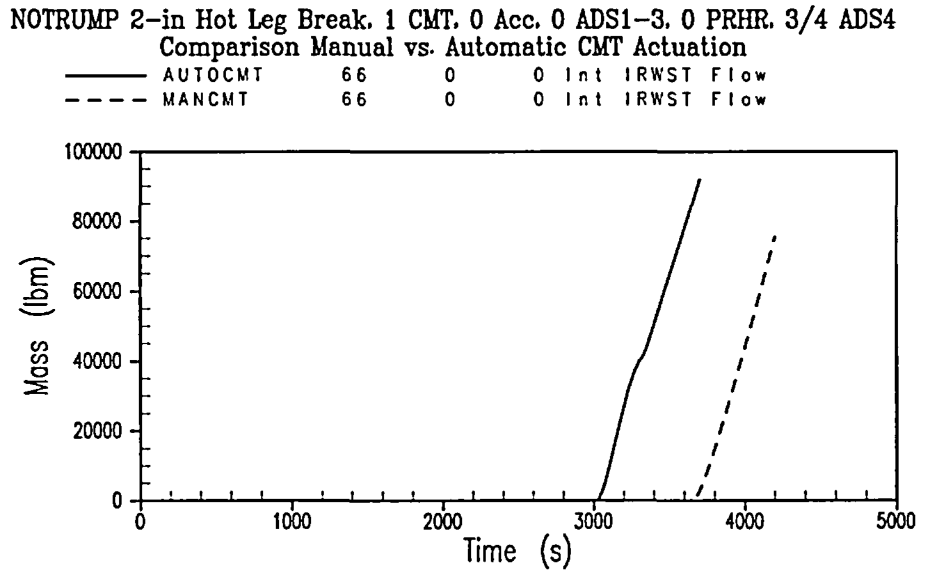


Figure 19.1.10.1-5c-6- 5: Integrated IRWST Flow for 2-inch Hot Leg Break PRA Case - NOTRUMP

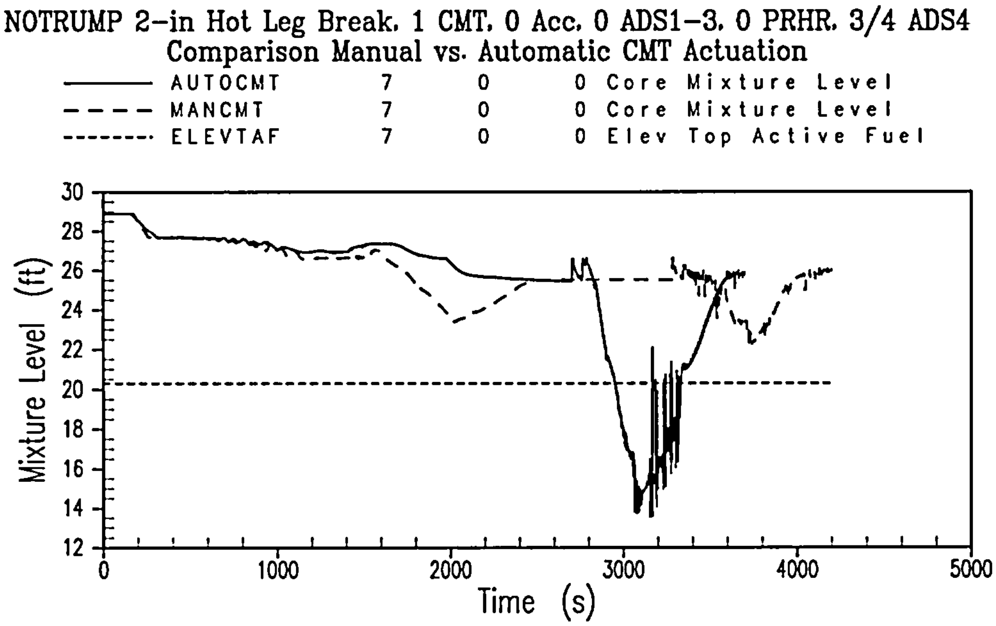


Figure 19.1.10.1-5e-6- 6: Core Mixture Level for 2-inch Hot Leg Break PRA Case - NOTRUMP

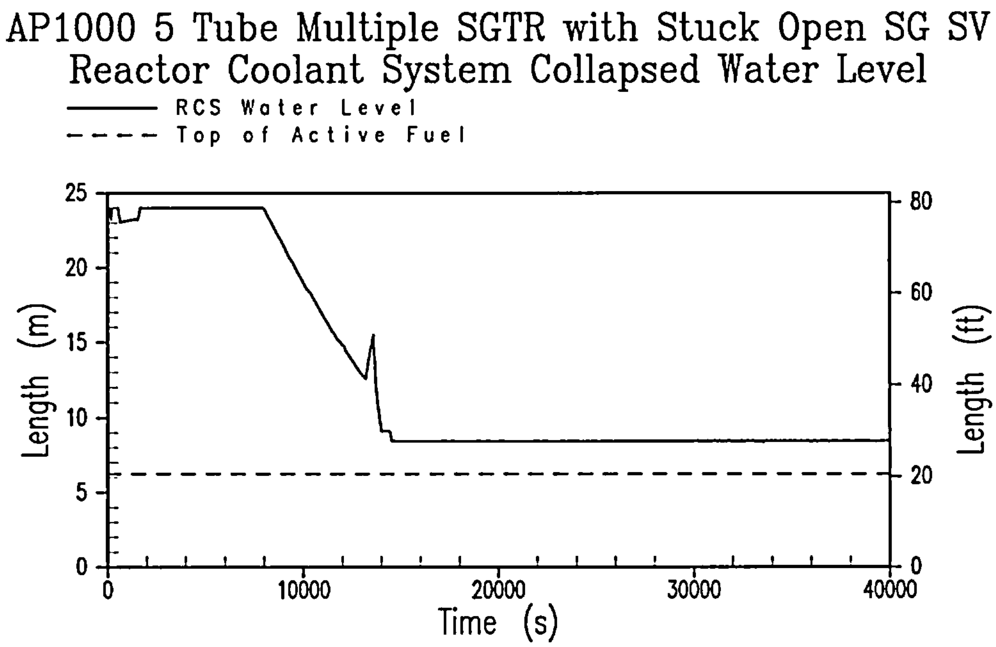


Figure 19.1.10.1-5g-1

AP1000 5 Tube Multiple SGTR with Stuck Open SG SV
Core Average Void Fraction

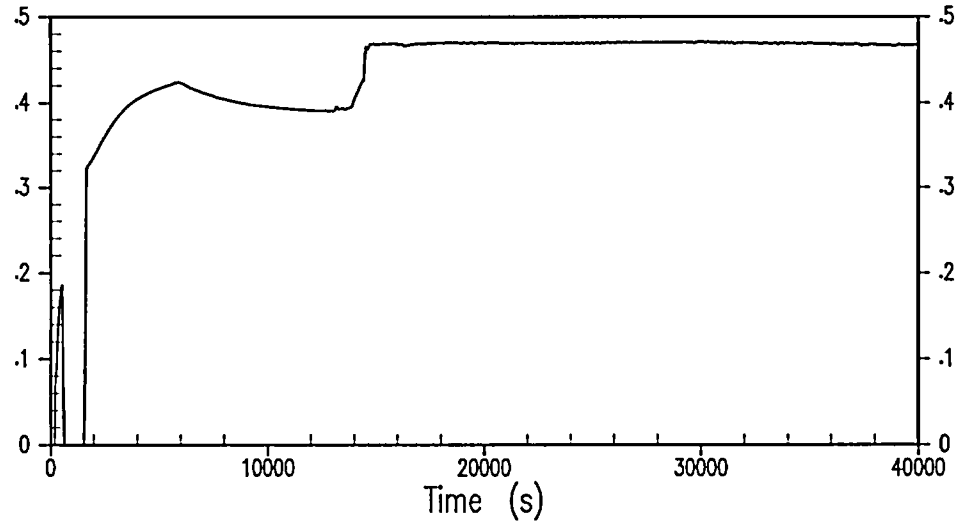


Figure 19.1.10.1-5g-2

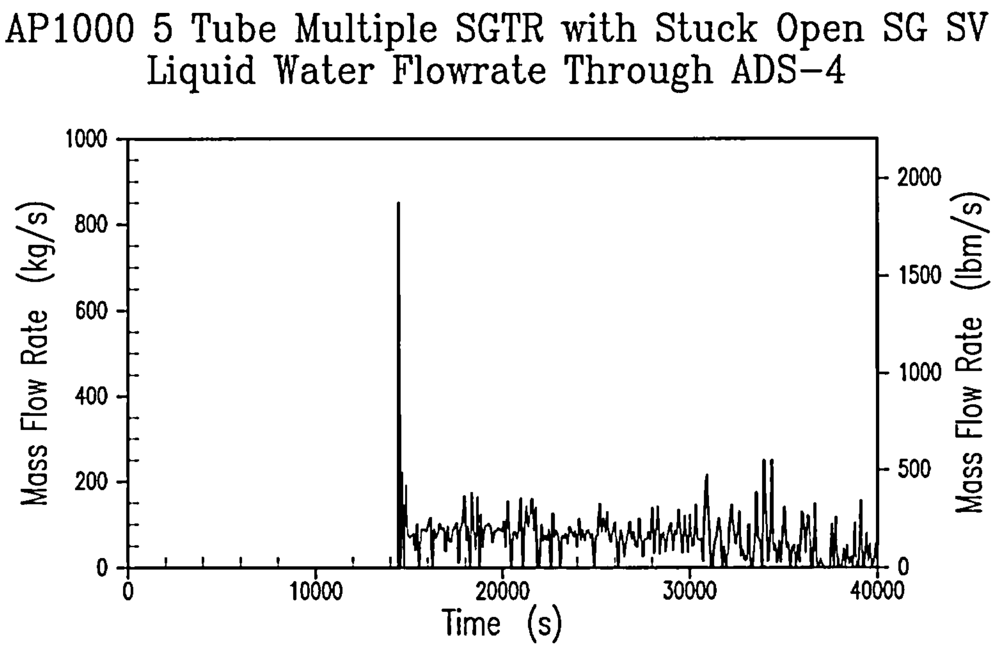


Figure 19.1.10.1-5g-3

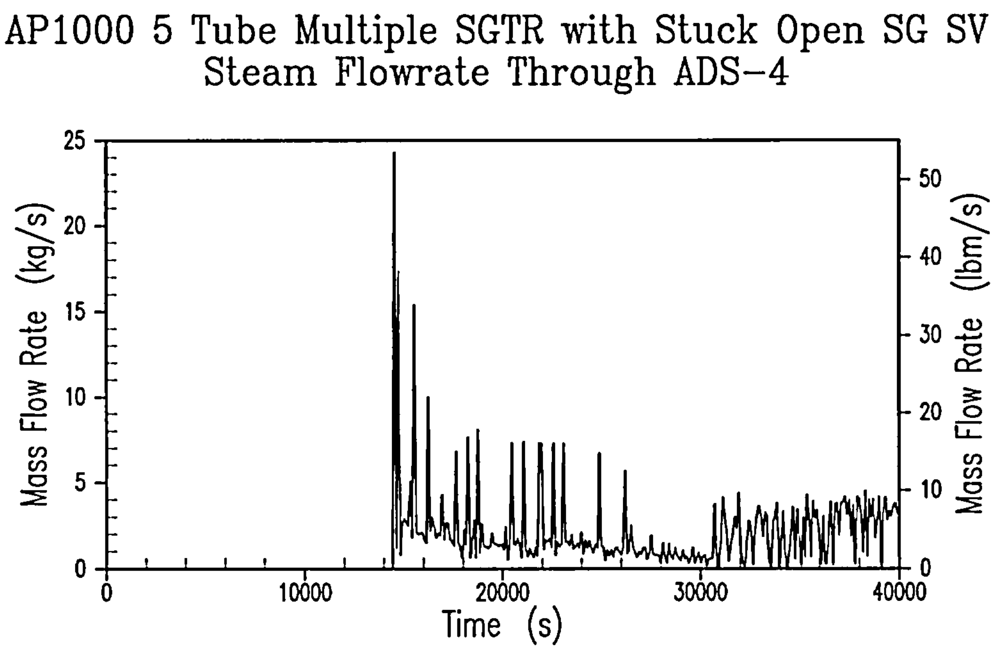


Figure 19.1.10.1-5g-4

Design Control Document (DCD) Revision:

None

PRA Revision:

~~None~~ **PRA Appendix A will be revised as shown below.**

A3.5 Success Criteria Analysis for Long-Term Cooling

This analysis considers the AP1000 long-term core cooling (LTCC) behavior following a guillotine double-ended direct vessel injection (DEDVI) line break to support the PRA success criteria evaluations. The limiting success criteria scenario is analyzed in order to perform a bounding case. This analysis is performed with WCOBRA/TRAC using the long-term cooling code version with realistic inputs.
The DEDVI line break LTCC scenario analyzed conservatively assumes that the break occurs in the PXS-B room. Since the size of this room is bigger than PXS-A, the containment water level during the transient is reduced. A short summary follows of the boundary conditions for the case analyzed herein:

- DEDVI LOCA in line B
- Available equipment – 1/1 ~~CMT~~-ACC (A), both IRWST injection lines open with 1/2 valves open in each, 1/2 recirculation lines available with both valves open in the line attached to DVI-B, 3/4 ADS-4, PCS water drain with 1/3 valves open
- Unavailable equipment – no ADS 1/2/3, CMT, PRHR, RNS injection/spill, IRWST gutter
- Containment isolation assumed to have failed (~~16~~18-inch HVAC line remains open)

A3.5.1 WCOBRA/TRAC LTCC Modeling Methodology

The simulation methodology used in the current analysis is essentially the same as the one used for the AP600 design certification process (Reference A-4).

- The T/H analysis is performed using the WCOBRA/TRAC T/H computer code (Reference A-27).
- The WCOBRA/TRAC AP1000 model is the same as the one used in the AP1000 DCD Post-LOCA Long-Term Cooling analysis (Reference A-26).
- The AP1000 LTCC simulations are performed using WCOBRA/TRAC in a ~~transient~~ window mode. The ~~transient~~-window mode approach has been validated by the Oregon State University Tests and was used in the AP600 Design Certification (Reference A-4).
- For each case, the AP1000 ~~initial and boundary~~ conditions are provided by a ~~MAAP4~~ **combined WGOTHIC analysis and hand calculation.** ~~MAAP4 is capable of simulating the behavior and the interaction between the AP1000 primary system, the passive safety systems, the containment, and the~~ **WGOTHIC can predict the performance** of containment systems – a feature not available with WCOBRA/TRAC.
- ~~Like the corresponding MAAP4 case, the following~~ The WCOBRA/TRAC success criteria simulation is performed with the following general assumptions:
  - 100-percent core power
  - ANS 1979 standard best-estimate decay heat
  - Nominal hydraulic resistance of the passive safety systems

A3.5.2 Methodology Implementation

The ~~transient-~~window mode calculation using WCOBRA/TRAC allows simulation of long transients with reasonable computer resources. As was shown in the validation of methods used in the DCD analysis (Reference A-26), the calculation may be initiated from an arbitrary set of initial conditions. After an initial period of 500 to 1000 seconds, the plant reaches a quasi-steady-state that depends only on the system boundary conditions. During this "steady-state" period, the boundary conditions are kept constant. After that, they are set as a function of time depending on the time window being simulated.

For the AP1000 Success Criteria analysis, a ~~transient-~~window mode calculation was performed for ~~a segment of the time period covered by the MAAP4 calculation for the same case. It was observed that WCOBRA/TRAC predicts higher ADS Stage 4 flows resulting in better depressurization of the primary system. Consequently, the predicted IRWST injection rates were higher when using WCOBRA/TRAC. Because of the faster IRWST drain prediction, it is estimated that the IRWST would reach its lowest level about 1.6 hours earlier than is predicted by MAAP4. Therefore, the IRWST level calculated by MAAP4 was adjusted to account for the more rapid draining predicted by WCOBRA/TRAC. The adjusted IRWST level was then used as a boundary condition for the case. The containment pressure, PXS-B level, IRWST, and PXS-B temperatures calculated by MAAP4, together with the adjusted IRWST level, were used to define the limiting boundary conditions for the WCOBRA/TRAC assessment of the performance of the AP1000 passive safety systems. The following subsection documents the results of the WCOBRA/TRAC simulation.~~**the limiting time period, as identified in the DCD LTC analysis of the DEDVI break immediately following the switchover to sump recirculation. The containment water level is computed considering the mass discharged through the open purge line as calculated by WGOTHIC.**

The containment~~pressure, PXS-B level~~**sump**, IRWST, and PXS-B temperatures calculated by ~~MAAP4, together with the adjusted IRWST level, were used to define the limiting boundary conditions for~~**WGOTHIC in the AP1000 DCD analysis are used** in the WCOBRA/TRAC assessment of the performance of the AP1000 passive safety systems. The following subsection documents the results of the WCOBRA/TRAC simulation.

A3.5.3 Predictions for a DEDVI Line Break in PXS-B Room with Three of Four ADS Stage 4, Containment Isolation Failed

This subsection presents the simulation results of the Success Criteria Case – a DEDVI line break located in the PXS-B room with three out of four ADS Stage 4 valves open and failure of the containment to isolate. The initial conditions are based on the ~~MAAP4 calculation results of the same~~**DCD analysis of the PXS "B" room break** accident scenario. They are selected such that the WCOBRA/TRAC simulation begins ~~4096~~**9300** seconds ~~(approximately 1 hour, 8 minutes)~~ after the break ~~– after IRWST injection has been fully established~~**the time at which switchover to sump recirculation occurs.**

For the WCOBRA/TRAC transient, the ~~initial IRWST~~**containment** water level is ~~126~~**107.2** feet~~and the liquid temperature is 120°F. The level in the PXS-B room at this time is 96.95 feet~~. The available ADS Stage 4 paths are open, and the containment pressure is set to ~~the MAAP value of~~ 14.7 psia. With these conditions, a 1000-second calculation is performed to ensure that a proper initial condition is achieved in the system, **and the window mode**~~. After that, the transient~~ calculation is initiated with ~~time-dependent~~**fixed** boundary conditions ~~taken from the MAAP4 calculation, but using an adjusted IRWST level function, as discussed earlier~~.

~~Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure A3.5-14). At the beginning of the analysis, the liquid level in the PXS-B room is below the DVI injection nozzle elevation and steam from the downcomer is vented out through the break (Figure A3.5-13). Water starts to flow back into the downcomer through the broken DVI line during the transient when the liquid level in the PXS-B room becomes high enough to provide sufficient driving head. With the onset of this flow, additional water supplied~~

~~into the downcomer through the DVI break supplements the IRWST injection. As this flow path becomes established, t~~The levels in the downcomer (Figure A3.5-1), the reactor core (Figure A3.5-2), and the upper plenum (Figure A3.5-8) are sufficient to maintain core cooling~~increase. The effect of this injection flow increase is also seen in Figures A3.5-3 and A3.5-4, which show a void fraction decrease in the upper half of the core~~.

The three available ADS Stage 4 valves provide enough venting capacity that adequate depressurization capability exists to achieve successful performance of the passive safety systems (Figures A3.5-9 and A3.5-10). The fuel remains covered throughout the transient, and adequate core cooling is provided to remove the decay heat. The hot rod cladding temperature at the top of the core is slightly above saturation temperature (Figure A3.5-12) throughout the transient.

~~As the transient proceeds, the IRWST drains to a minimum level slightly above 107 feet. After that time, the level continues to decrease slightly due to loss of fluid from the containment, as predicted by MAAP4. The transient is terminated at about 4.2 hours after the break occurs with no additional leakage from the containment, and with system pressure constant, stable DVI injection flows, and decreasing decay heat.~~

A5.6 T/H Uncertainty Analysis for Long-Term Cooling

The objective of these analyses is to analyze the AP1000 long-term core cooling (LTCC) behavior following a guillotine double-ended direct vessel injection (DEDVI) line break to support the PRA T/H uncertainty evaluations. In order to bound the T/H uncertainty, this analysis is performed using the DCD code and conservative methods.

Two cases of LTCC following a DEDVI line break are analyzed. These cases were determined by T/H uncertainty evaluations performed for AP1000 (in Section A5). One of these cases considers that the containment is isolated (Case F), and the other case considers that the containment isolation has failed (Case G). It is conservatively assumed that the DEDVI line break occurs in the PXS-B room. Since the size of this room is bigger than PXS-A, it reduces the containment water level during recirculation. It also takes more time for the water to fill it to the DVI nozzle elevation, where water can start flowing into the downcomer through the broken DVI line. In both cases, the general assumptions and methodology of the calculations are essentially the same. Conservative boundary and initial conditions are applied consistent with these multiple failure PRA-based scenarios to ensure that the T/H uncertainties contained within the success criteria are bounded.

A short summary follows of the two T/H uncertainty cases described herein.

- Case F:
  - DEDVI LOCA in line B
  - Available equipment – 1/1 ~~CMT~~ ACC (A), ~~both~~ one IRWST injection lines open with 1/2 valves open in ~~each~~ it, only 1 recirculation line available with ~~both~~ one valves open and this is ~~the line attached to DVI-B,~~ 3/4 ADS-4, PCS water drain with 1/3 valves open
  - Unavailable equipment – no ADS 1/2/3, PRHR, CMT, RNS injection/spill, IRWST gutter
  - Containment isolation is assumed to have worked.

- Case G:
  - DEDVI LOCA in line B
  - Available equipment – 1/1 ~~CMT~~ ACC (A), ~~both~~ one IRWST injection lines open with 1/2 valves open in ~~each~~ it, 1/2 recirculation lines open with both valves open (line B), 4/4 ADS-4, PCS water drain with 1/3 valves open
  - Unavailable equipment – no ADS 1/2/3, PRHR, CMT, RNS injection/spill, IRWST gutter
  - Containment isolation is assumed to have failed (18-inch HVAC line remains open).

A5.6.1 WCOBRA/TRAC LTCC Modeling Methodology

The simulation methodology used in the current analyses is ~~essentially the same as the one used for the AP600 design certification process (Reference A-4).~~ as follows:

- The T/H uncertainty analyses are performed using the WCOBRA/TRAC thermal hydraulic computer code (Reference A-27).
- The WCOBRA/TRAC AP1000 model is the same as the one used in the AP1000 DCD Post-LOCA Long-Term Cooling analysis (Reference A-26)
- The AP1000 LTCC simulations are performed using WCOBRA/TRAC in a ~~transient~~ window mode. The ~~transient~~ window mode approach has been validated by the Oregon State University Tests and was used in the AP600 Design Certification (Reference A-4).
- For each case, the AP1000 ~~initial and~~ boundary conditions are provided by a ~~MAAP4~~ combined WGOTHIC analysis and hand calculation. ~~MAAP4 is capable of simulating the behavior and the interaction between the AP1000 primary system, the passive safety systems, the containment, and the~~ WGOTHIC can predict the performance of containment systems – a feature that is not present in WCOBRA/TRAC.
- Like the ~~MAAP4, the~~ DCD LTC analysis, these WCOBRA/TRAC simulations ~~is~~ are performed with the following conservative general assumptions:
  - 102-percent core power
  - Appendix K decay heat
  - Maximum hydraulic resistance of the passive safety systems

A5.6.2 Methodology Implementation

The ~~transient~~ window mode calculation using WCOBRA/TRAC allows simulation of long transients with reasonable computer resources. As was shown in the validation of methods used in the DCD analysis (Reference A-26), the calculation may be initiated from an arbitrary set of initial conditions. After an initial period of 500 to 1000 seconds, the plant reaches a quasi-steady-state that depends mostly on the system boundary conditions. During this "steady-state" period, the boundary conditions are kept constant. After that, they are set as a function of time depending on the time window being simulated.

For the AP1000 T/H uncertainty analysis, a ~~transient~~ window mode calculation was performed for Case F and Case G. ~~within the~~ The time period covered by the ~~MAAP4~~ WCOBRA/TRAC calculations for the cases is the plant condition identified as limiting in the DCD analysis transient simulation of the DEDVI break immediately following the switchover to sump. ~~those cases. It was observed that WCOBRA/TRAC predicts higher ADS Stage 4 flows resulting in better depressurization of the primary system. Consequently, the predicted IRWST injection rates were higher when using WCOBRA/TRAC. Because of the faster IRWST draining, it was estimated that the IRWST would reach its lowest level about 2 hours earlier than as predicted by MAAP4.~~

For each of the cases analyzed here (Case F and Case G), the ~~IRWST~~ level of water in containment is identified with due consideration of any mass discharged through any open containment vent path. ~~calculated by MAAP4 was adjusted to account for the more rapid draining predicted by WCOBRA/TRAC. The adjusted IRWST levels were then used as boundary conditions for each of the cases, F and G.~~

The containment pressure, ~~PXS-B level,~~ IRWST, sump, and PXS-B temperatures calculated by ~~MAAP4, together with the adjusted IRWST level, were used to define the limiting conditions~~ WGOTHIC in the AP1000 DCD analysis are used in Case F~~used~~ to assess the performance of the AP1000 passive safety system. For Case G, asmospheric pressure is specified, and the water level is adjusted to account for the water mass lost out of the unisolated containment as computed by WGOTHIC.

The following two sections document the results of the WCOBRA/TRAC simulations for these limiting windows performed for Cases F and G.

A5.6.2.1 Case F – DEDVI Line Break in the PXS-B Room with Three of Four ADS Stage 4, Containment Isolated

This subsection presents the simulation results of T/H uncertainty Case F – DEDVI line break located in the PXS-B room with three out of four ADS Stage 4 valves opened and the containment isolated. The initial conditions are based on the ~~MAAP4~~ DCD calculation results of the PXS "B" room break ~~same~~ accident scenario. They are selected such that the WCOBRA/TRAC simulation begins ~~3992~~ 9300 seconds ~~(approximately 1 hour, 6 minutes)~~ after the break – ~~shortly after IRWST injection begins.~~ the time at which the switchover to sump recirculation occurs.

For this transient, the ~~initial IRWST~~ containment water level is ~~126.4~~ 107.1 feet. ~~and its~~ tTemperature is ~~121~~198°F in the sump and 142°F in ~~. The initial level in~~ the PXS-B room ~~is 95.8 feet~~. The available ADS Stage 4 paths are opened, and the containment pressure is set to its initial value of ~~42.9~~ 24.5 psia. Under these conditions, a 1000-second calculation is performed to ensure that the initial steady-state conditions are achieved in the system~~. After that~~, and the ~~transient~~ window mode calculation is initiated with ~~time-dependent~~ fixed boundary conditions ~~taken from the MAAP4 calculation, but with adjusted IRWST level decrease, as discussed earlier.~~

~~Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure A5.3-14). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from the downcomer is vented out through the break (Figure A5.3-13). Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient.~~ This is the time when the level is at a minimum in containment, yet decay heat remains relatively high ~~in the PXS-B room becomes high enough to provide sufficient driving head. At the onset of this event, the additional amount of water supplied into the downcomer through the DVI break supplements the IRWST injection. This leads to enhanced core cooling, and~~

~~momentarily, faster depressurization occurs at about 2.05 hours into the transient (Figure A5.3-11). Consequently, the IRWST injection is increased even further, and as a result, the~~ The levels in the downcomer (Figure A5.3-1), the reactor core (Figure A5.3-2), and the upper plenum (Figure A5.3-8) are ~~also increased. The effect of this injection flow increase can also be seen on Figure A5.3-4, which shows a sharp void fraction decrease in the upper half of the fuel region~~adequate to maintain acceptable core cooling.

The available three out of four ADS Stage 4 valves provide enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures A5.3-9 and A5.3-10). The fuel remains covered throughout the transient and adequate core cooling is provided to remove the decay heat. The hot rod cladding temperature at the top of the core is about 20°F above saturation (Figure A5.3-12) and ~~is steadily decreasing~~exhibits no excursions.

~~As the transient proceeds, the IRWST drains to a minimum of 107 feet at about 3.9 hours after the break. After that time, the level is kept constant at 107 feet, as predicted by MAAP4. The transient is terminated at about 4.2 hours after the break with the system~~The window mode demonstrates that AP1000 is in a continuing depressurization phase with stable DVI injection flows, and decreasing decay heat, for the limiting time in the Case F scenario.

A5.6.2.2 Case G – DEDVI Line Break in the PXS-B Room with Four of Four ADS Stage 4, Containment Isolation Failed

This subsection presents the simulation results of T/H uncertainty Case G – DEDVI line break located in the PXS-B room with all ADS Stage 4 valves available and with containment isolation failure. The initial conditions are based on the ~~MAAP4~~DCD calculation results of the PXS "B' room break ~~same~~ accident scenario. They are selected such that the WCOBRA/TRAC simulation begins ~~3298~~9300 seconds ~~(approximately 55 minutes)~~ after the break ~~– shortly after IRWST injection begins~~the time at which the switchover to sump recirculation occurs.

For this transient, the ~~initial IRWST~~ containment water level is ~~127.9~~ 106.7 feet. ~~and its~~ ~~t~~Temperature is ~~120.5~~198°F in the sump and 142°F~~. The initial level~~ in the PXS-B room ~~is 93.1 feet~~. All the ADS Stage 4 paths are opened, and the containment pressure is set ~~to its initial value of 17.08 psia, as calculated by MAAP4. Under~~ 14.7 psia. Under these conditions, first a 1000-second calculation is performed so that the ~~initial~~ proper steady-state is achieved in the system, and. ~~After that,~~ the ~~transient~~ window mode calculation is initiated with ~~time-dependent~~ fixed boundary conditions ~~taken from the MAAP4 calculation, but with the adjusted IRWST level decrease~~.

~~Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure A5.3-28). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from the downcomer is vented out through the break. Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient.~~ This is the time when the level has about reached its minimum value in containment, yet decay heat is still relatively high ~~in the PXS-B room becomes high enough to provide sufficient driving head for this to happen. This time, unlike the Case F DVI break scenario, the transition into reversed injection of water through the break into the downcomer occurs a little earlier, and is somewhat softer. As a result, the increased depressurization rate observed in Case F does not occur.~~ Still, the levels in the downcomer (Figure A5.3-15), the reactor core (Figure A5.3-16) and the upper plenum (Figure A5.3-22) are maintained high enough by the available DVI injection to provide acceptable core cooling.

The availability of all ADS Stage 4 valves provides enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures A5.3-23 and A5.3-24). The fuel remains covered throughout the transient, and adequate core cooling is provided to remove the decay heat. The hot rod cladding temperature at the top of the core is about 20°F above saturation (Figure A5.3-26) ~~and steadily decreasing~~.

~~As the transient proceeds, the IRWST drains to a minimum of 106.9 feet at about 3.7 hours after the break. After that time, the level is kept constant at 106.9 feet, as predicted by MAAP4. The transient is terminated at about 4.4 hours after the break with~~ The window mode calculation shows the system being in a phase with stable DVI injection flows, adequate ADS 4 flows, and decreasing decay heat for the limiting time in the Case G scenario.

The following figures will replace the existing ones in PRA Appendix A.

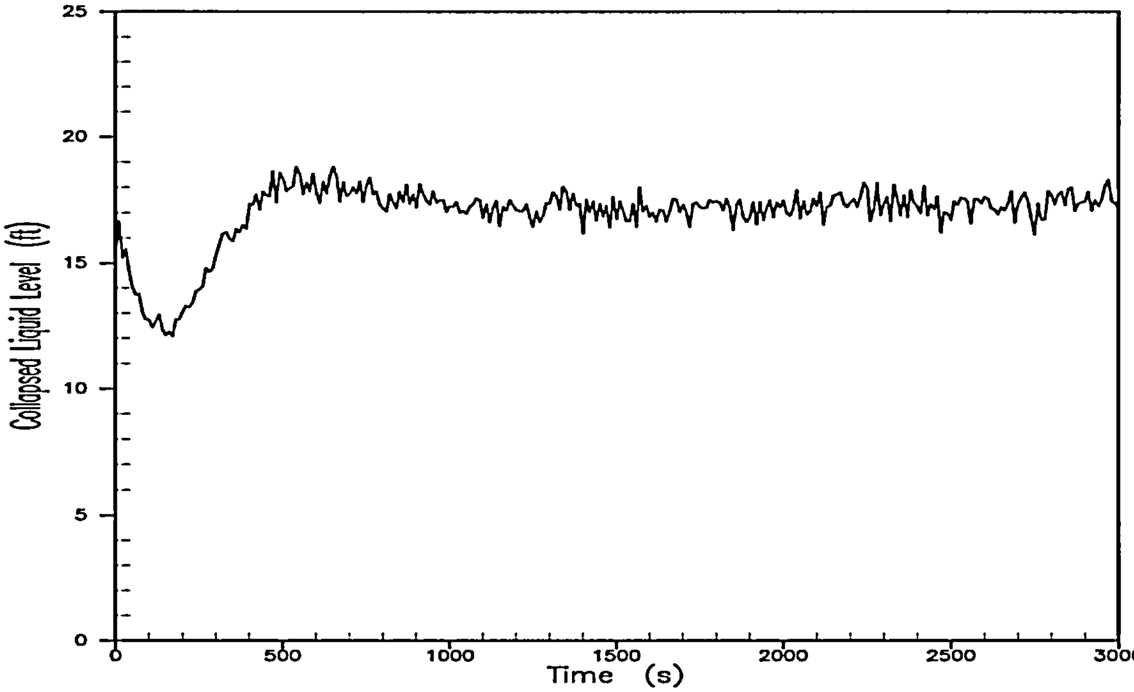


Figure A3.5-1

LTCC DEDVI Break Success Criteria – Collapsed Level of Liquid in Downcomer

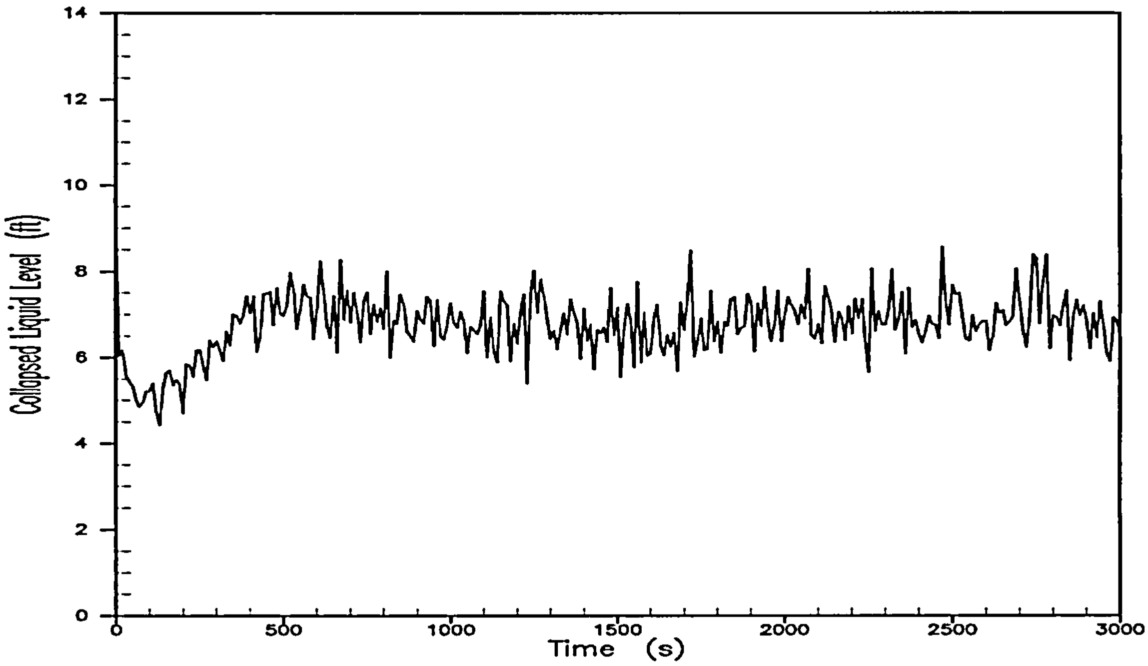


Figure A3.5-2

LTCC DEDVI Break Success Criteria – Collapsed Level of Liquid Over Heated Length of Fuel

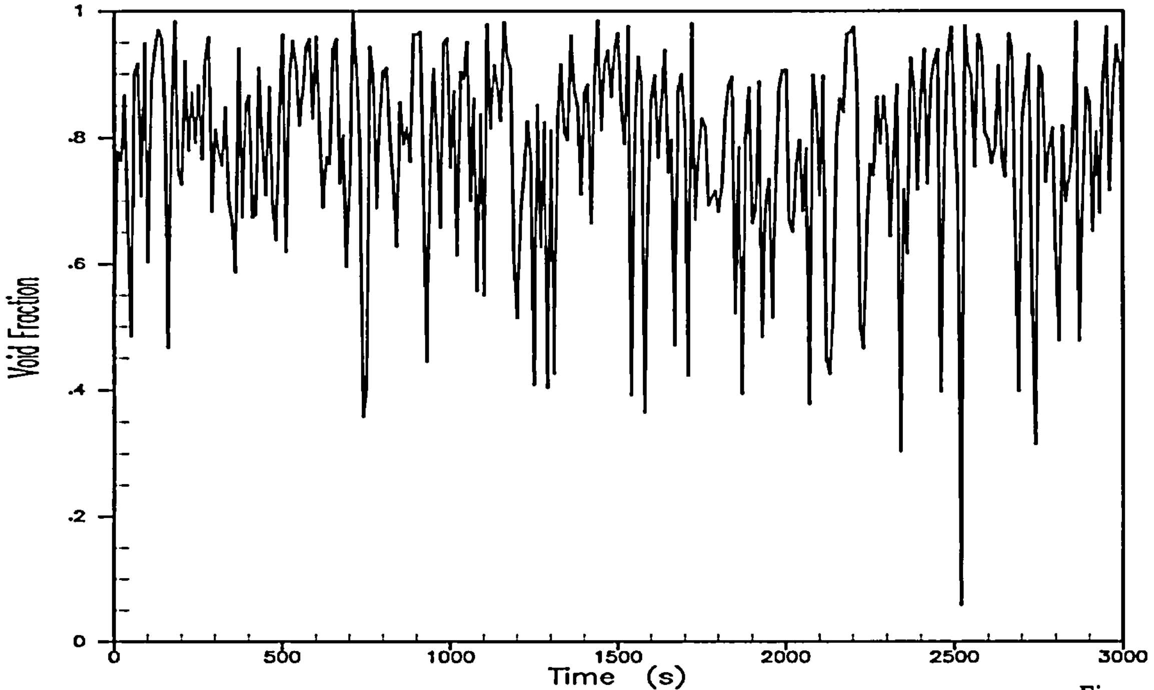


Figure A3.5-3

LTCC DEDVI Break Success Criteria – Void Fraction in Core Cell Level 16 of 17

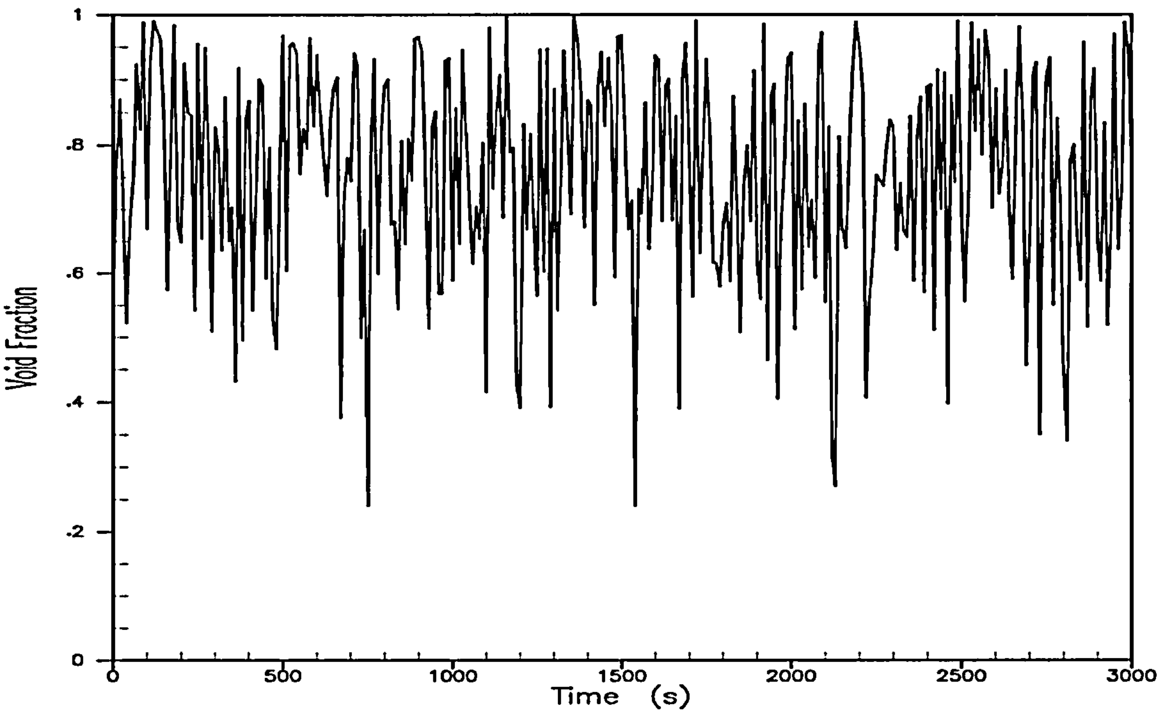


Figure A3.5-4

**LTCC DEDVI Break Success Criteria –
Void Fraction in Core Cell Level 17 of 17**

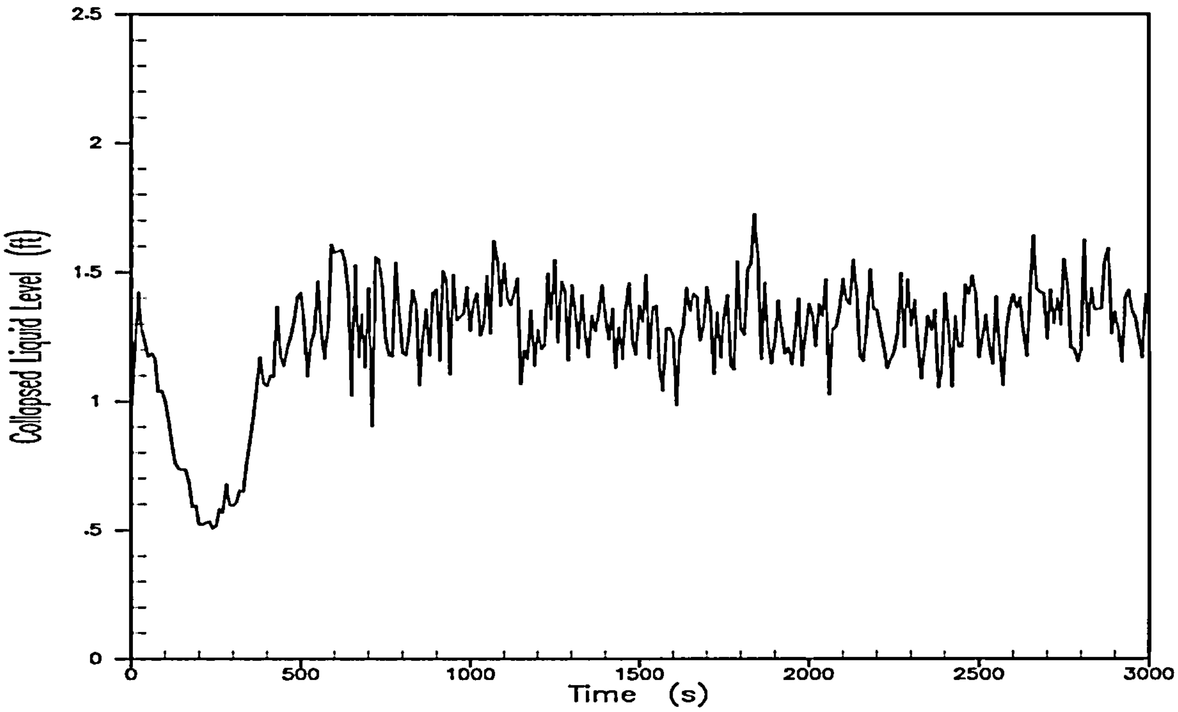


Figure A3.5-5

LTCC DEDVI Break Success Criteria -
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop

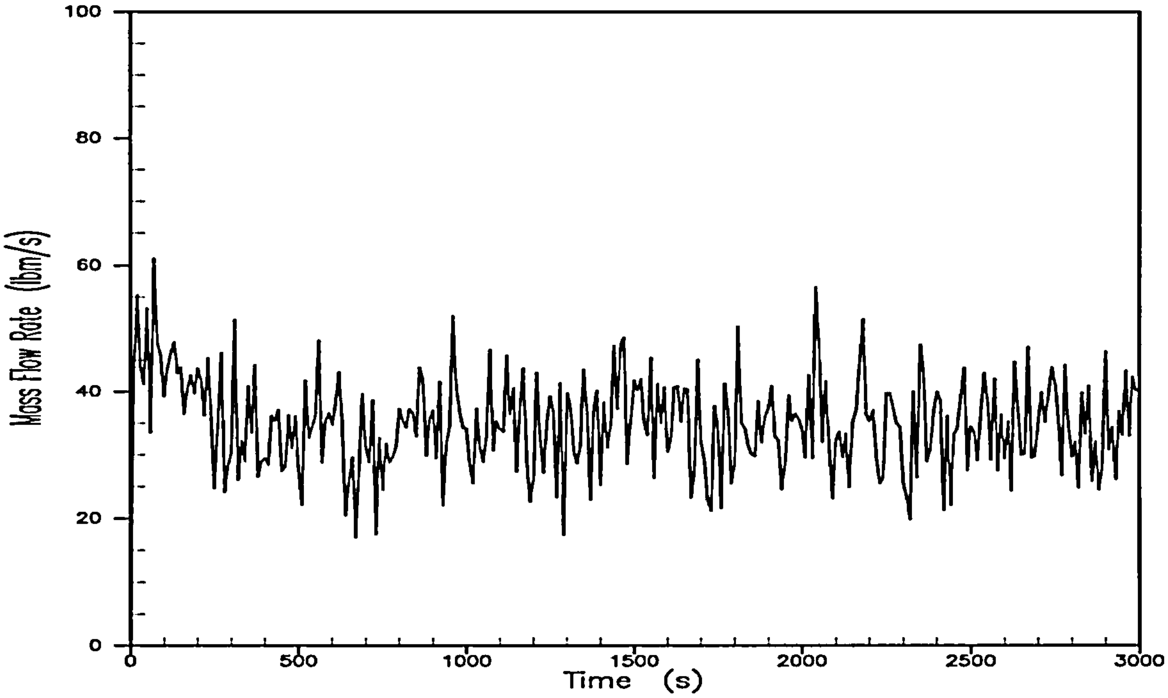


Figure A3.5-6

LTCC DEDVI Break Success Criteria - Vapor Rate Out of Core

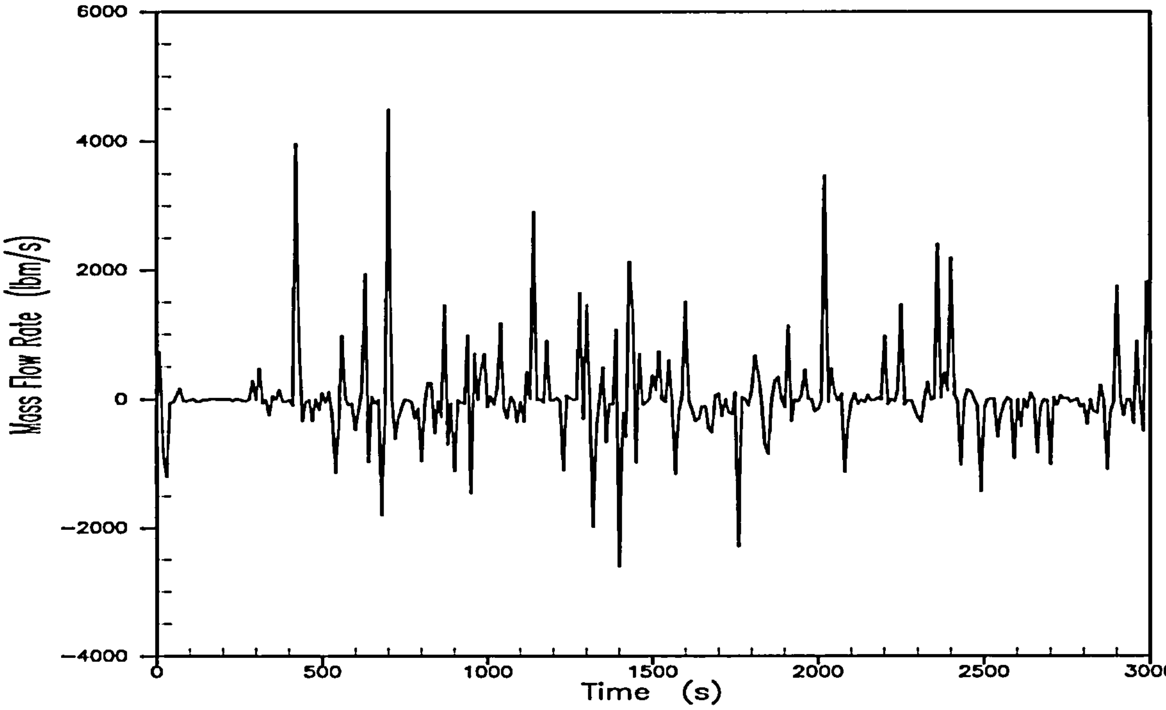


Figure A3.5-7

LTCC DEDVI Break Success Criteria – Liquid Flow Rate Out of the Core

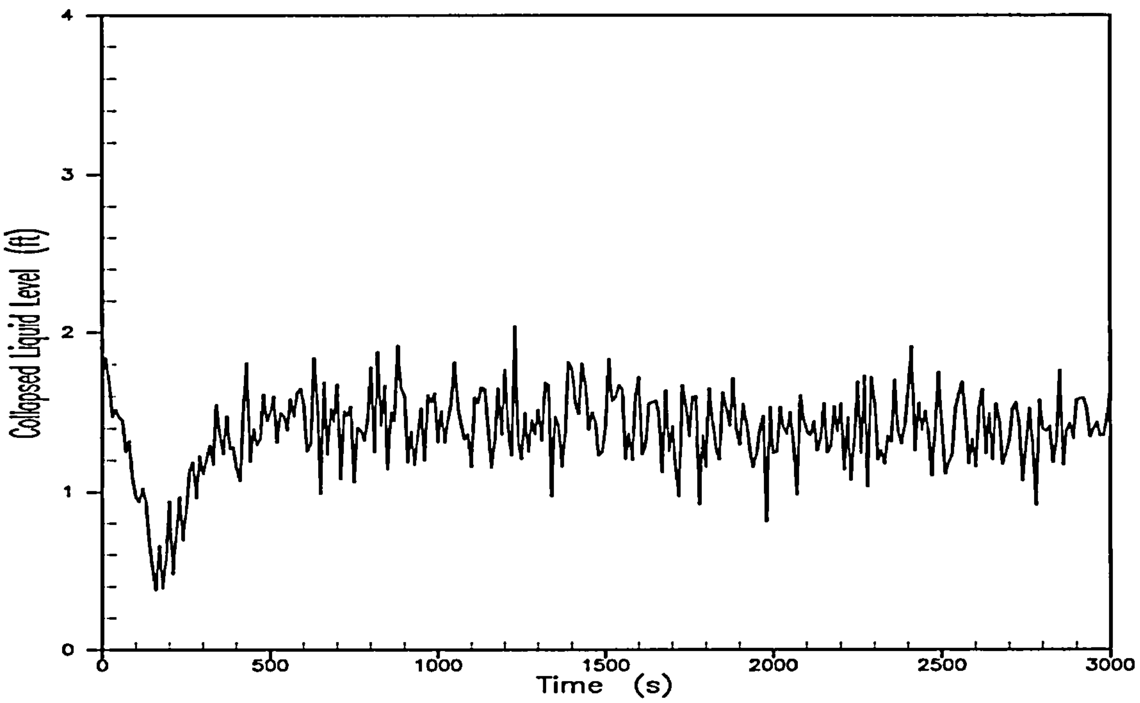


Figure A3.5-8

LTCC DEDVI Break Success Criteria –
Collapsed Liquid Level in the Upper Plenum

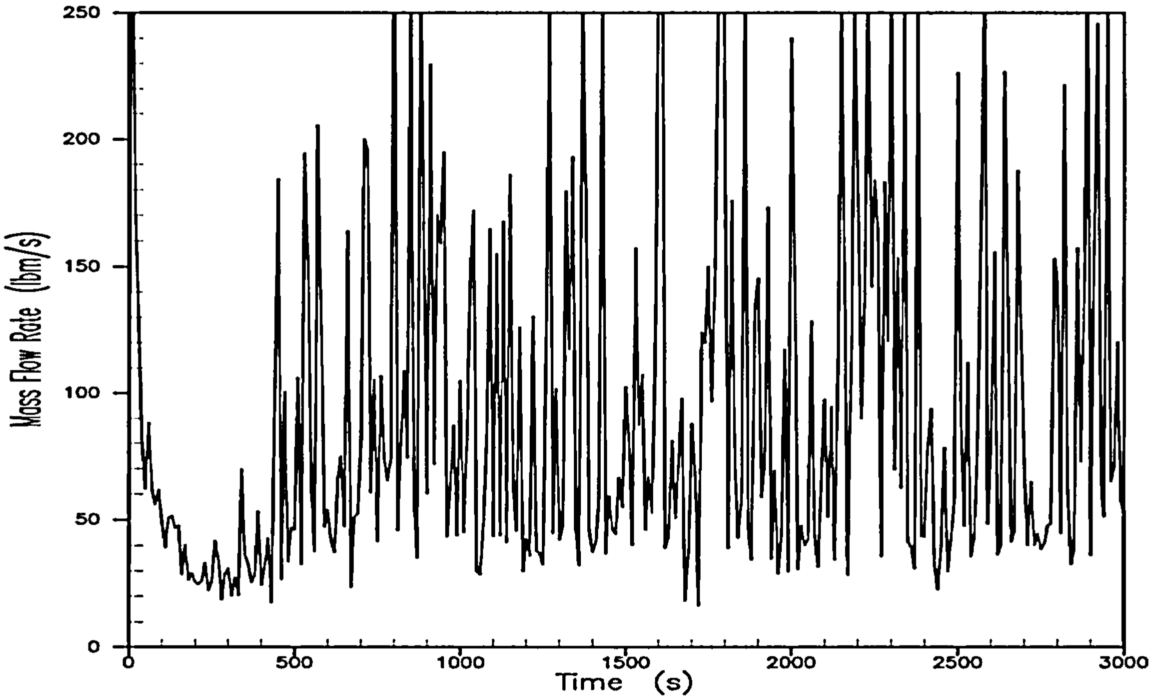


Figure A3.5-9

LTCC DEDVI Break Success Criteria – Mixture Flowrate Through ADS Stage 4A Valves

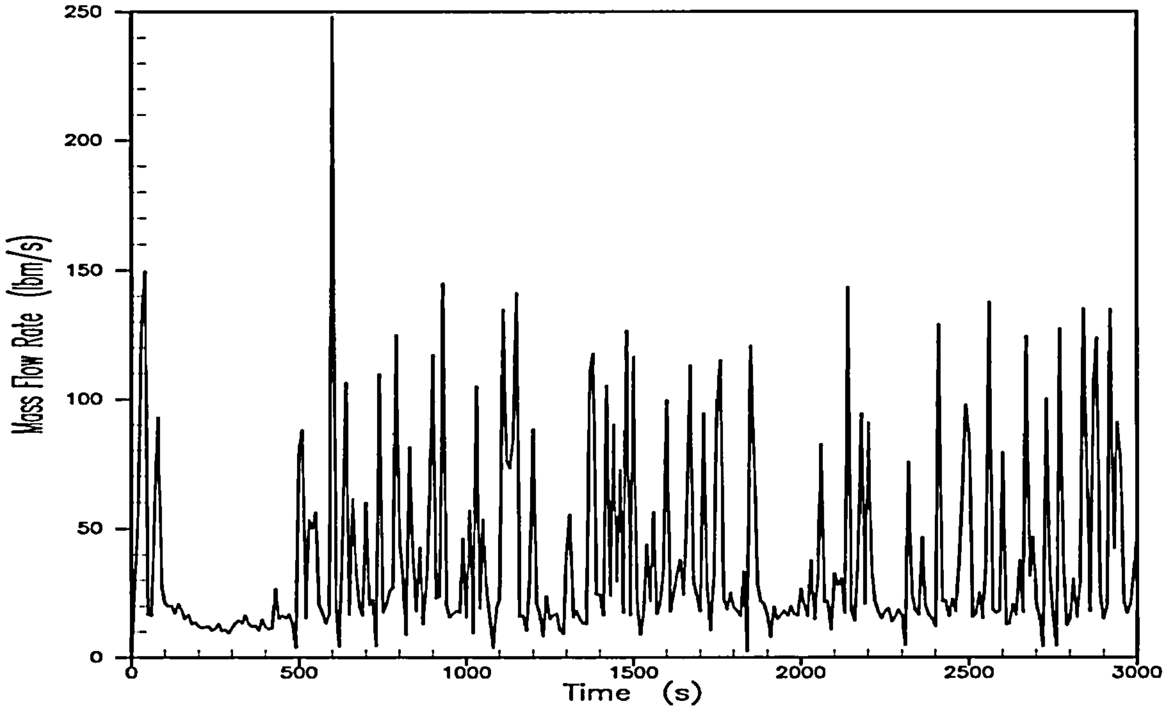


Figure A3.5-10

LTCC DEDVI Break Success Criteria –
Mixture Flowrate Through ADS Stage 4B Valves

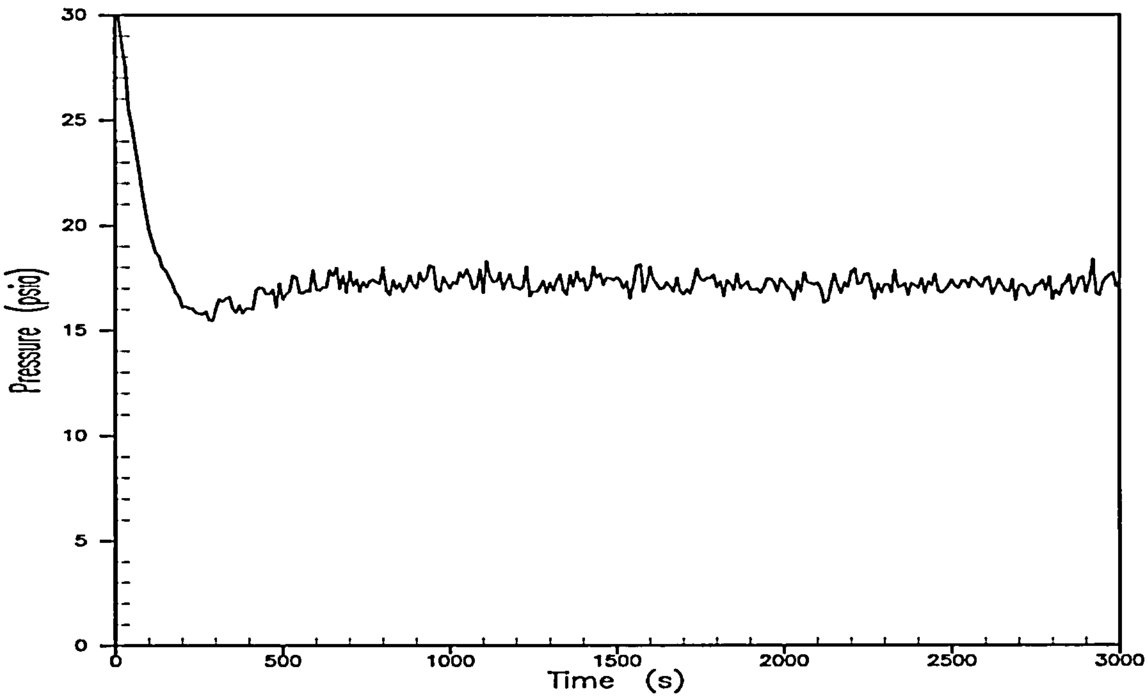


Figure A3.5-11

LTCC DEDVI Break Success Criteria – Upper Plenum Pressure

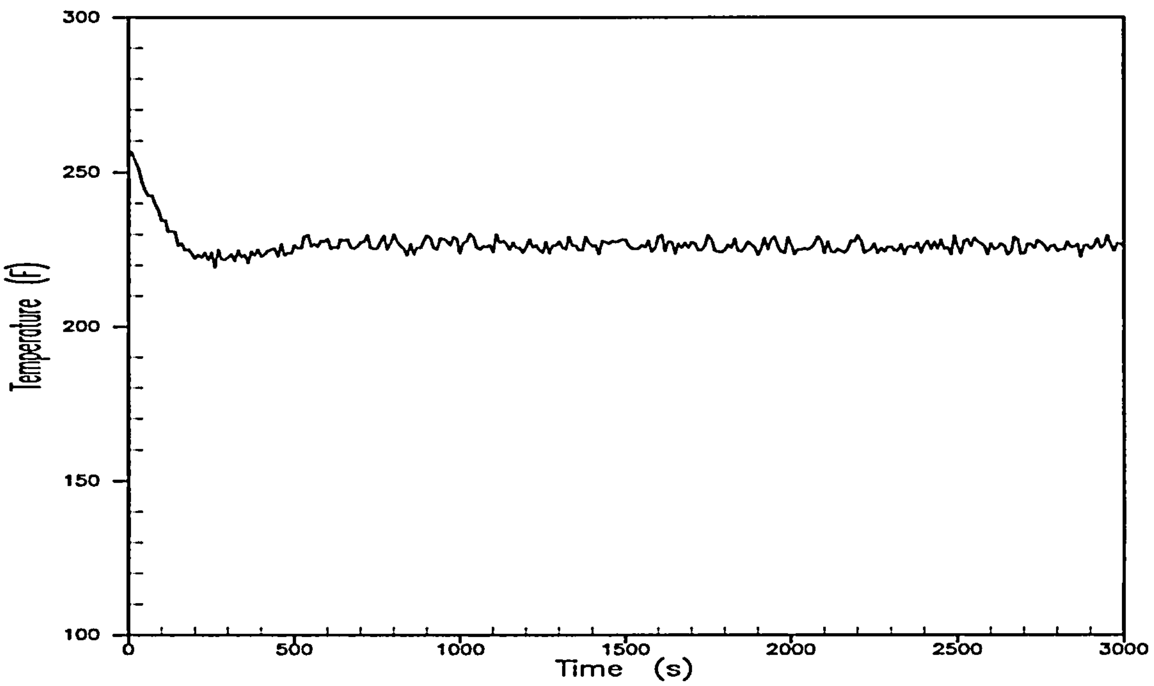


Figure A3.5-12

LTCC DEDVI Break Success Criteria – Hot Rod Clad Temperature in Cell 17

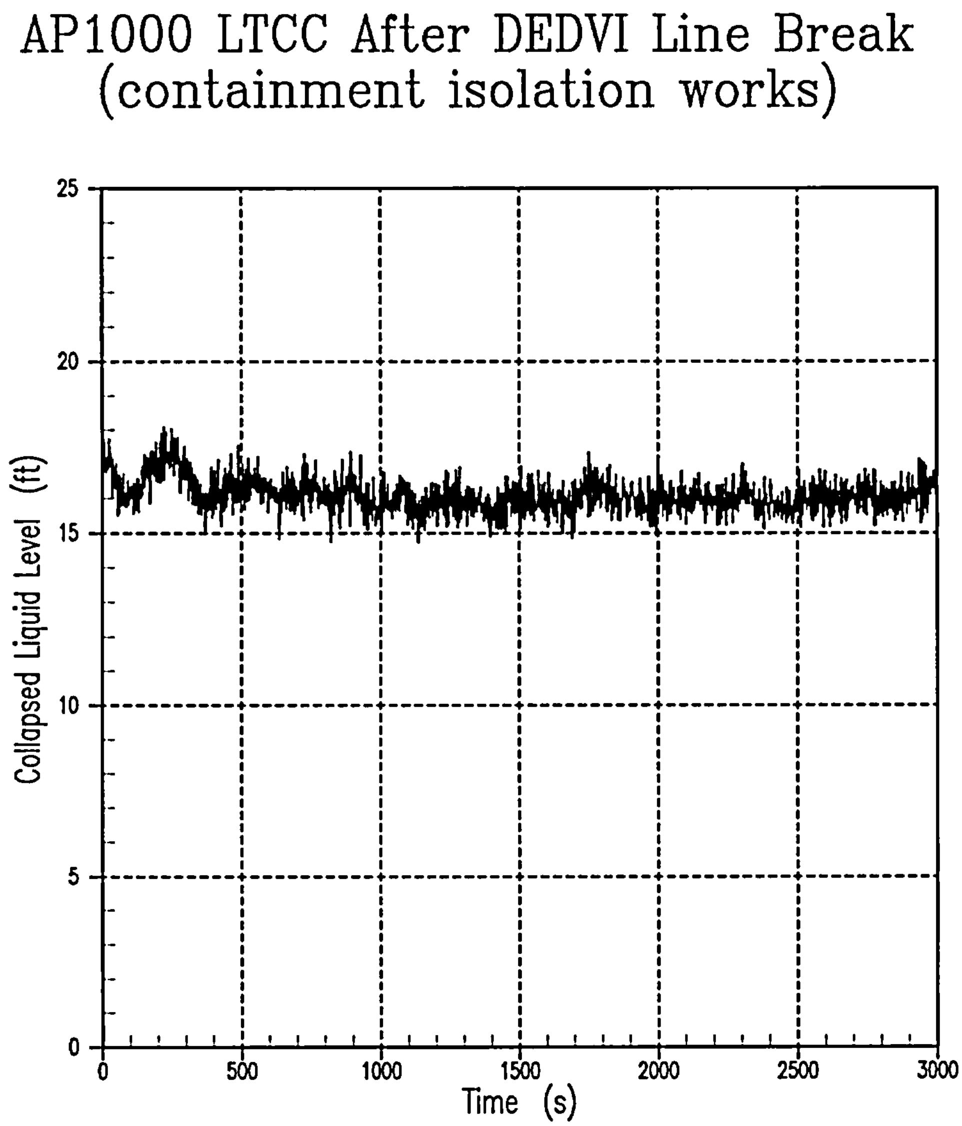


Figure A5.3-1

Case F – Collapsed Level of Liquid in the Downcomer

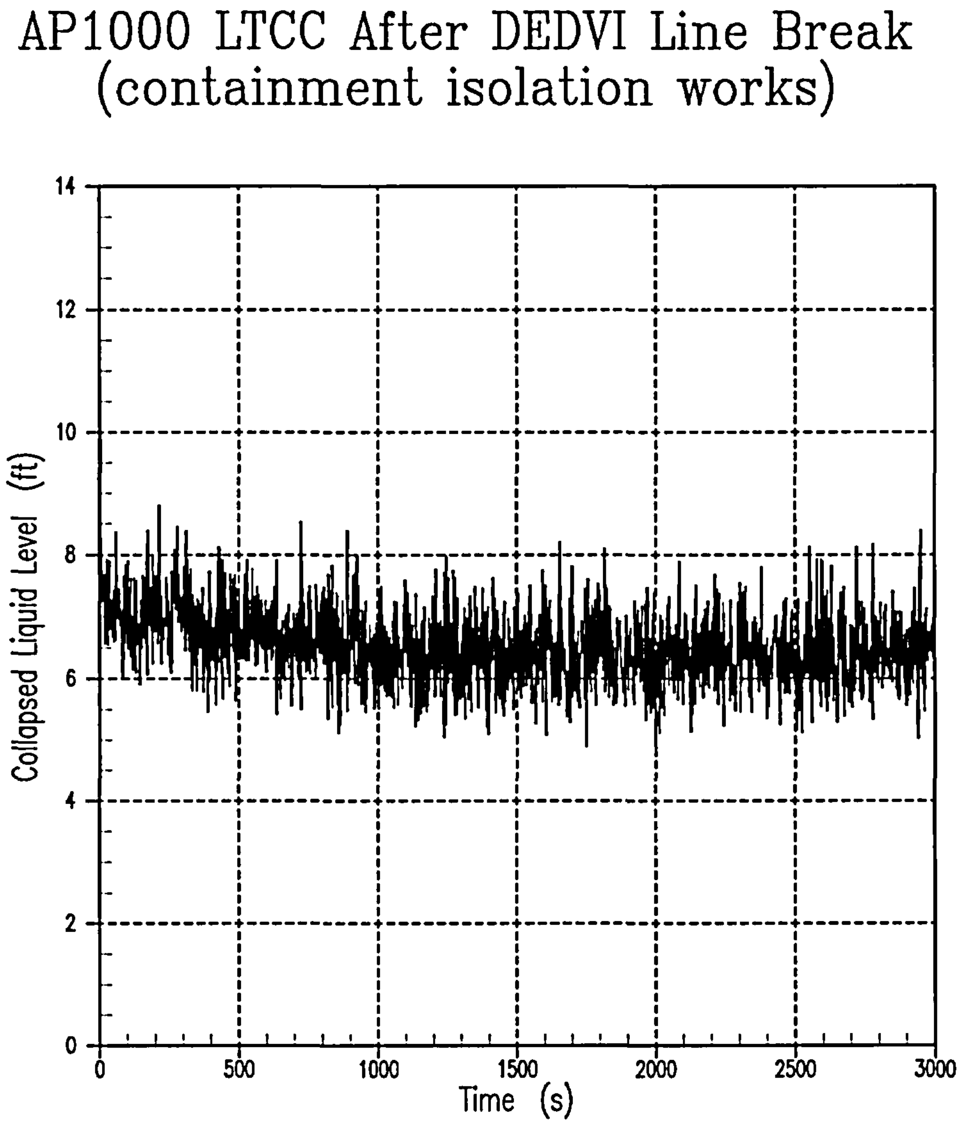


Figure A5.3-2

Case F – Collapsed Level Liquid Over the Heated Length of the Fuel

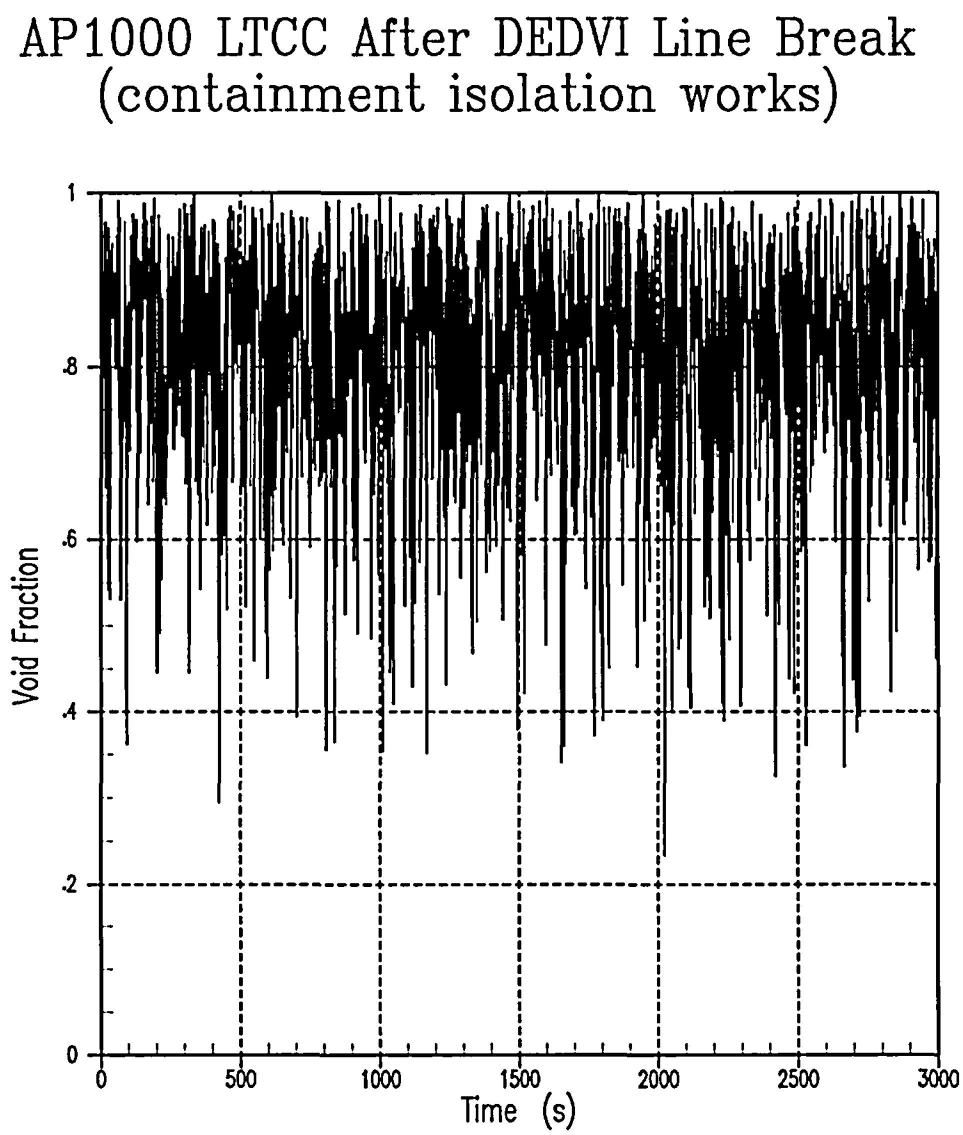


Figure A5.3-3

Case F – Void Fraction in Core Cell Level 16 of 17

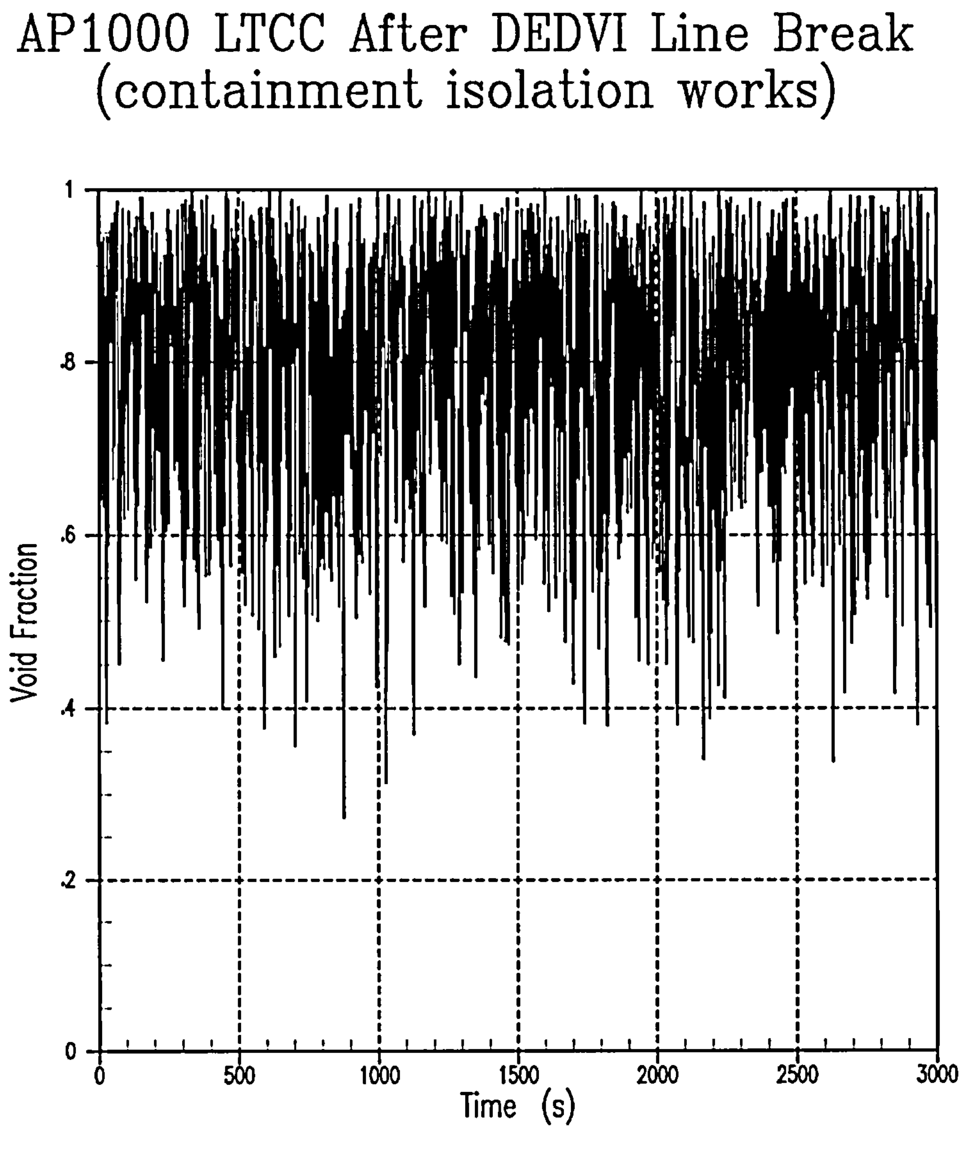


Figure A5.3-4

Case F – Void Fraction in Core Cell Level 17 of 17

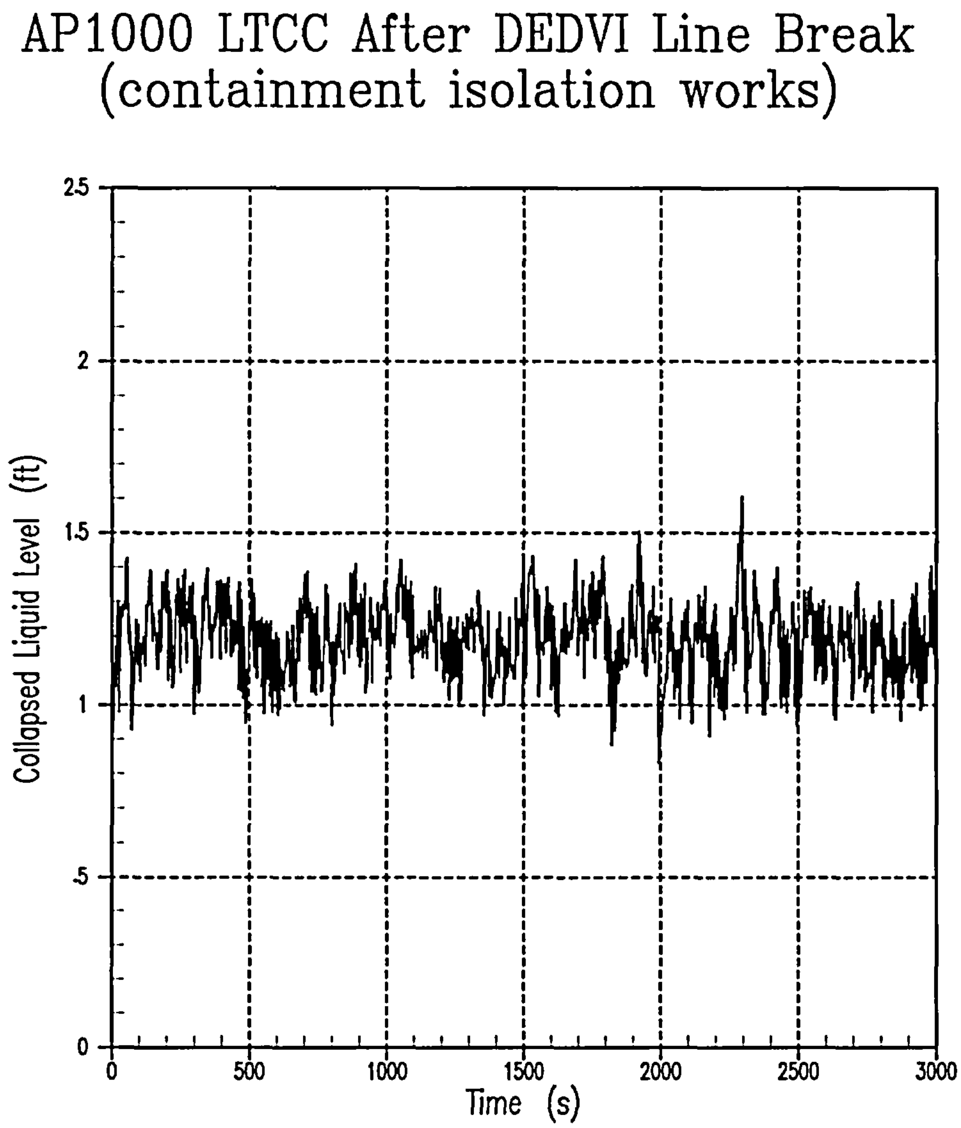


Figure A5.3-5

Case F – Collapsed Liquid Level in the Hot Leg of Pressurizer Loop

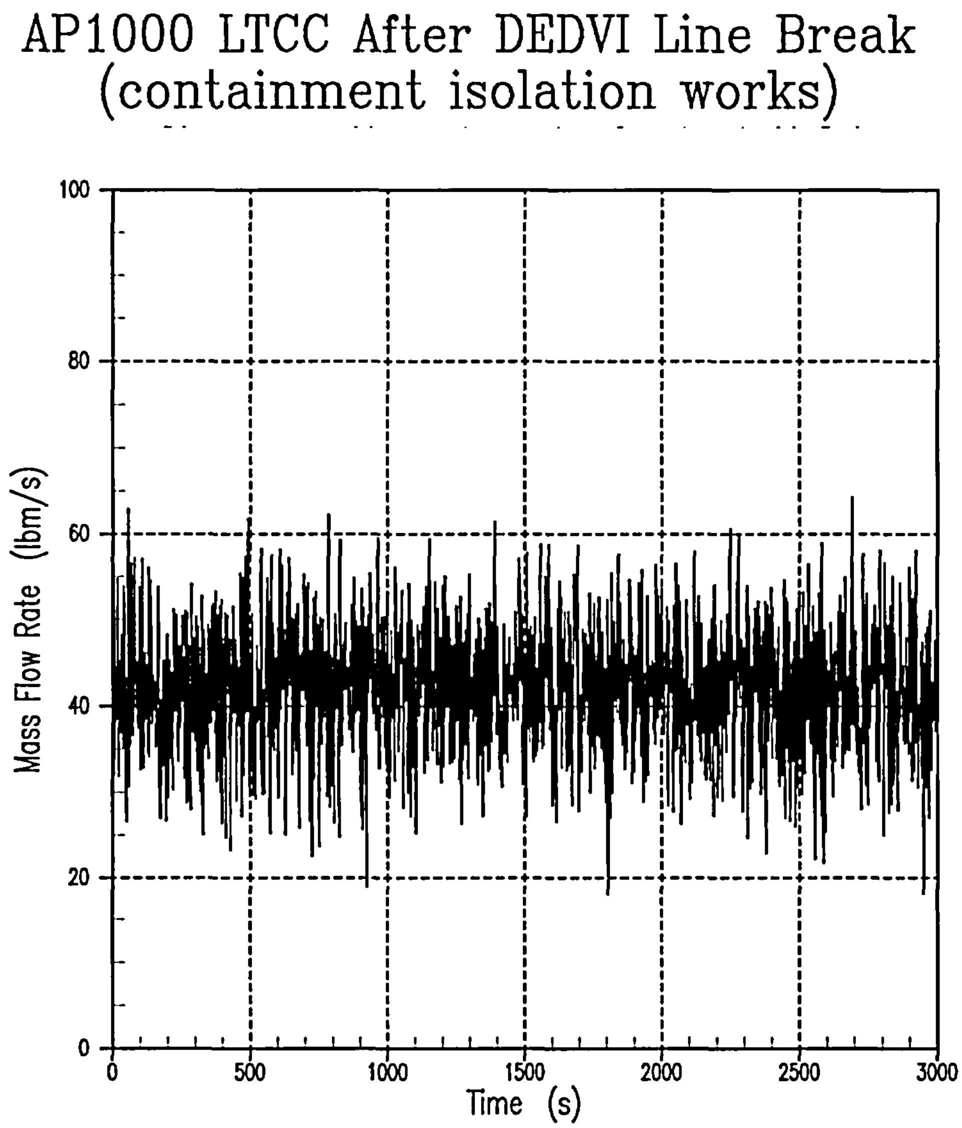


Figure A5.3-6

Case F – Vapor Rate out of the Core

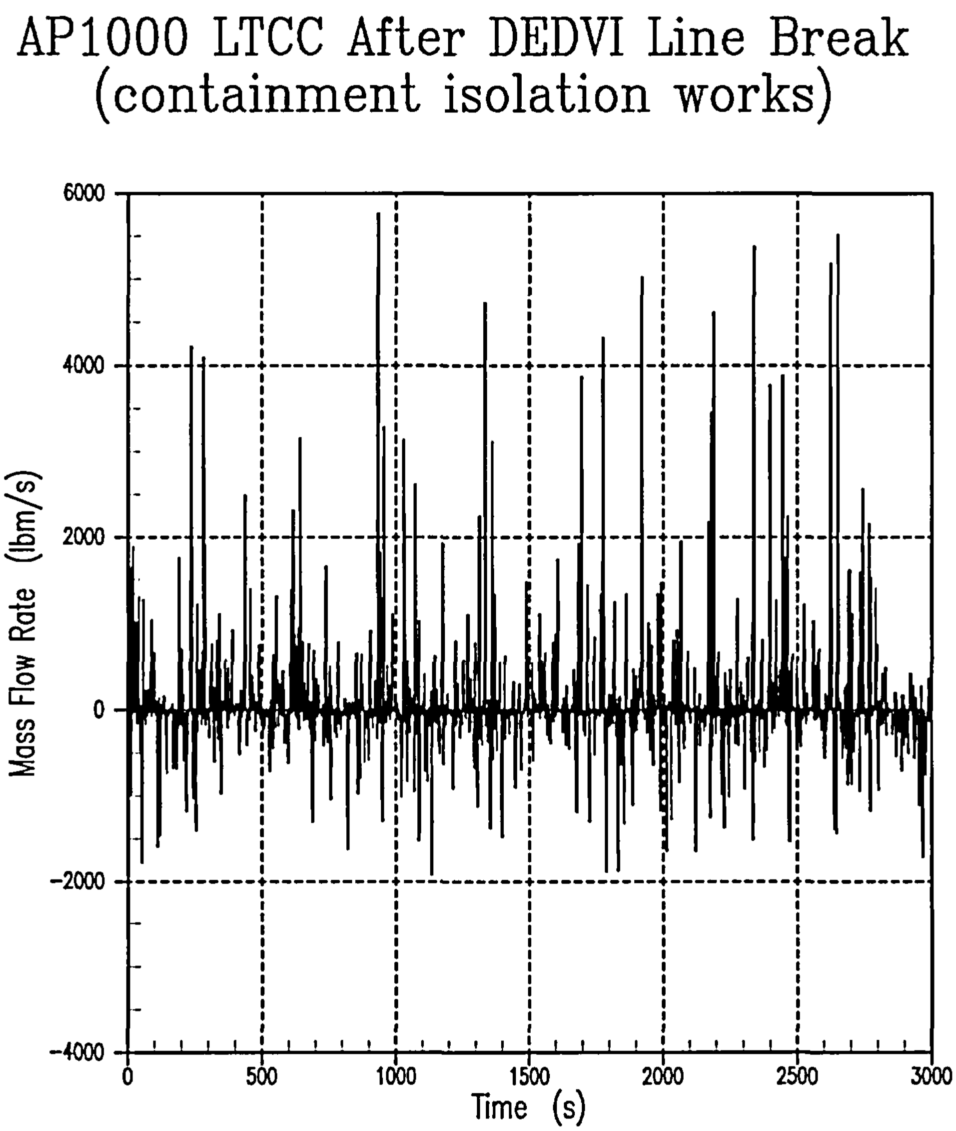


Figure A5.3-7

Case F – Liquid Flow Rate Out of the Core

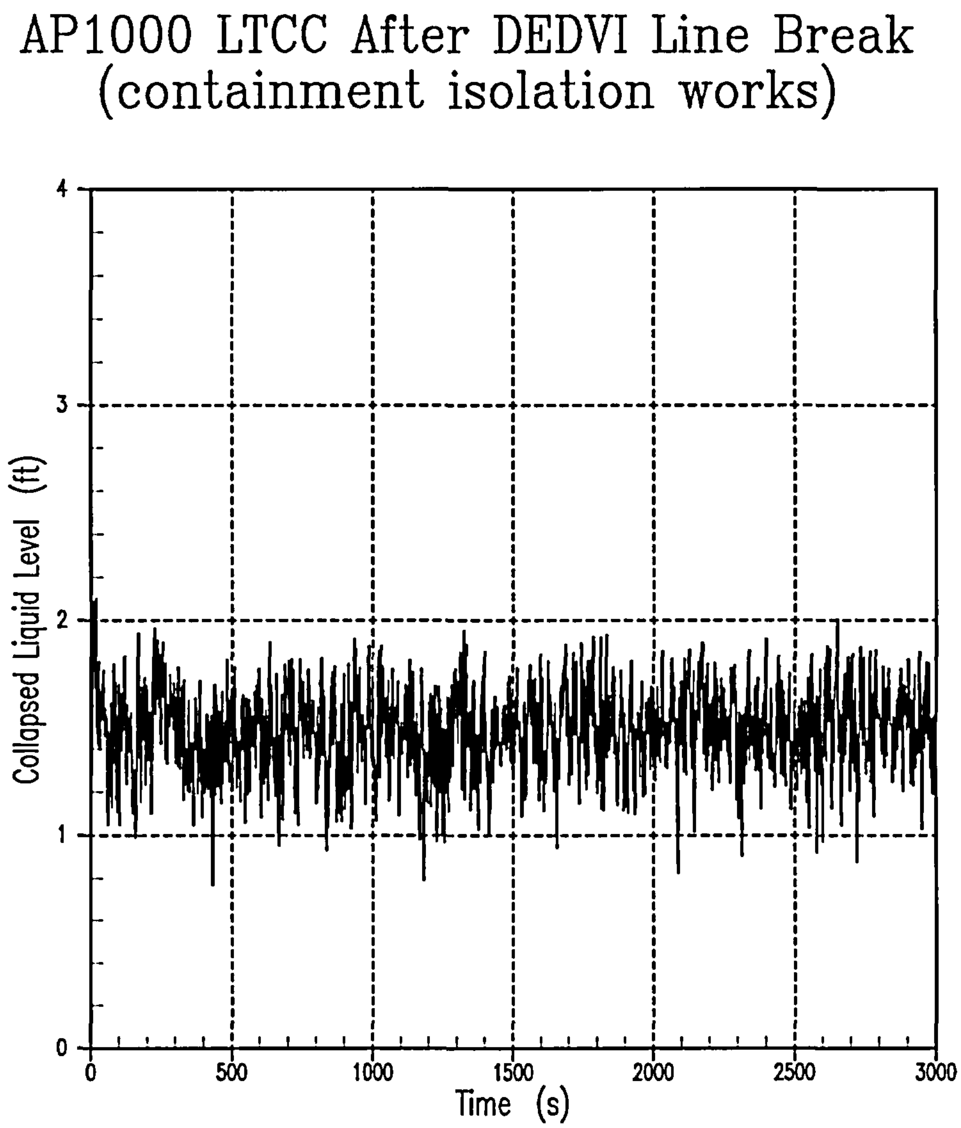


Figure A5.3-8

Case F – Collapsed Liquid Level in the Upper Plenum

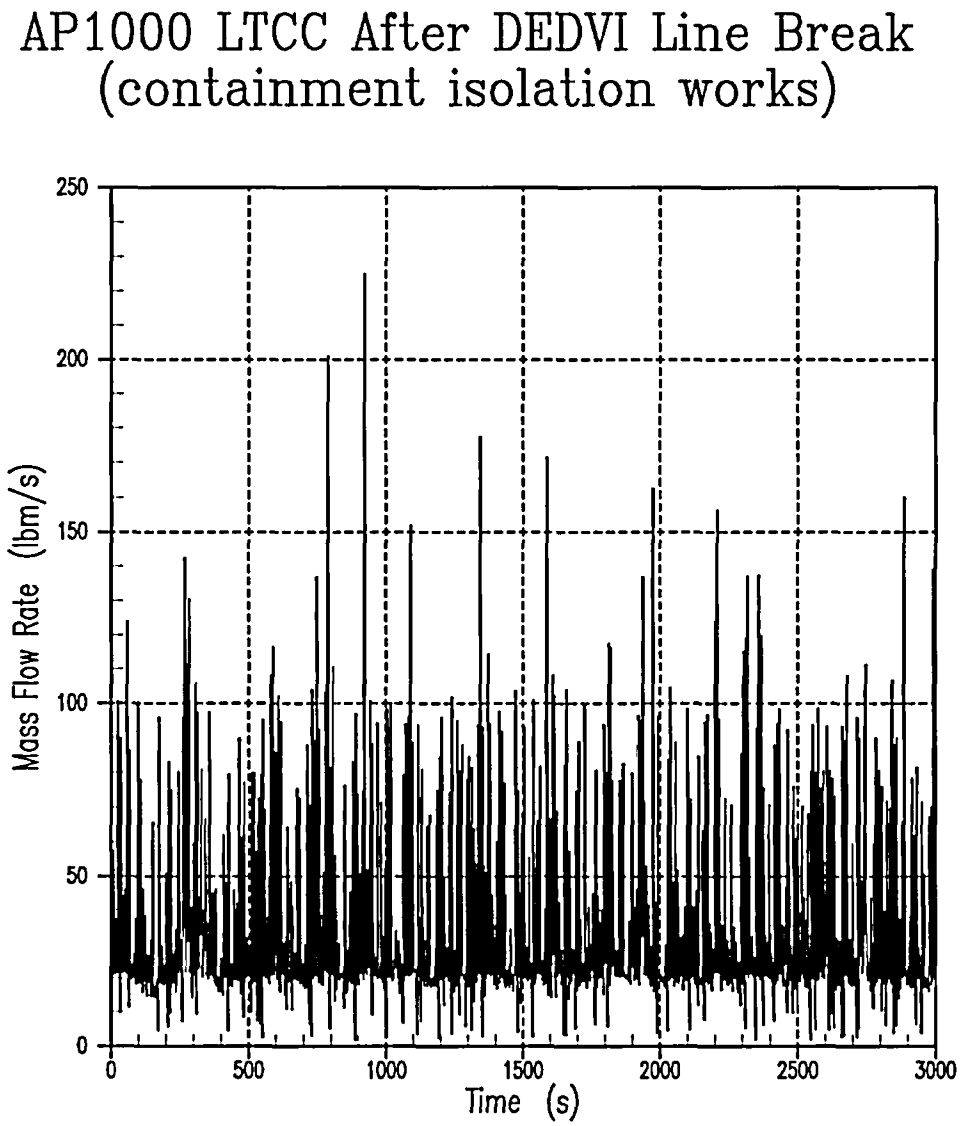


Figure A5.3-9

Case F – Mixture Flowrate Through ADS Stage 4A Valves

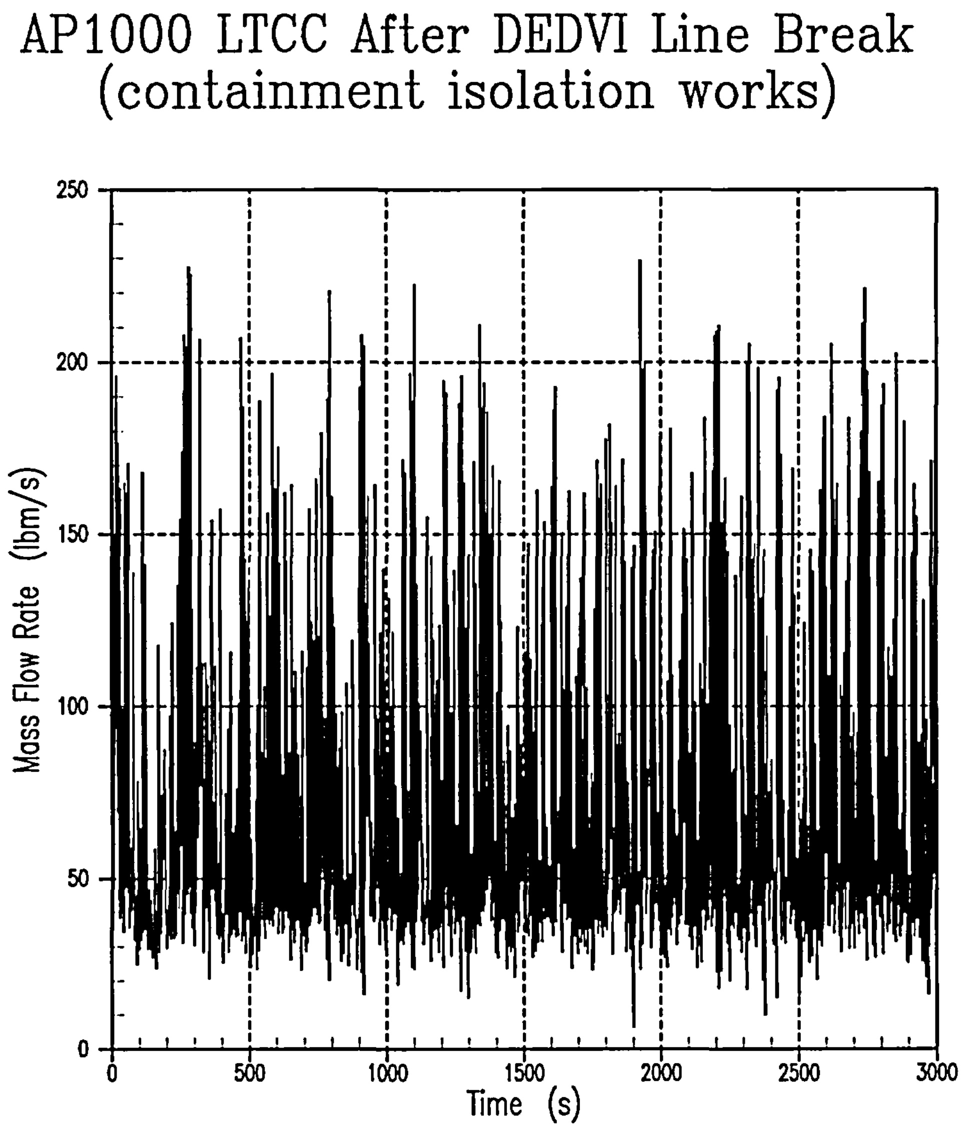


Figure A5.3-10

Case F – Mixture Flowrate Through ADS Stage 4B Valves

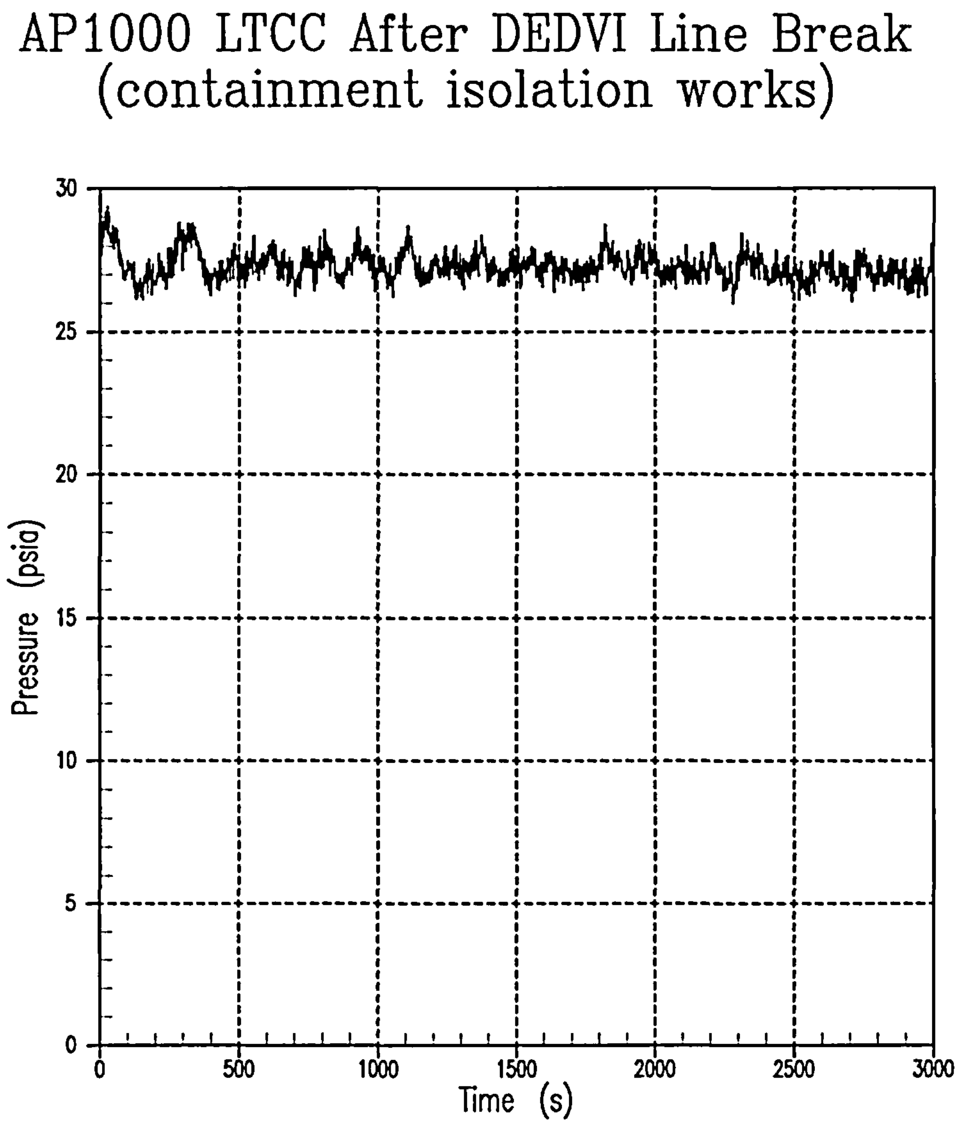


Figure A5.3-11

Case F – Upper Plenum Pressure

AP1000 LTCC After DEDVI Line Break
(containment isolation works)

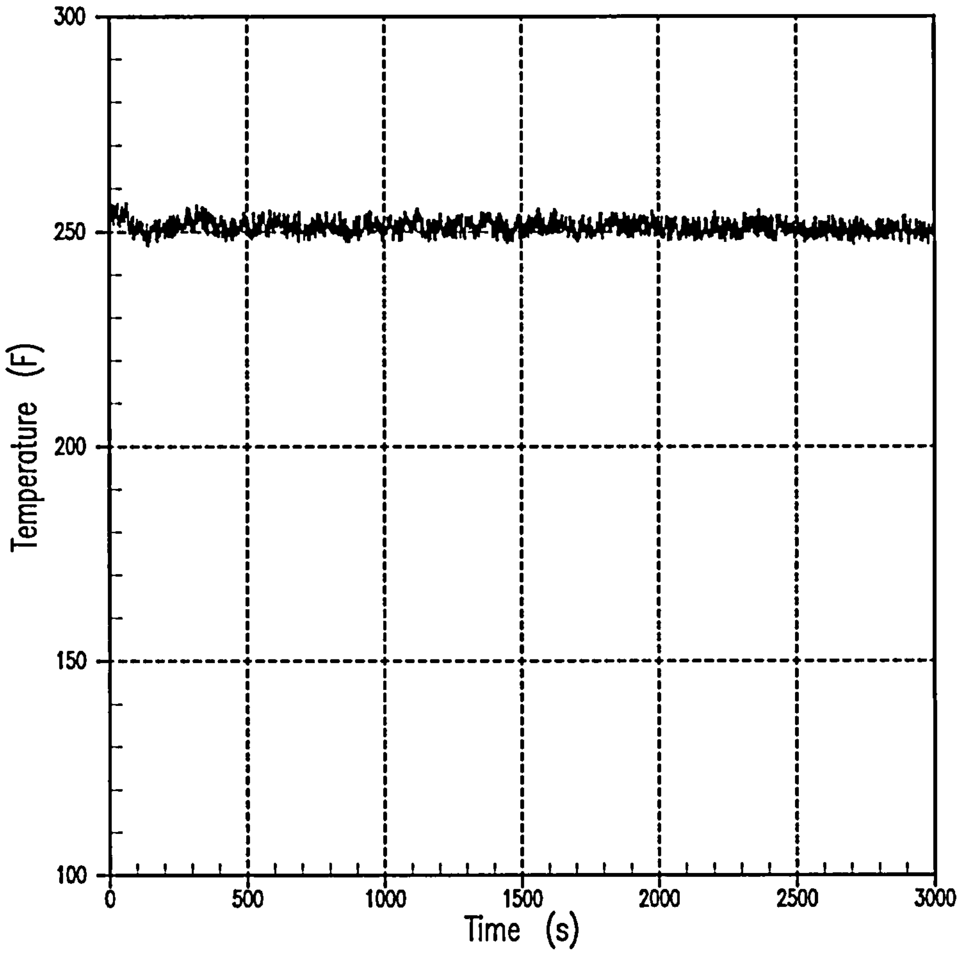


Figure A5.3-12

Case F – Hot Rod Clad Temperature in Cell 17

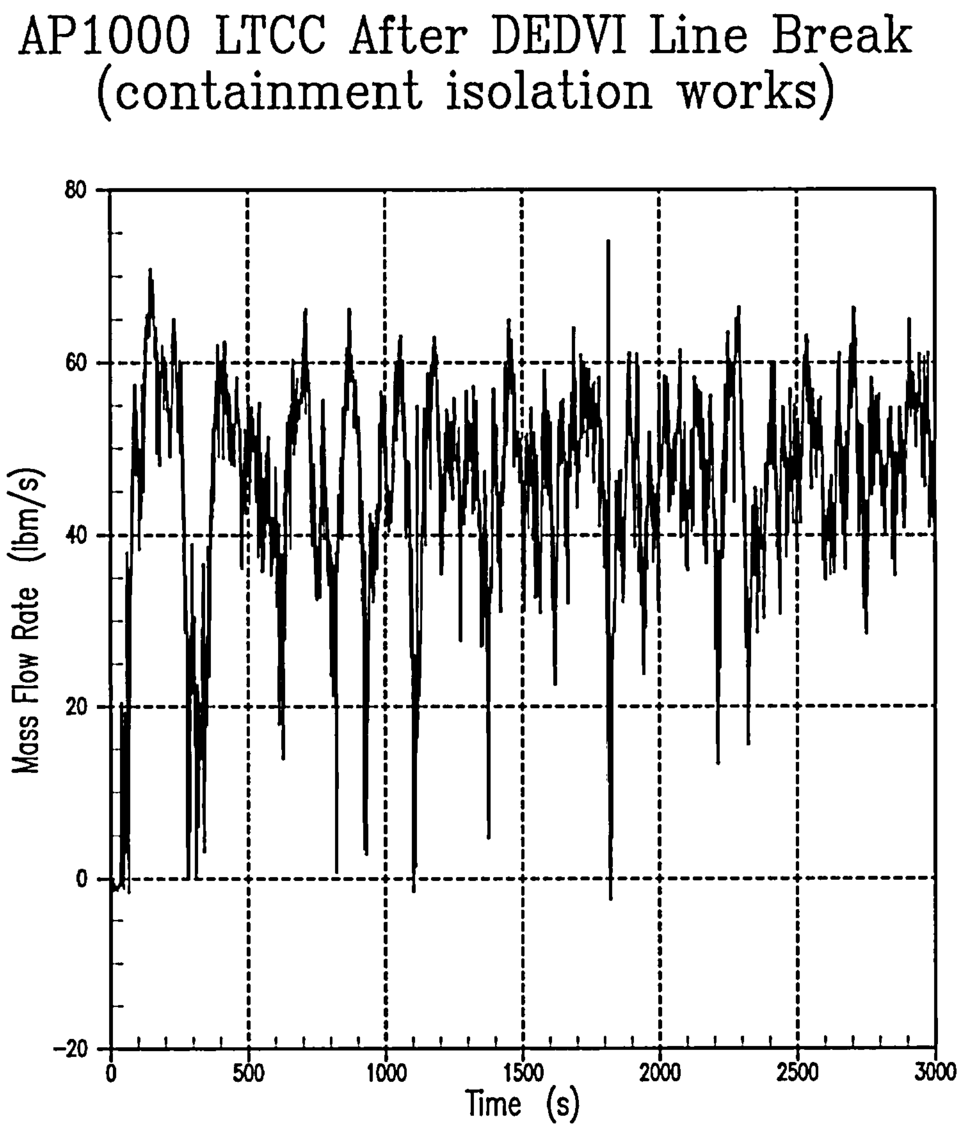


Figure A5.3-13

Case F – DVI-B Mixture Flow Rate

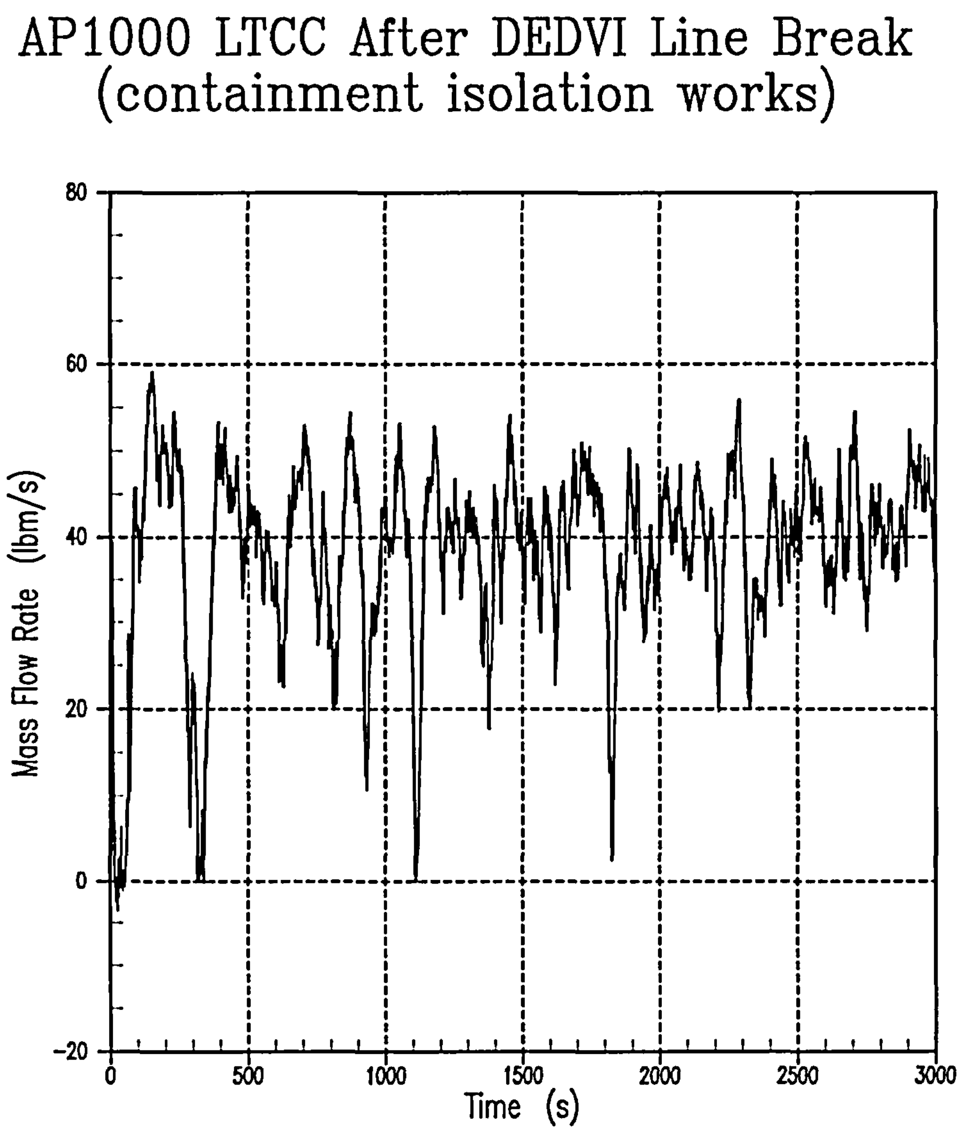


Figure A5.3-14

Case F – DVI-A Mixture Flow Rate

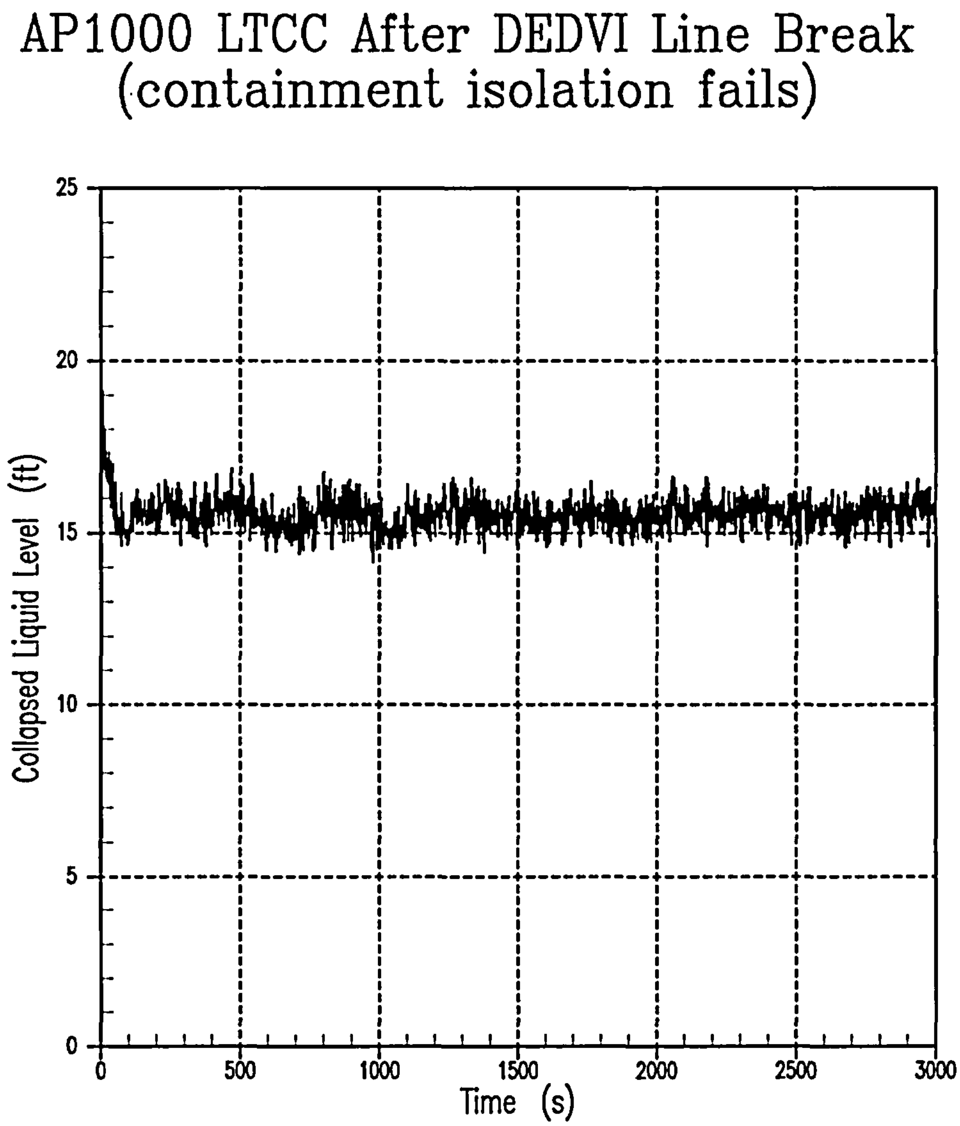


Figure A5.3-15

Case G – Collapsed Level of Liquid in the Downcomer

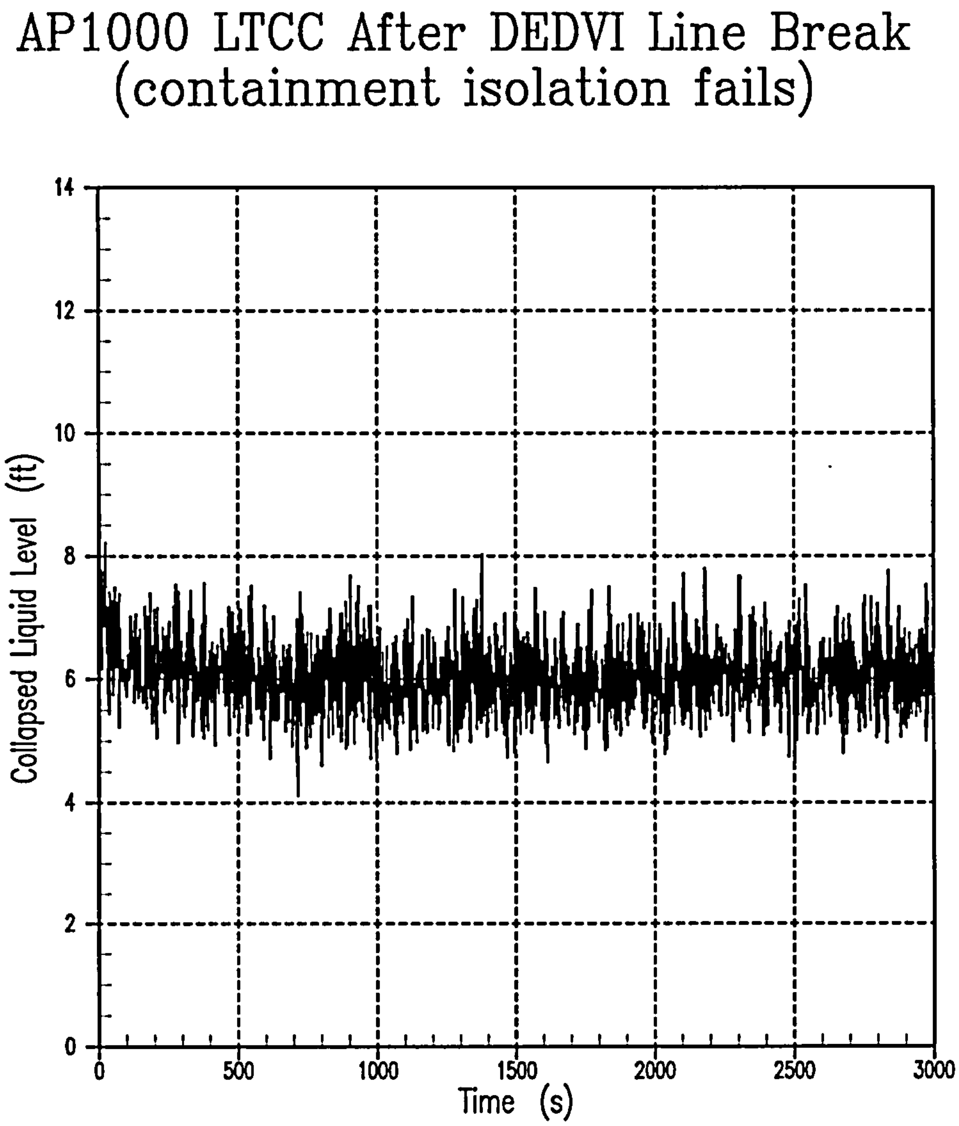


Figure A5.3-16

Case G – Collapsed Level of Liquid Over the Heated Length of the Fuel

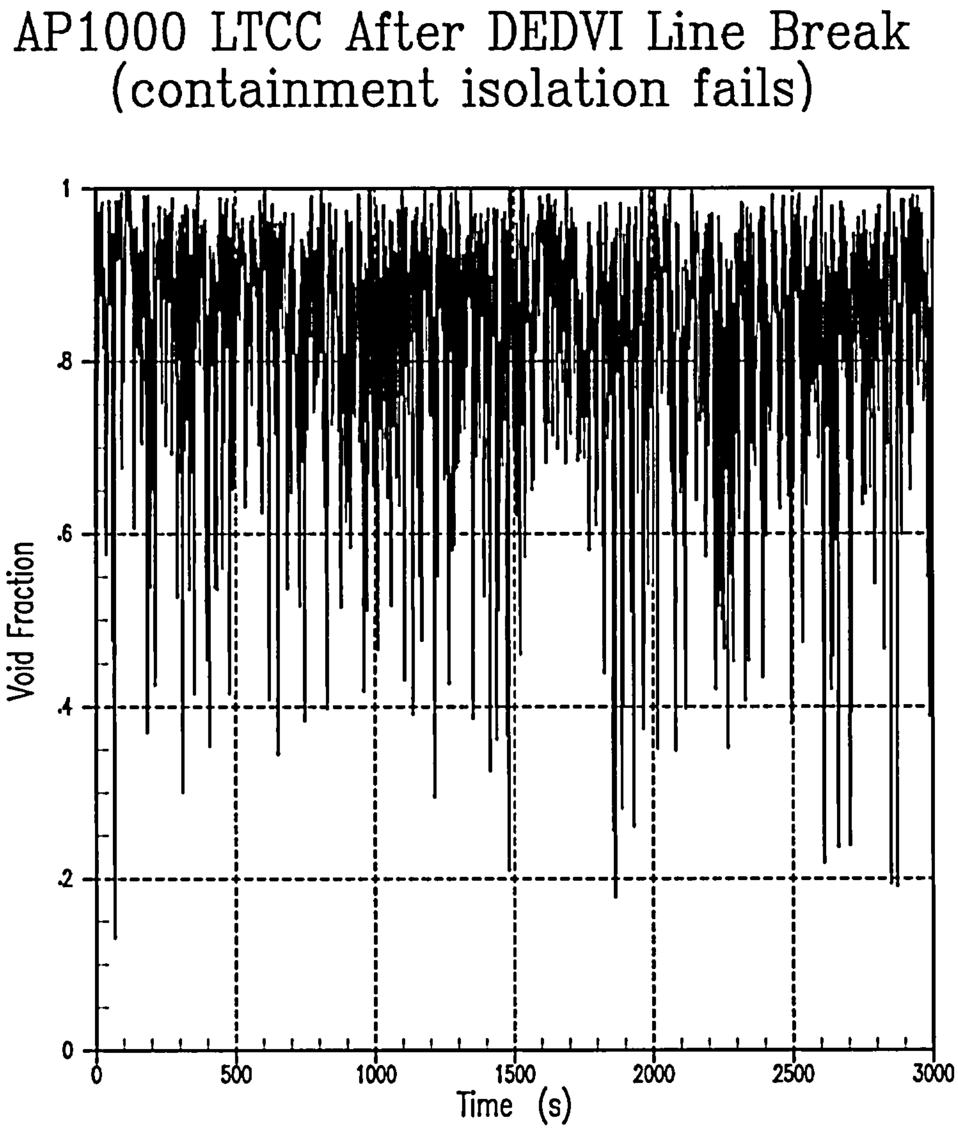


Figure A5.3-17

Case G – Void Fraction in Core Cell Level 16 of 17

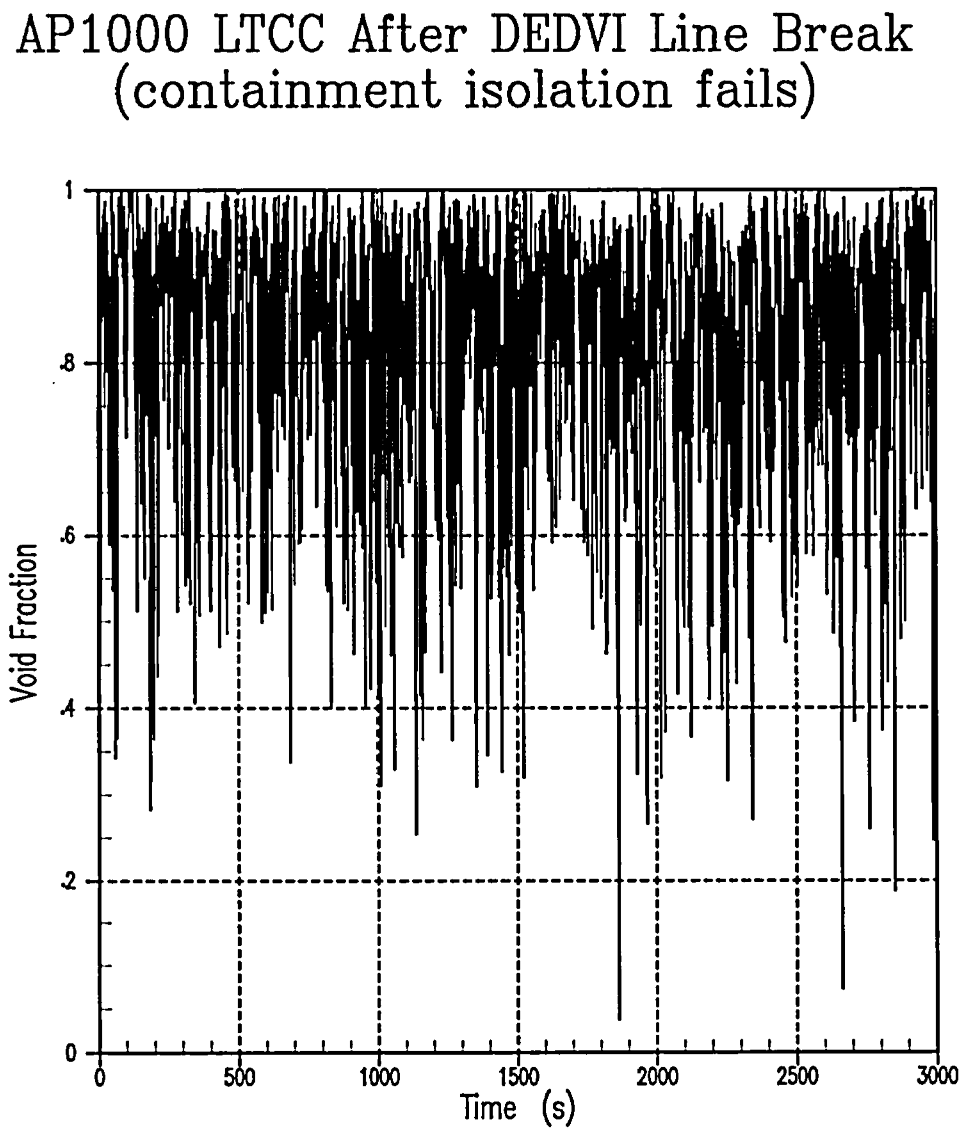


Figure A5.3-18

Case G – Void Fraction in Core Cell Level 17 of 17

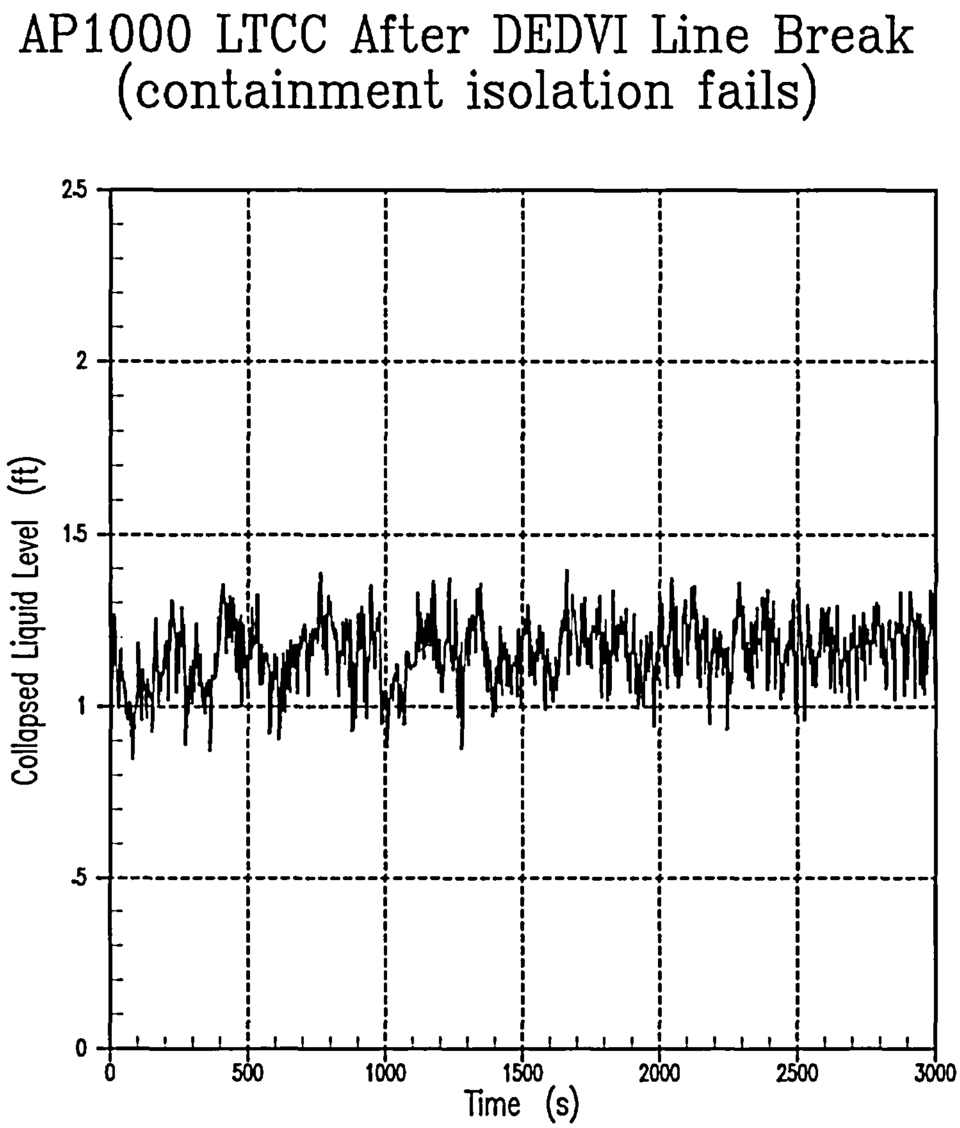


Figure A5.3-19

Case G – Collapsed Liquid Level in the Hot Leg of Pressurizer Loop

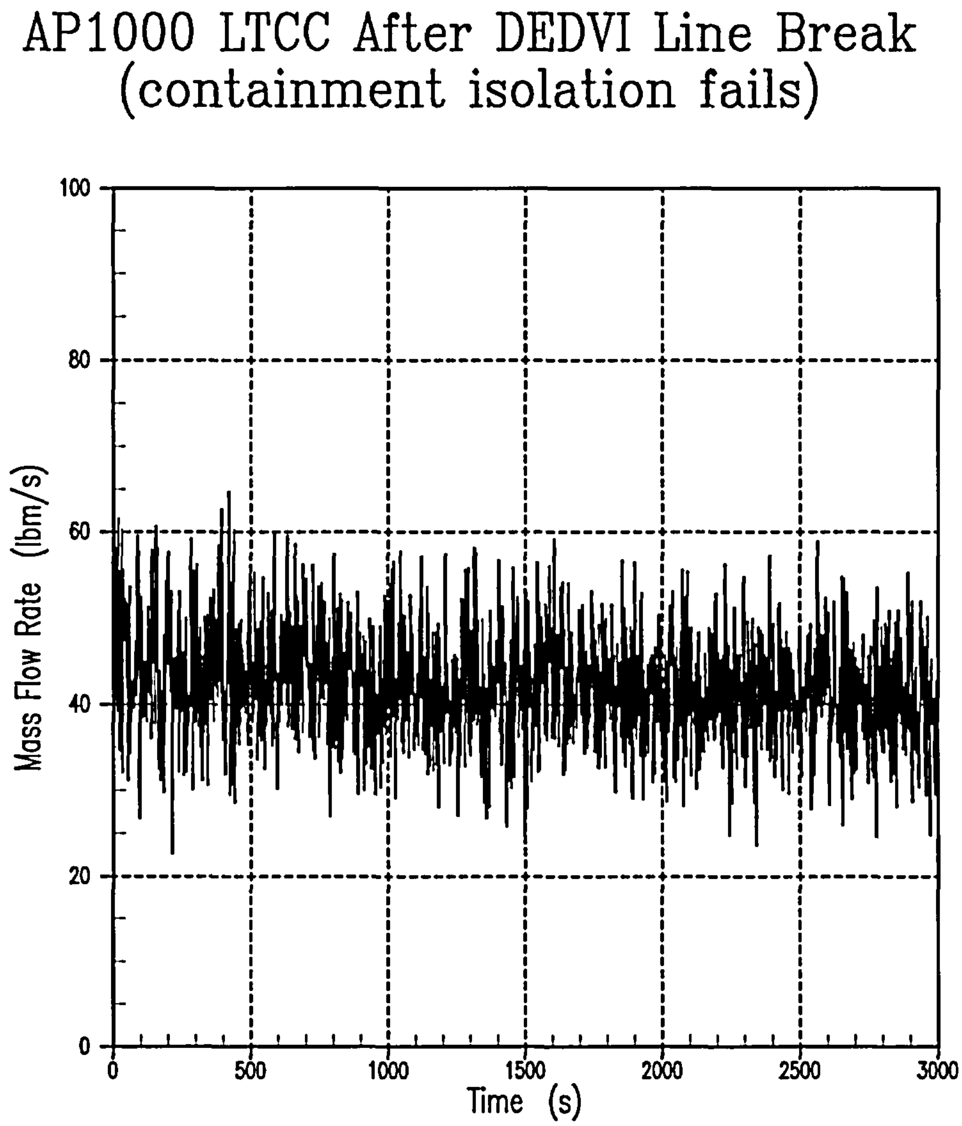


Figure A5.3-20

Case G – Vapor Rate out of the Core

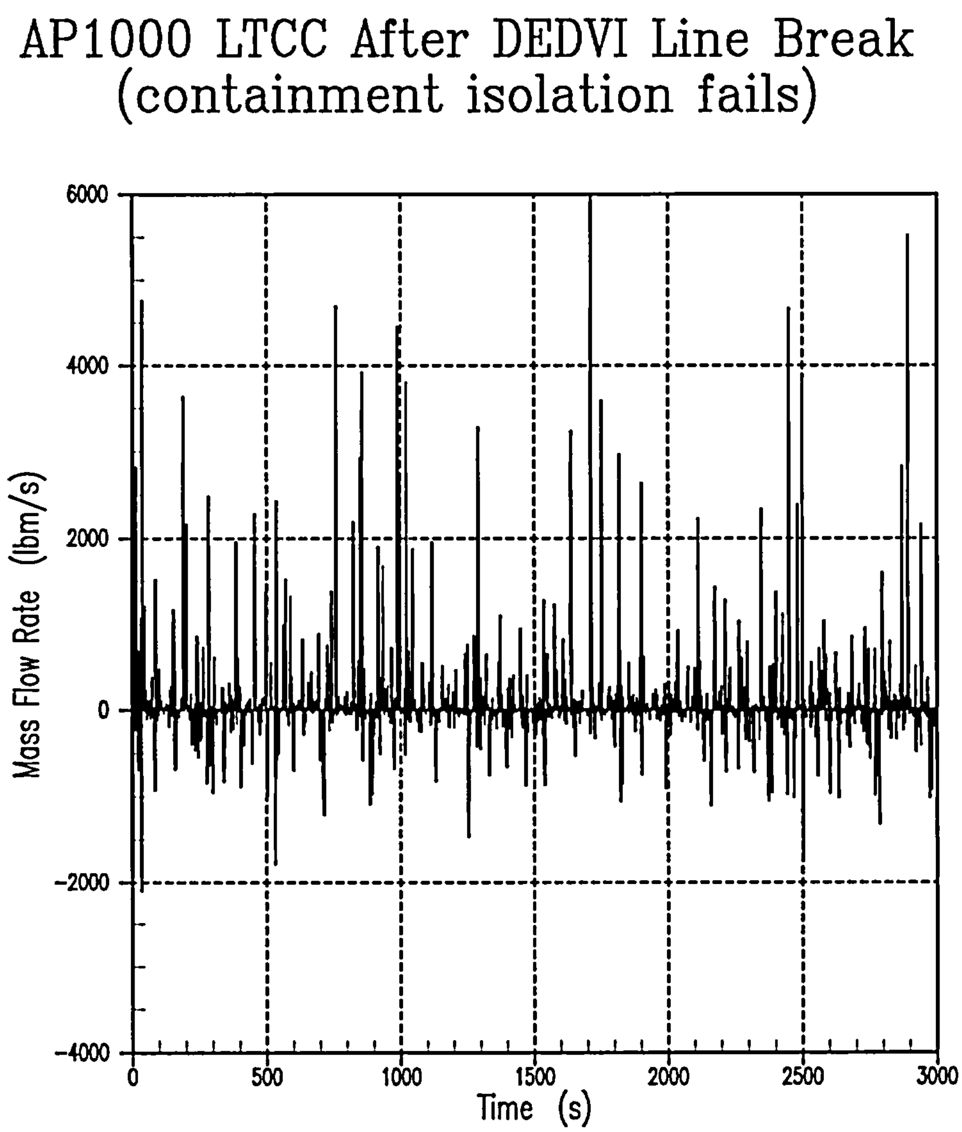


Figure A5.3-21

Case G – Liquid Flow Rate Out of the Core

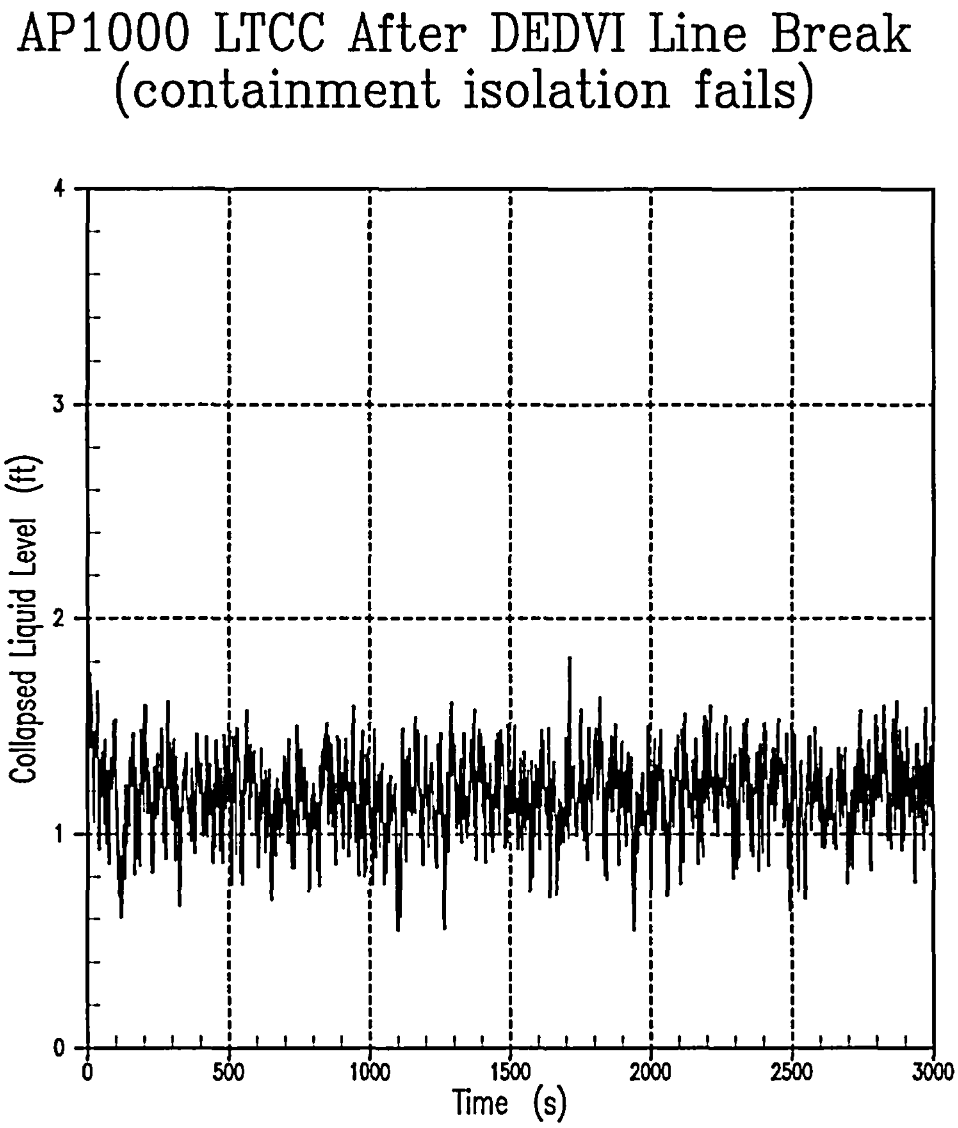


Figure A5.3-22

Case G – Collapsed Liquid Level in the Upper Plenum

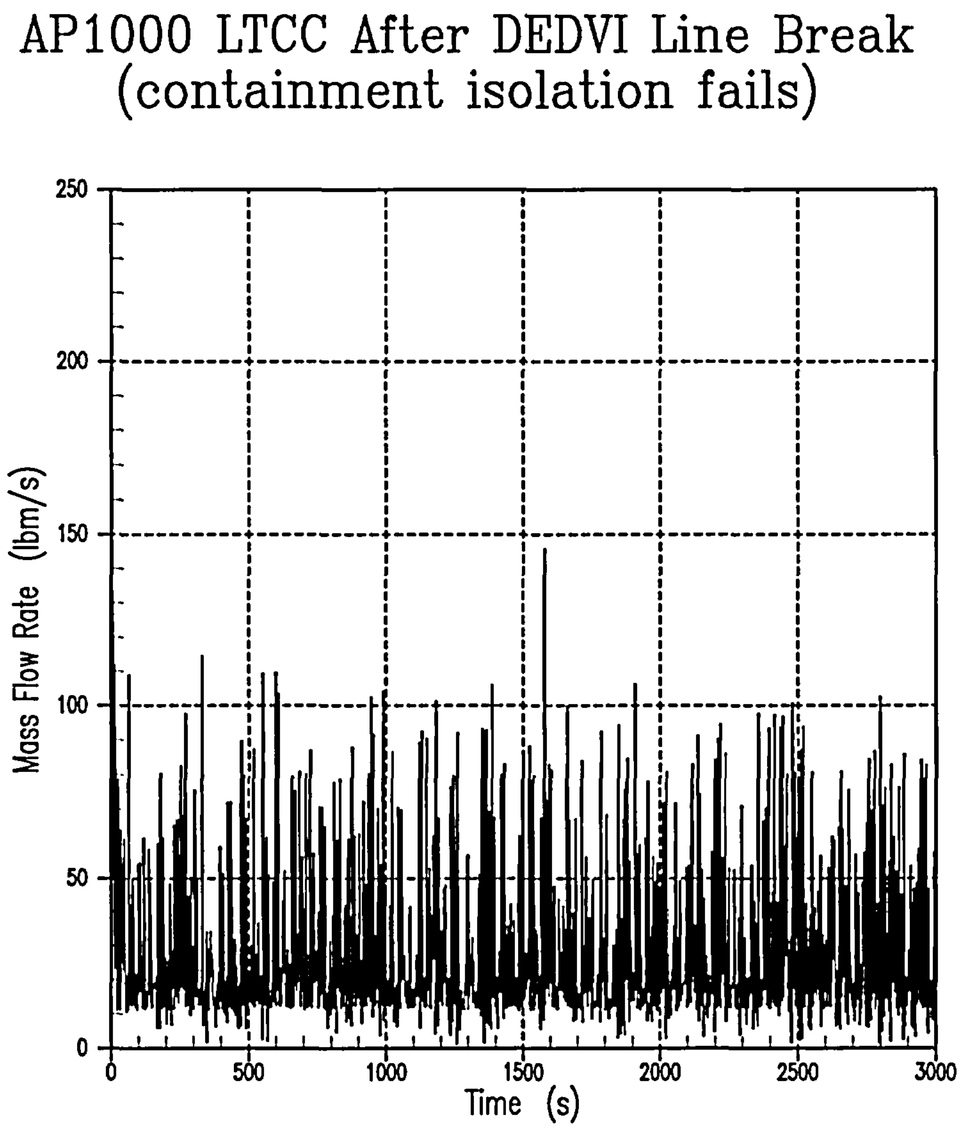


Figure A5.3-23

Case G – Mixture Flowrate Through ADS Stage 4A Valves

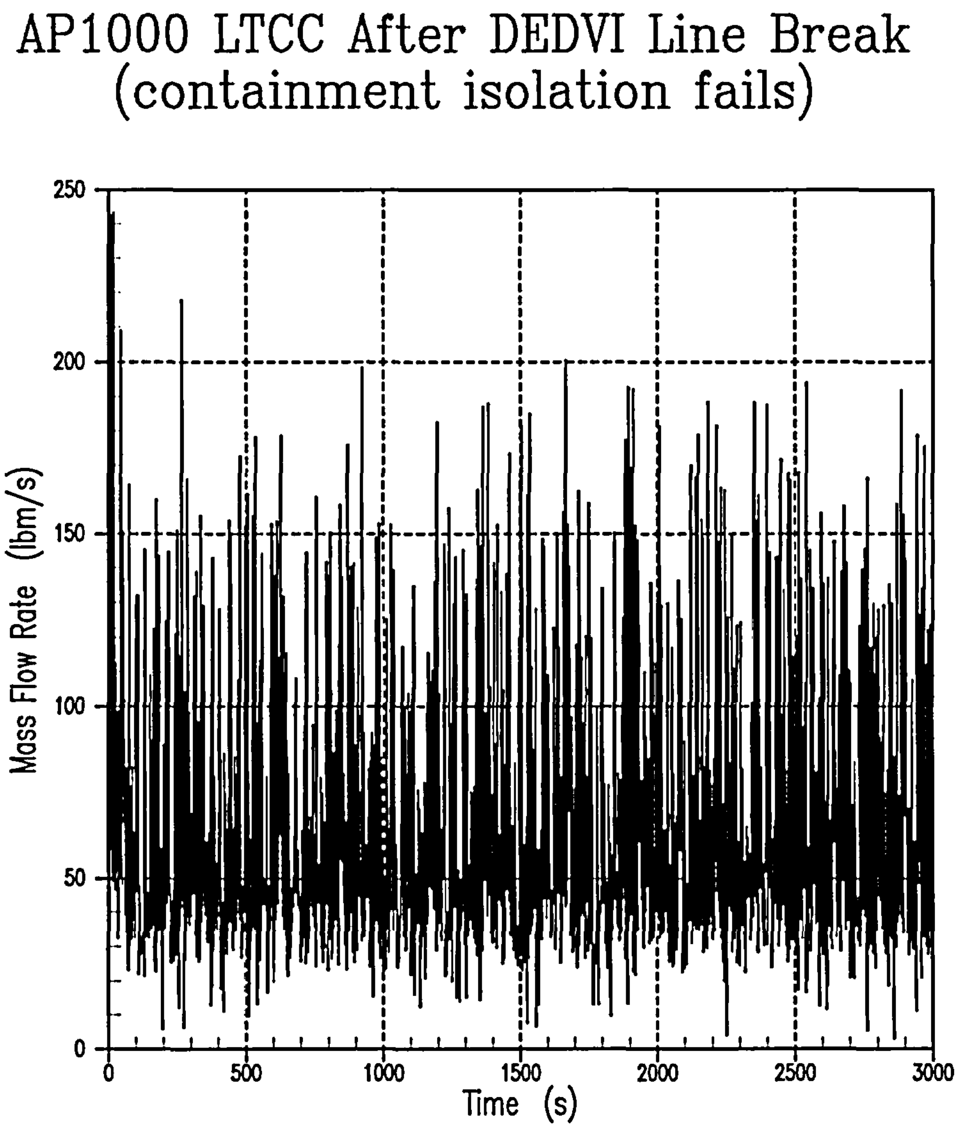


Figure A5.3-24

Case G – Mixture Flowrate Through ADS Stage 4B Valves

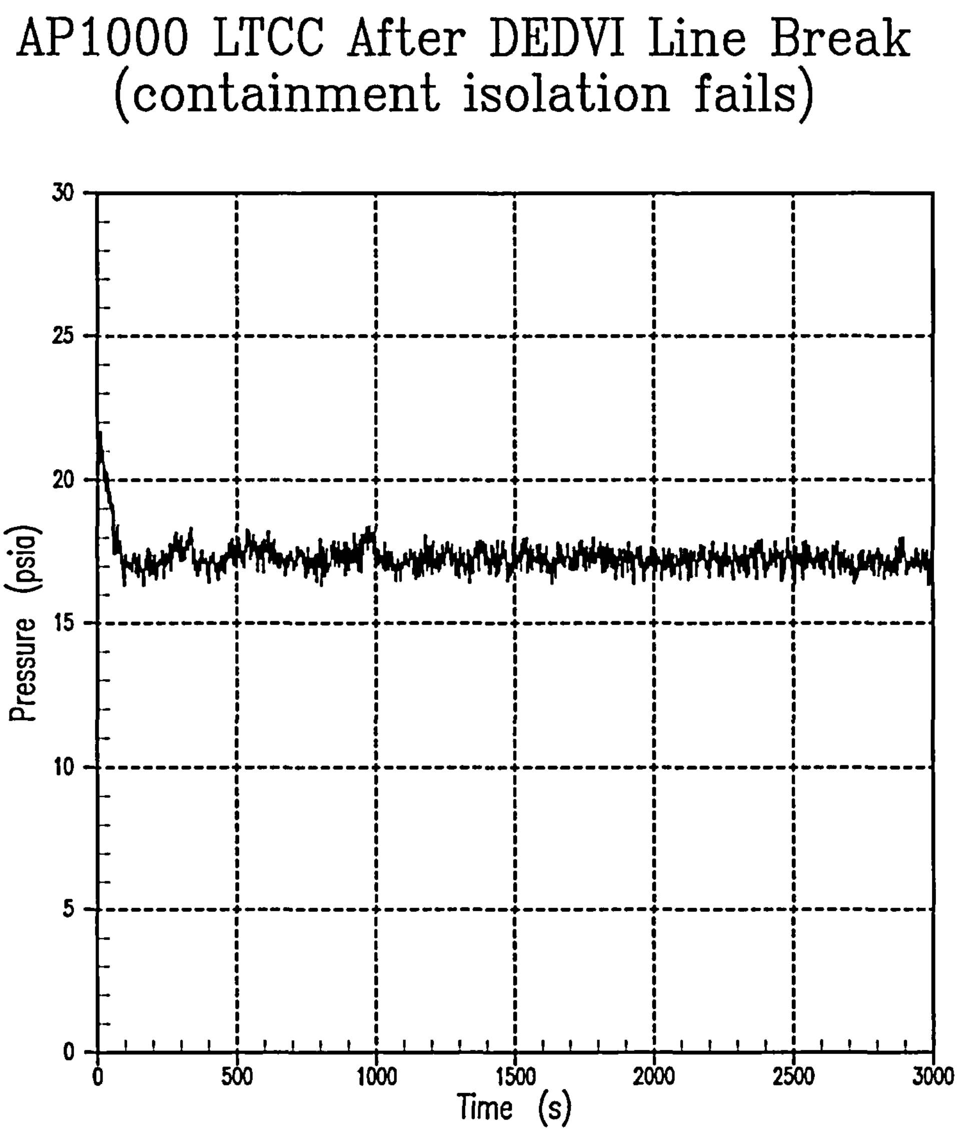


Figure A5.3-25

Case G – Upper Plenum Pressure

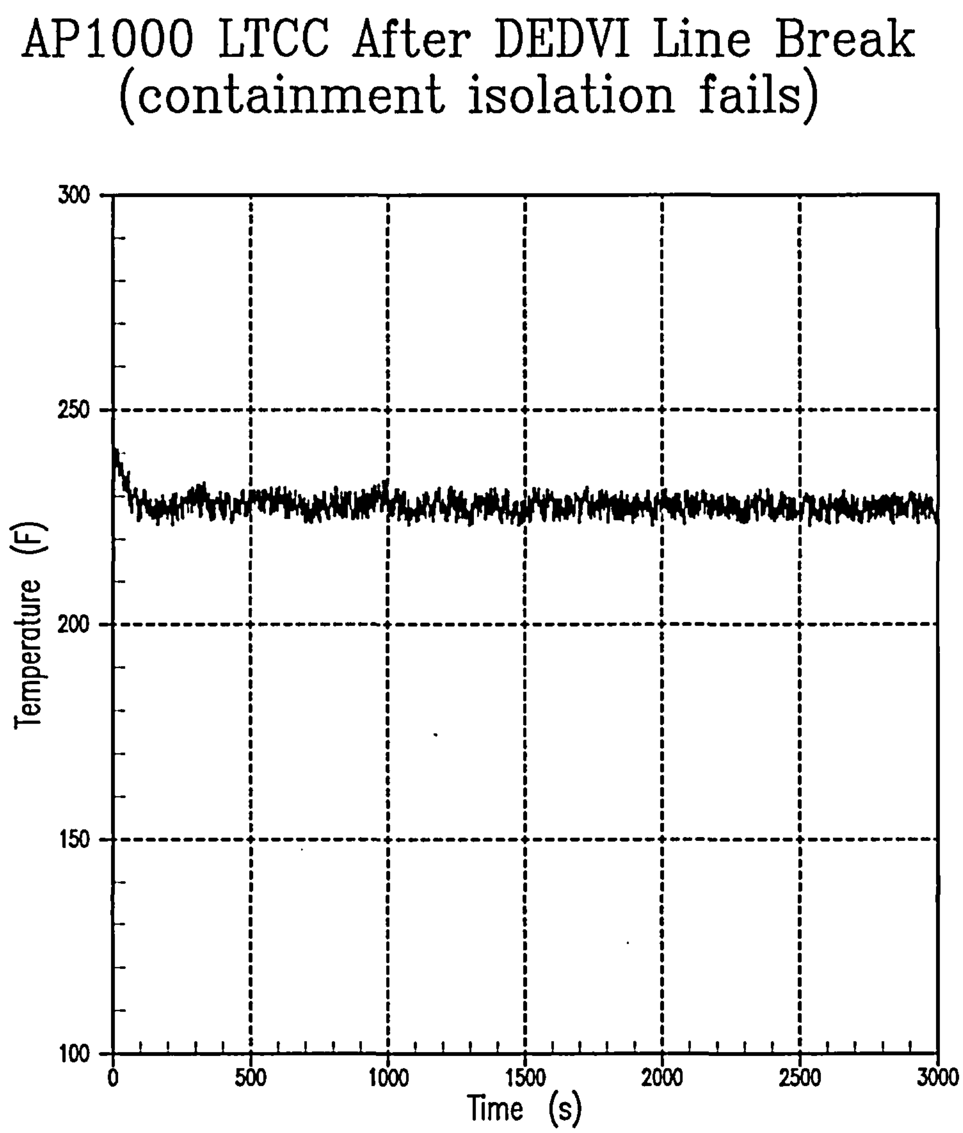


Figure A5.3-26

Case G – Hot Rod Clad Temperature in Cell 17

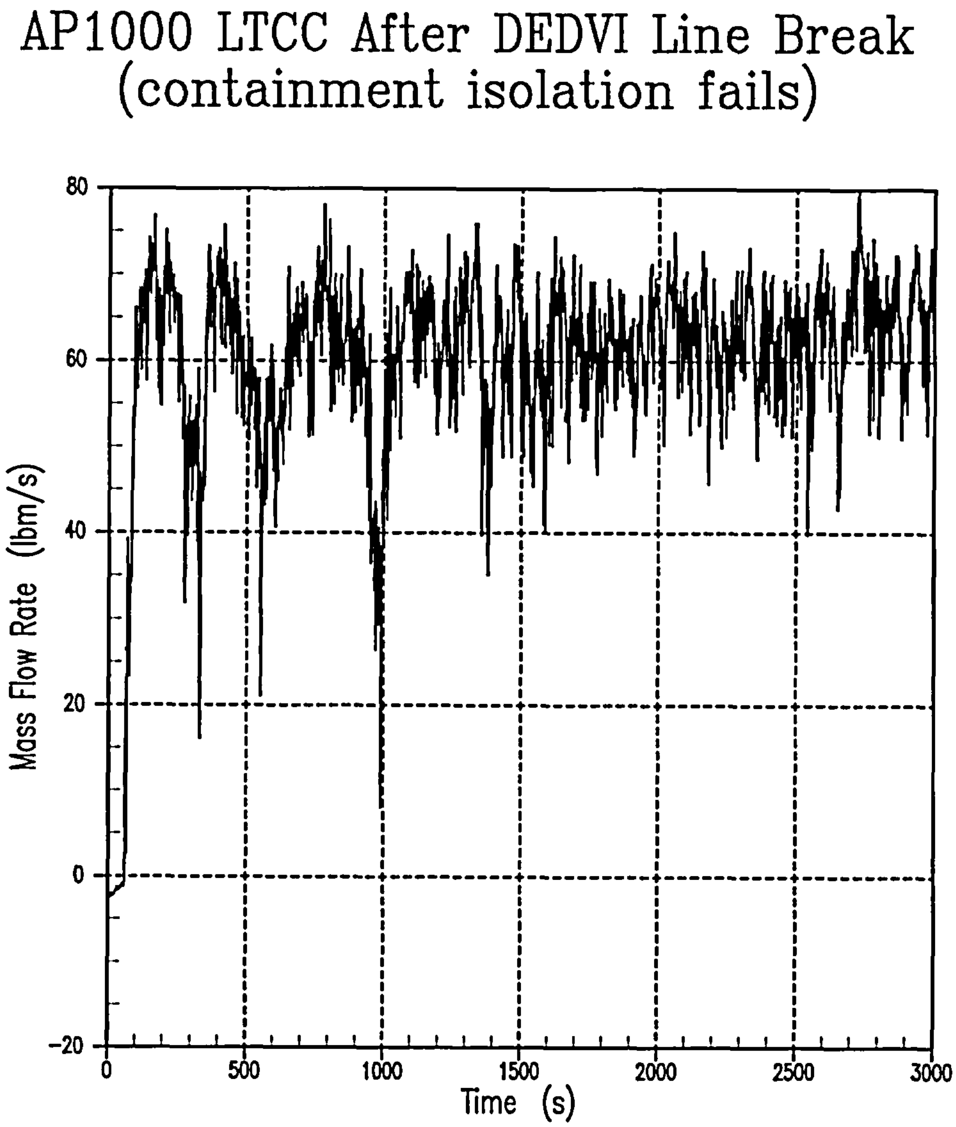


Figure A5.3-27

Case G – DVI-B Mixture Flow Rate

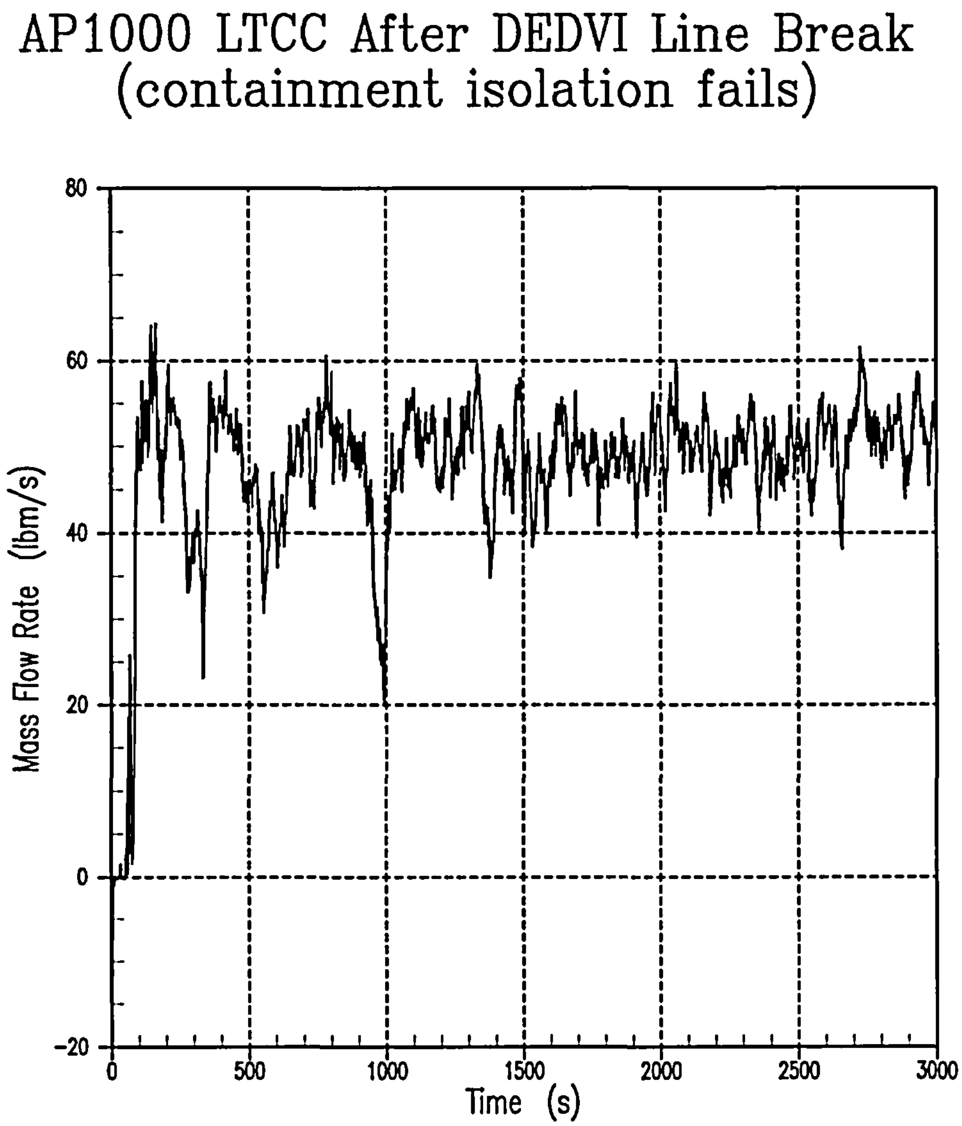


Figure A5.3-28

Case G – DVI-A Mixture Flow Rate

Westinghouse Response to NRC Additional Comment (from telecon):

In response to DSER Open Item 15.2.7-1 Item 7, a bounding calculation was performed with WGOTHIC to determine a conservative lower bound for the containment backpressure for the DEDVI case. For this analysis, it was assumed that the cold water spilled from the loop side of the DVI would be introduced into the compartment as 100-micron droplets. This assumption assures maximum condensation of steam in the compartment, and the containment pressure for this case was approximately 2 psi lower than the case where the cold water is spilled directly to the floor of the compartment as is shown in Figure 19.1.10.1-5 R2-1. For the NOTRUMP DEDVI case, it was determined that from the time that flow from the ADS-4 valves becomes unchoked until stable IRWST injection becomes established, the containment pressure is always above the assumed backpressure boundary condition of 25 psia. These conclusions formed the response to OI 15.2.7-1 Item 7.

An additional sensitivity study was performed with WCOBRA/TRAC to determine the effect of using this bounding containment backpressure for the AP1000 long-term cooling analysis. The DCD limiting case of a postulated DEDVI break in the PXS "B" Room was reanalyzed using the pressure curve shown in Figure 19.1.10.1-5 R2-1 from the time of sump injection onward. Figure 19.1.10.1-5 R2-2 compares the core collapsed liquid level predictions of the DCD analysis (dashed line) and the sensitivity case (solid line); the time axis of the WCOBRA/TRAC runs has a 2500 second time offset from the WGOTHIC case. In Figure 19.1.10.1-5 R2-2, the sensitivity case exhibits no adverse impact on core cooling relative to the DCD analysis; no cladding temperature excursion is predicted for either analysis. The long term cooling analyses for the DCD and PRA are not sensitive to the slightly lower backpressure indicated by the WGOTHIC bounding analysis.

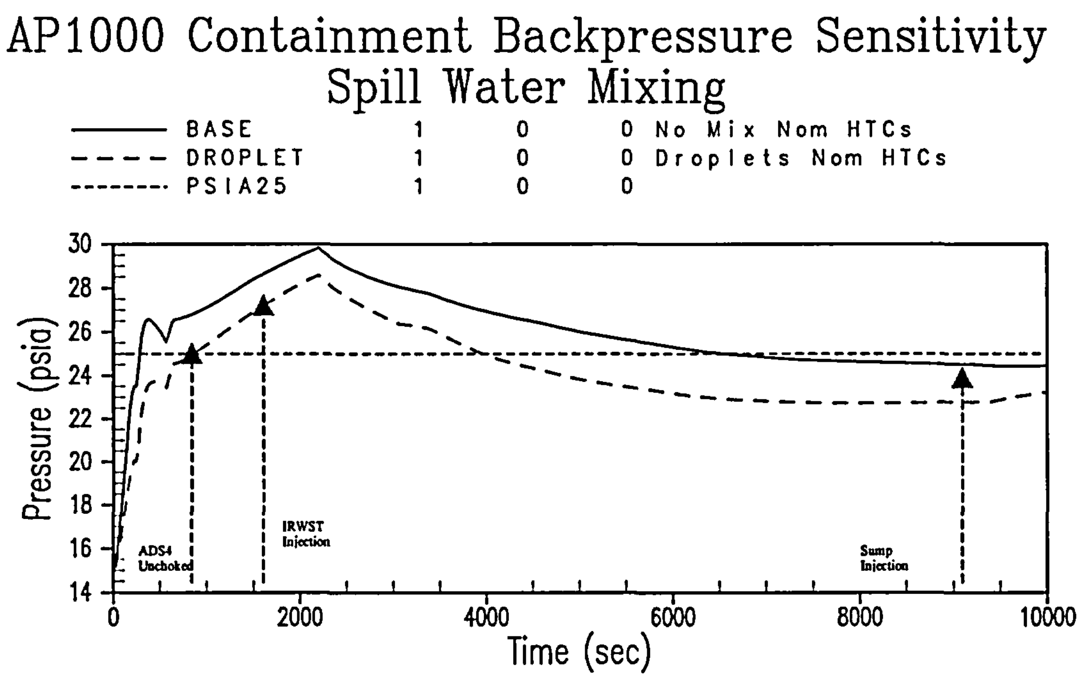
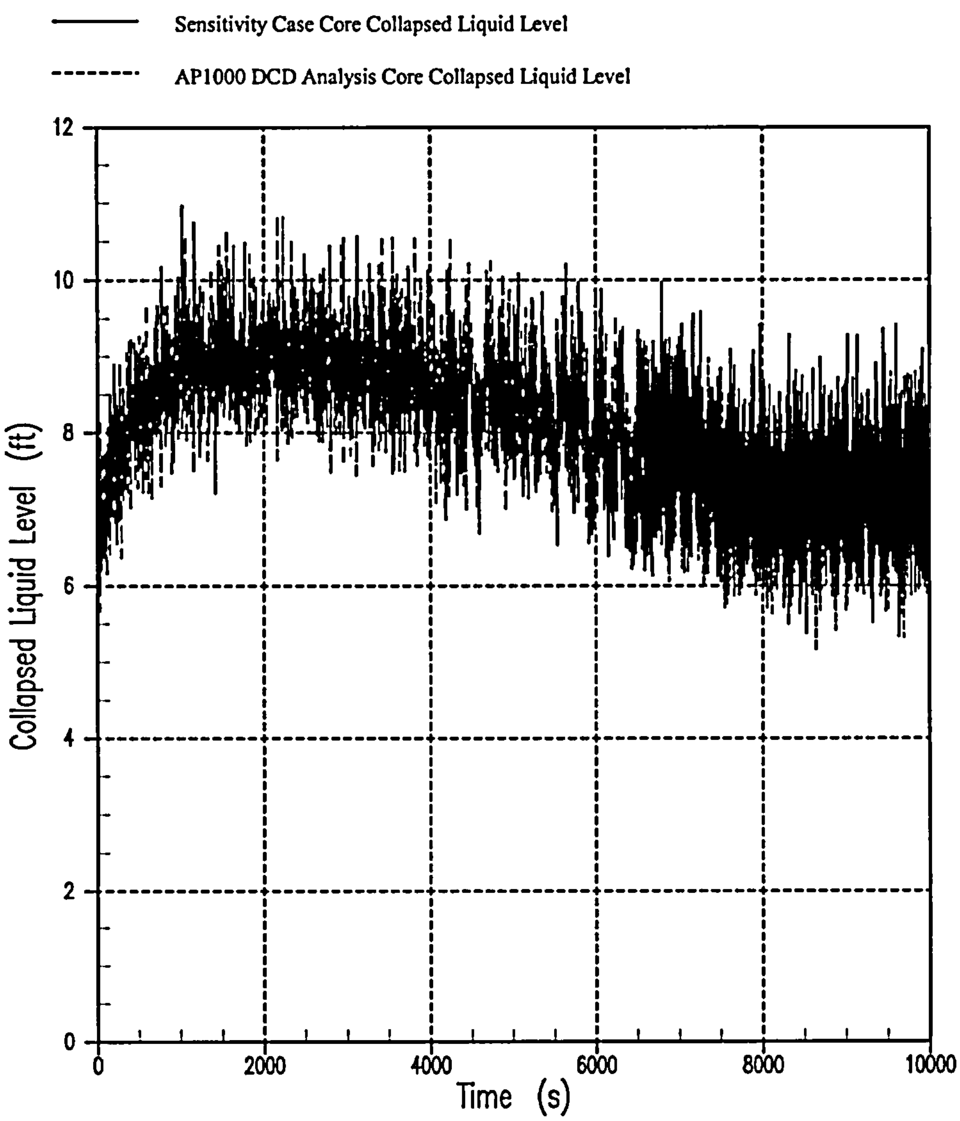


Figure 19.1.10.1-5 R2-1: Cold Water Droplet Size Sensitivity

Figure 19.1.10.1-5 R2-2: DEDVI Break Long-Term Cooling



AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-8 Response Revision 2

Original RAI Number(s): None

Summary of Issue:

Deterministic Approach

The applicant used the deterministic approach to estimate the HCLPF values of primary system component supports. The components included in the approach are: polar crane, baffle plate supports, heat exchanger for the passive residual heat removal system, core makeup tank and valves. The applicant used lower bound values, and it appears that there was no need for invoking factors of conservatism to arrive at the HCLPF values. It is noted that the core makeup tank has a HCLPF value of 0.54g; therefore, any increase in seismic response of the containment internal structure due to lift off of the internal structure or the nuclear island structure would necessitate a review of this HCLPF value. This is Open Item 19A.2-8.

Additional comments during meeting on July 10, 2003

This response is incomplete; it does not include the effect of CIS lift off.

Additional comments during meeting during the week of October 6, 2003

Westinghouse has not adequately justified that the effect of lift off of the nuclear island basemat is not significant in the seismic margin evaluation. The results that are presented in the Revision 1 response may cause the HCLPF values to fall below the review level earthquake of 0.5g. Further, the CIS lift off with the nuclear island basemat uplift may make the seismic response worse. Westinghouse has not presented results from a combined model with liftoff from the CIS and nuclear island basemat to demonstrate that the results are not higher than those presented.

Westinghouse Response (Completely revised in this Revision 2):

This revision of the response provides revised results of the effects of nuclear island basemat uplift. The revised analyses model the footprint of the basemat more accurately. They also include damping that is more applicable for the review level earthquake of 0.5g. Parametric analyses are also described that justify the modeling of soil mass. Uplift of the containment internal structures and containment vessel is precluded by the addition of shear studs on the outside of the containment vessel. The effect of the nuclear island basemat uplift on the response spectra is considered in the HCLPF evaluation.

Nuclear Island Basemat Uplift

The analyses of the nuclear island for the SSE input of 0.3g considering uplift of the basemat are described in the response to DSER Open Item 3.7.2.3-1. In these analyses, and in the initial

analyses at the 0.5g review level earthquake, the nuclear island basemat model used an equivalent rectangle for the basemat, and damping that is applicable for SSE input.

A new model was developed that models more accurately the actual footprint of the basemat. The actual footprint configuration is modeled as shown in Figure 19A.2-8-1. The overall width is 161' whereas the equivalent rectangle only had a width of 140'. Both have the same overturning resistance in linear analyses where soil springs take tension. Both models have the same eccentricity between the center of mass of the nuclear island and the centroid of the basemat.

The new model includes damping applicable for the review level earthquake of 0.5g. For the reinforced concrete Auxiliary and Shield Building (ASB), a value of 10% modal damping is used for a 0.5g earthquake because of the increased amount of cracking of the concrete expected throughout the cylindrical portion. Damping is modeled using the alpha-beta method with10% damping at frequency values of 3 Hz and 25 Hz. These frequency values are selected to give 10% damping at the 3 hertz fundamental frequency of the ASB and to limit the damping at the dominant frequencies of the steel containment vessel (SCV) and containment internal structures (CIS) to 7% or lower. The alpha-beta damping is shown in Figure 19A.2-8-2 as a function of frequency. As shown in this figure, the damping in the frequency range from 3 Hz to 25 Hz is conservative compared to 10 % damping. The damping ratio is less than 7 % at the primary horizontal frequency of 6 Hz of the SCV. The damping ratio is 6 % at the primary horizontal frequency of the CIS of 8 Hz.

Time history analyses were performed using the new model of the basemat combined with the auxiliary shield building (ASB) stick model described in DSER Open Item Number 3.7.2.3-1. The soil was modeled with ANSYS COMBIN37 elements which do not include soil mass. Figure 19A.2-8-3 shows basemat displacement plots from the time history analysis performed using a peak ground acceleration of 0.5g. The upper plot shows the linear analysis results, and the lower plot shows the liftoff (non-linear) analysis results. The displacements are shown both at the time when the peak bearing pressure occurs and at the time when peak uplift occurs. The maximum liftoff is 0.12 inches at the east edge. This is considerably less than the vertical displacement of 0.29" for the previous non-linear analyses given in the Revision 1 response.

Figures 19A.2-8-4 to 19A.2-8-8 compare the response spectra from the non-linear analyses (with lift-off) against those from linear analyses (no lift-off). The comparison shows that the effect of liftoff on the horizontal seismic response is insignificant. There are small differences in the vertical response spectra in the higher frequency region for the Shield Building cylinder up to elevation 265'. The effects of lift off are much less than in the analyses described in the Revision 1 response.

Effect of Soil Mass

The uplift analyses use a non-linear spring to represent the soil and do not include soil mass. The effect of soil mass was addressed in the Revision 1 response to DSER Open Item 3.7.2.3-1 where results were provided for a model with a large soil mass at the top of the spring. Additional studies are described below which further demonstrate that the mass of the soil is not

significant. These studies make comparisons to the original uplift analyses using the equivalent rectangular footprint and damping associated with the SSE.

The effect of liftoff on the vertical accelerations and floor response spectra at elevation 116.5' is shown in Figure 19A.2-8-9. This figure shows the time history from the linear and non-linear analyses and the difference in these two accelerations. The difference shows that the lift off is of short duration and is typical of an impact as the gap closes. This same effect was reproduced in a simplified free fall impact analysis in which the vertical model was analyzed assuming an initial gap of 0.08 inches corresponding to the maximum lift off of the center of mass. The accelerations and response spectra from this analysis are compared against the differences in the non-linear and linear time history analyses of the full model in Figure 19A.2-8-10. The results for the simplified vertical model are higher due to the single point of impact in the simplified vertical model versus the multiple points of impact in the original model.

The simplified vertical only impact model was used to investigate the modeling of soil mass. Results for the following cases are shown in Figure 19A.2-8-11:

- Zero soil mass as included in original analyses described in the response to DSER Open Item 3.7.2.3-1.
- Soil column 60 feet high with 7 masses.

The results show that the modeling of soil mass has little effect on the response.

Containment Internal Structure Uplift

The evaluation of uplift of the containment internal structures and containment vessel due to the SSE is described in the Revision 1 response to DSER Open Item 3.8.5.4-1. Non-linear analyses were performed on a model of the containment internal structures and containment vessel applying a factor α to all of the seismic loads (a value of 1.0 for α is equivalent to the plant SSE level). Figure 19A.2-8-12 shows the vertical displacement of the vessel at the center of containment and at the edge as a function of the value of α for seismic loads in the EW and NS directions. At a value of 1.0, corresponding to input ground motion of 0.3g, the center of containment does not lift off and the vertical deflection at the edge is 0.043 inches.

The SSE stability analyses are very conservative because:

- The seismic loads are applied statically without consideration of their variation due to the dynamic response of the structures. The time history lift off analyses of the nuclear island basemat described above show lower lift off than in non-linear equivalent static analyses of the same model.
- The equivalent static loads are based on the maximum loads associated with each component for the complete seismic time history without consideration of the time of occurrence.
- The loads associated with the CIS, SCV, and RCL are added absolutely.

- Overturning is assumed to occur about a support at elevation 90.6'. The concrete cradle extends up to elevation 100' and provides additional resistance to overturning not considered in the analyses.

Damping will increase during earthquakes higher than the SSE level (0.3g). An increase in damping of the steel vessel from 4 to 7 percent will reduce the response by 25% (amplification factors for the ground response spectra are given in DCD Table 3.7.1-3 and are consistent with Regulatory Guide 1.60). Recognizing the conservatism of the SSE analyses and the increase in damping, the uplift response for the review level earthquake input of 0.5g is estimated to be represented by an α value of about 1.3 [0.5 / (0.3 x 1.25) = 1.3] in Figure 19A.2-8.12. The lift off of the center of mass would be about 0.014 inches based on the equivalent static analyses. This lift off of the center of mass is smaller than that considered in the simplified impact analyses described above for the nuclear island basemat. It would result in only a small increase in the seismic response of the containment internal structure.

An evaluation was also performed for uplift of the containment internal structures with separation occurring at the inside face of the containment vessel. This was similar to that described above for the separation at the outside surface of the containment vessel. The results are shown in Figure 19A.2.8-13. Lift off initiates at the edge at an α value of 1.3 and at the center at an α value of 1.8. There is no significant lift off at levels up to the review level earthquake.

To provide additional margin, shear studs have been added on the outside of the containment vessel. The studs transfer shear loads from the containment internal structures through the containment vessel into the nuclear island basemat and prevent lift off under the review level earthquake. The shear studs are designed to resist seismic overturning loads due to the review level earthquake of 0.5g. They are also analyzed to confirm their integrity for loads associated with containment pressurization.

<u>Effect of nuclear island basemat uplift on response spectra</u>

The effect of uplift on the response spectra is small. It generally has little effect in the seismic margin evaluation because:

- High Frequency Content

  High frequency content caused by liftoff is intermittent during the seismic response due to the impact of the NI basemat on the foundation media. This excitation is not a damaging since the response is limited and resonance effects are greatly reduced.

- High frequency seismic response of the shield building cylinder

  The vertical response spectra are amplified at and above the vertical frequency of the shield building cylinder. This amplification is due to the additional excitation due to impact of the cylinder with the foundation rock as shown by the simplified impact model. This high

frequency response will not be as pronounced outside the shield building since the other buildings are lower and have a higher vertical frequency.

- Side Soil Effects

  The liftoff analyses did not include the effect of side soils. AP600 studies of seismic soil pressure distribution concluded that side soil effects can reduce (potentially significantly) the vertical seismic response.

Effect of nuclear island basemat uplift on HCLPF values

The effect of uplift on the response spectra on the HCLPF values was reviewed. This review addressed each of the structures and components considered in the seismic margin evaluation.

- Buildings/Structures & Primary Components

  The uplift had no affect on the HCLPF values. It was found either that the vertical response frequency is less than 10 hertz, or that the horizontal response controls.

- Generic HCLPF values

  The HCLPF values in PRA Table 55.1 for generic components were reviewed. The generic median capacity was found to be equal to or above the AP1000 vertical floor response spectra that include the basemat uplift/impact effect.

- Valves

  The HCLPF values for valves are controlled by the horizontal response and do not change when considering the basemat uplift/impact effect.

- Electrical Equipment

  For the electrical equipment, the HCLPFs were found to be reduced for only two components. These changes are not significant since the RTDs changed from 3.75 pga to 3.54 pga, and the Incore Thermocouples changed from 3.94 pga to 3.71 pga.

It is concluded from this review that uplift of the basemat does not result in changes to the HCLPF values used in the seismic margin analysis. It is noted that the electrical equipment RTDs and Incore Thermocouples HCLPFs did lower slightly but will have no affect on the seismic margin evaluations. Further, all HCLPF values remain above 0.5g.

Design Control Document (DCD) Revision:

Revise 3.8.2.1.2 as shown below:

3.8.2.1.2 Containment Vessel Support

The bottom head is embedded in concrete, with concrete up to elevation 100′ on the outside and to the maintenance floor at elevation 107′ 2″ on the inside. The containment vessel is assumed as an independent, free-standing structure above elevation 100′. The thickness of the lower head is the same as that of the upper head. There is no reduction in shell thickness even though credit could be taken for the concrete encasement of the lower head.

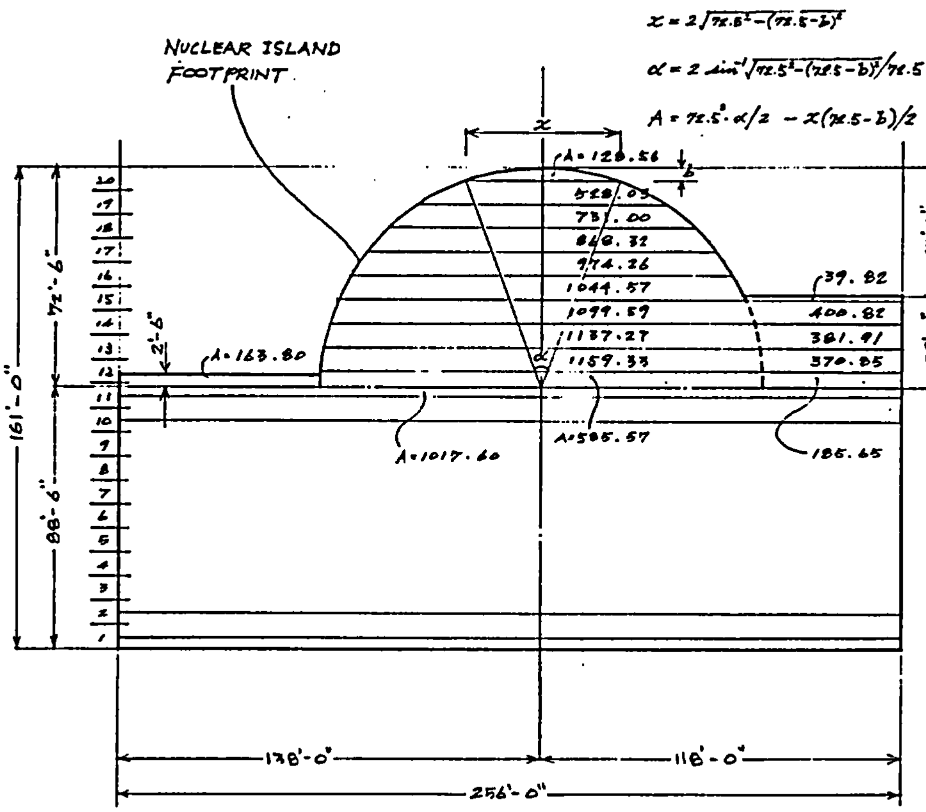
Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by **shear studs,** friction and bearing. **The shear studs are not required for design basis loads. They provide additional margin for earthquakes beyond the safe shutdown earthquake.**

Seals are provided at the top of the concrete on the inside and outside of the vessel to prevent moisture between the vessel and concrete. A typical cross section design of the seal is presented in Figure 3.8.2-8, sheets 1 and 2.

PRA Revision:

Revise the following components in Table 55.1:

RTD	-	-	~~4.46~~**3.54**	[6]
Incore Thermocouple	-	-	~~4.69~~**3.71**	[6]



NODE	Y (ft)	AREA (ft²)
621	1072.50	128.6
620	1064.45	528.0
619	1056.40	731.0
618	1048.35	868.3
617	1040.30	974.3
616	1032.25	1084.4
615	1024.20	1500.4
614	1016.15	1519.2
613	1008.10	1530.2
612	1000.05	1952.6
611	992.00	2060.8
610	983.95	"
609	975.90	"
608	967.85	"
607	959.80	"
606	951.75	"
605	943.70	"
604	935.65	"
603	927.60	"
602	919.55	2060.8
601	911.50	1030.4

Σ AREA 32455.4

Σ Y × AREA 31726957.8

ΣAY/ΣA = 977.6

EQUAL WITH EQUIVALENT RECTANGULAR CENTER. OK

Figure 19A.2-8-1
Actual Footprint Modeling

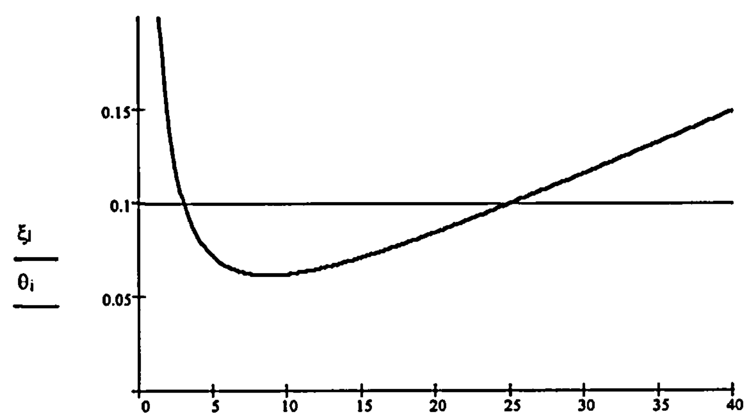


Figure 19A.2-8-2
Alpha-Beta Damping Curve

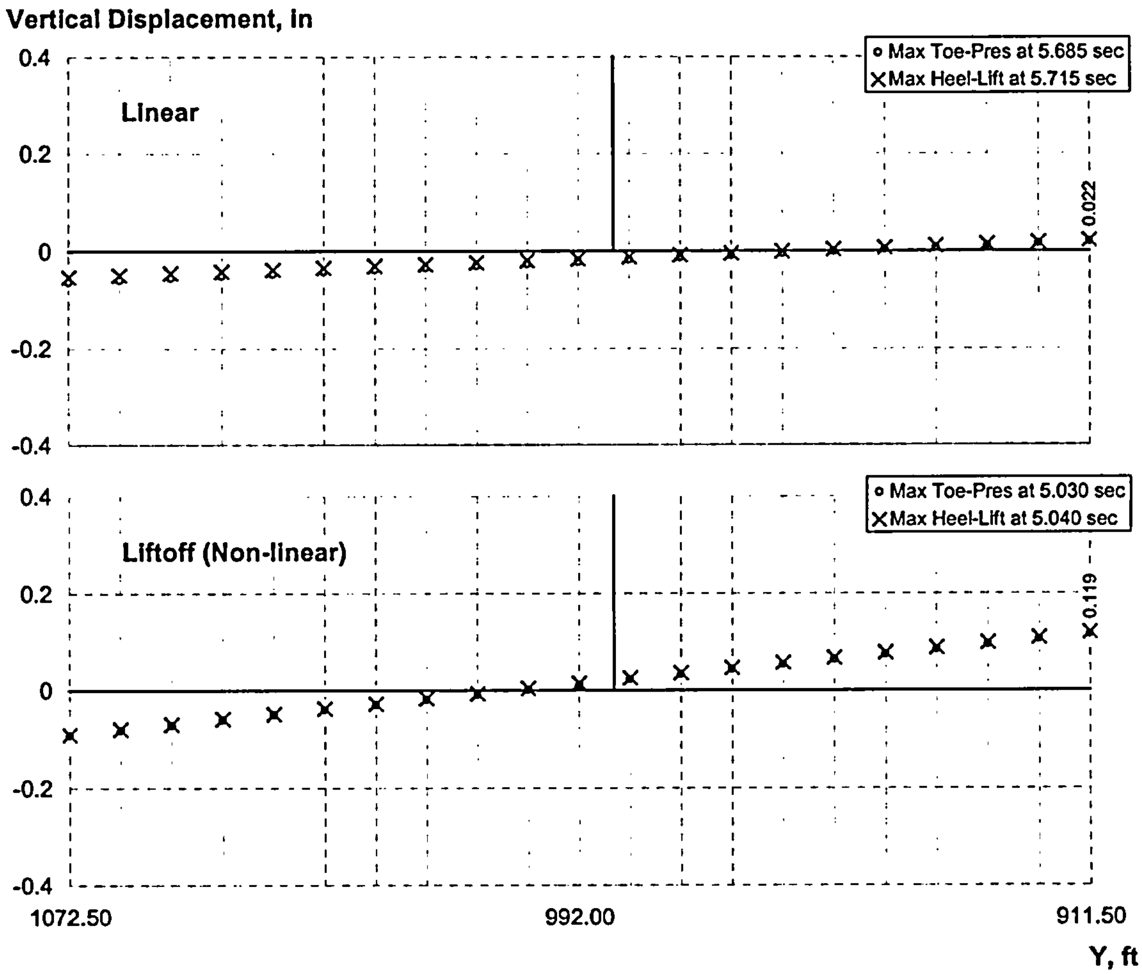
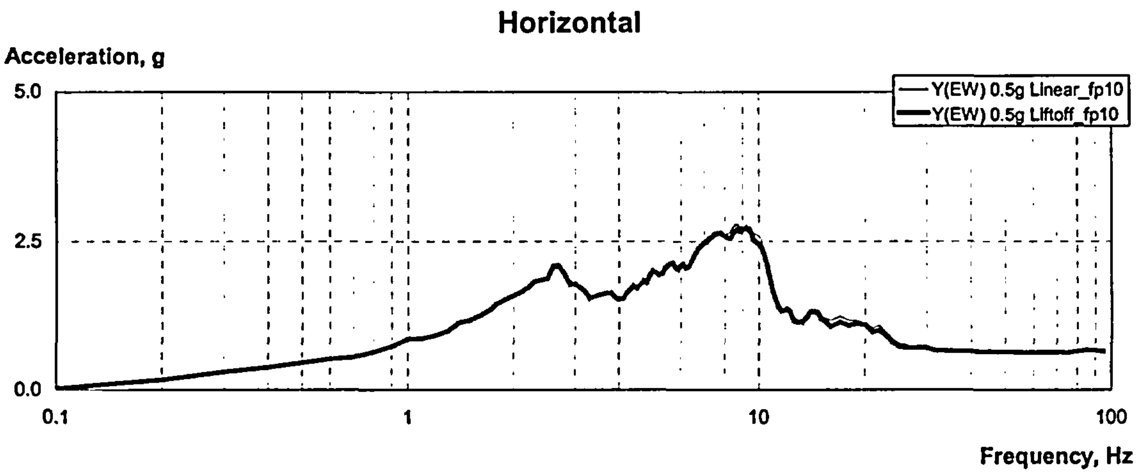


Figure 19A.2-8-3
Basemat Displacement



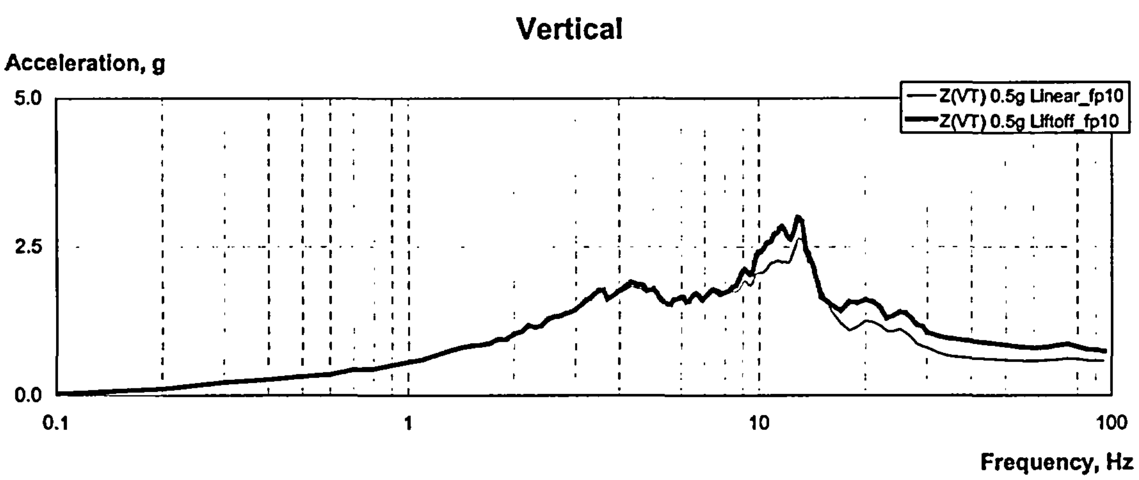
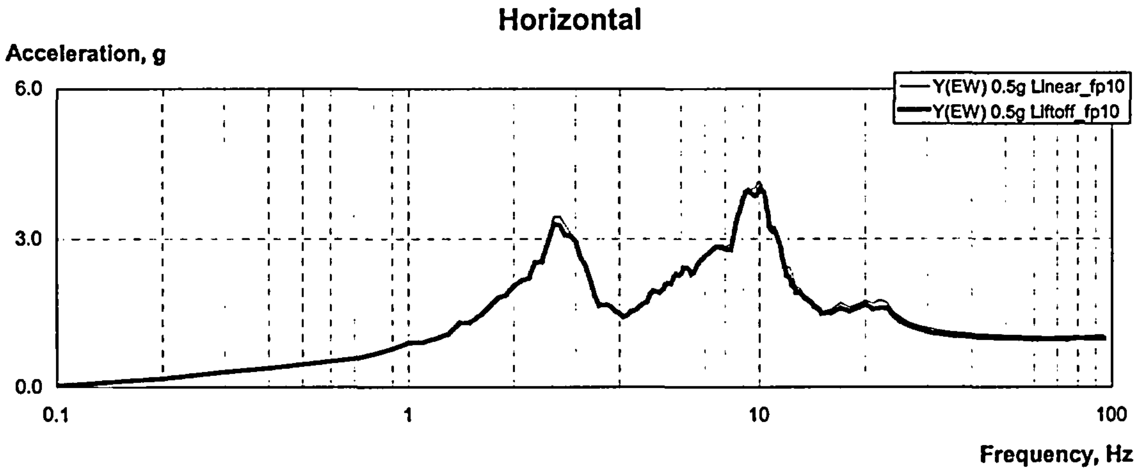


Figure 19A.2-8-4: Floor Response Spectra of ASB Node at EL. 116.50'



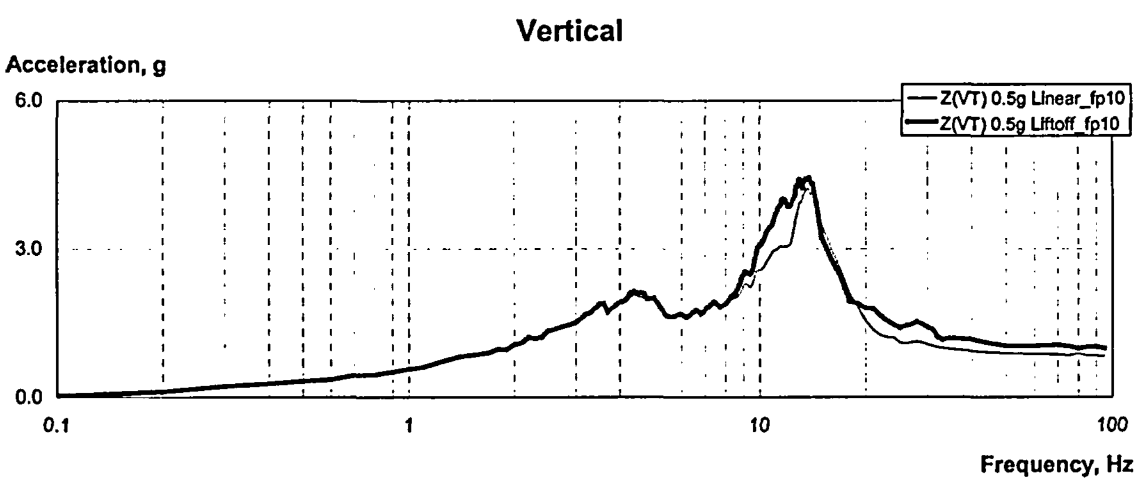
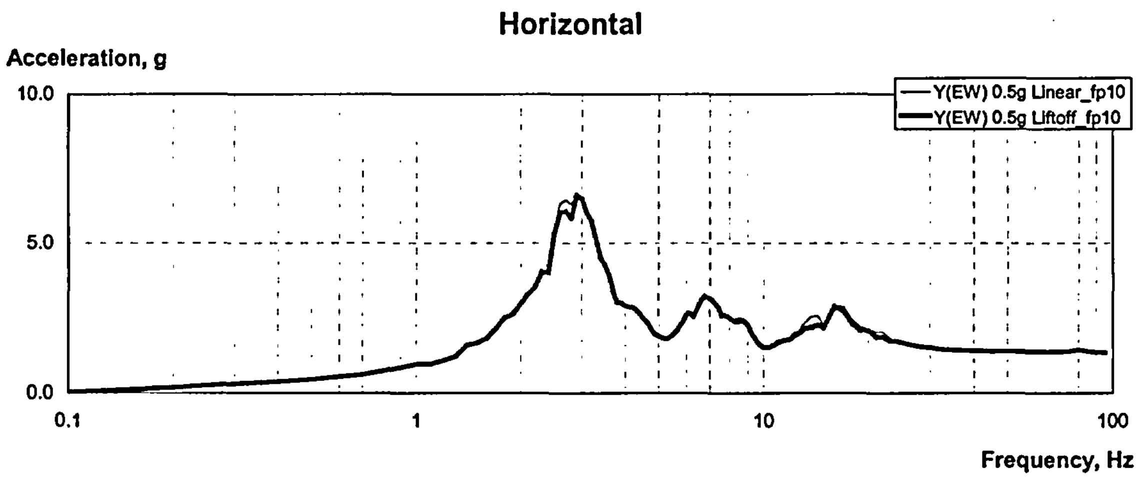


Figure 19A.2-8-5: Floor Response Spectra of ASB Node at EL. 179.56'



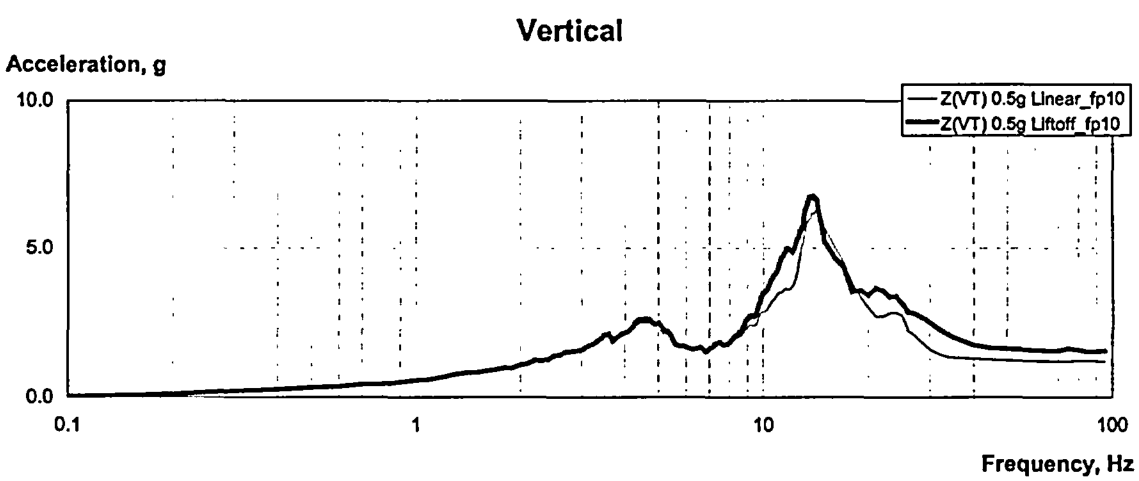
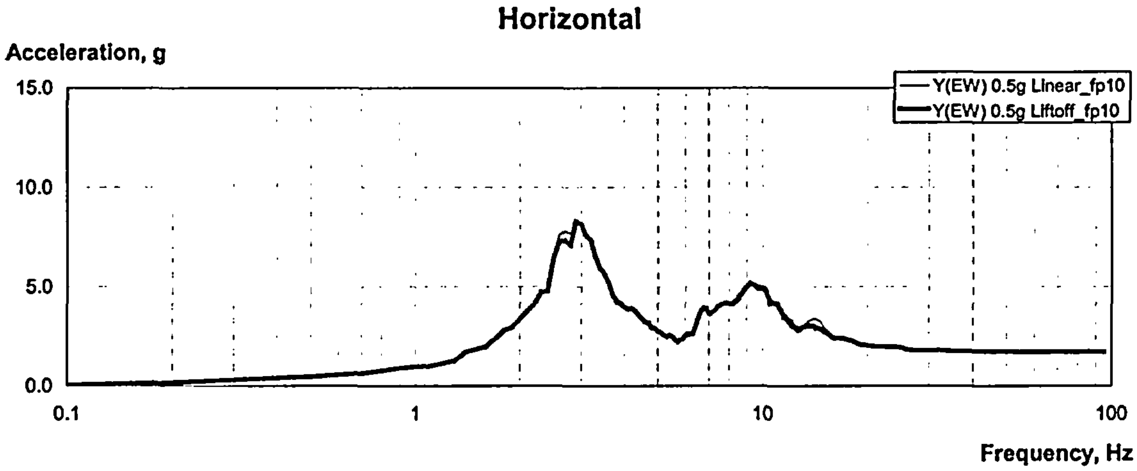


Figure 19A.2-8-6: Floor Response Spectra of ASB Node at EL. 265.00'



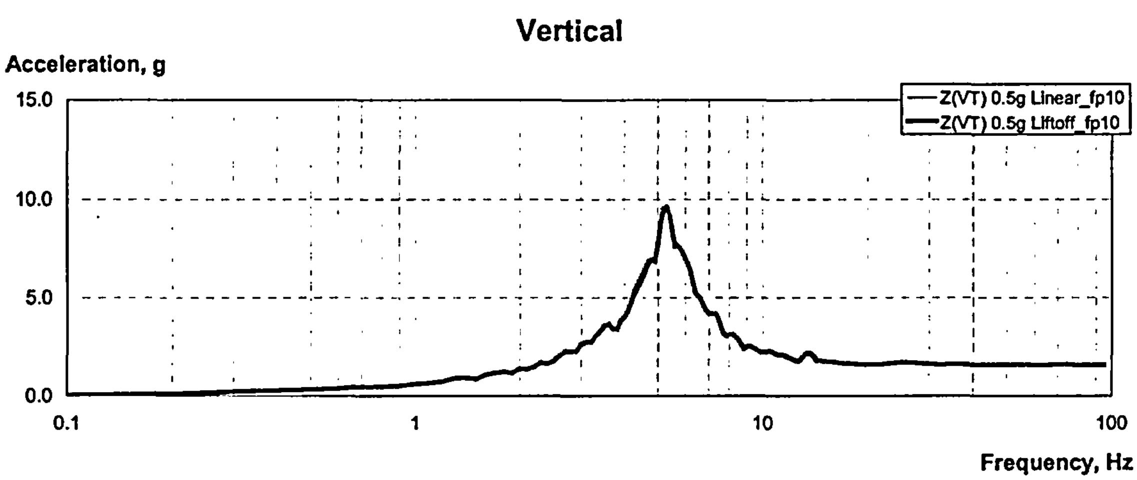
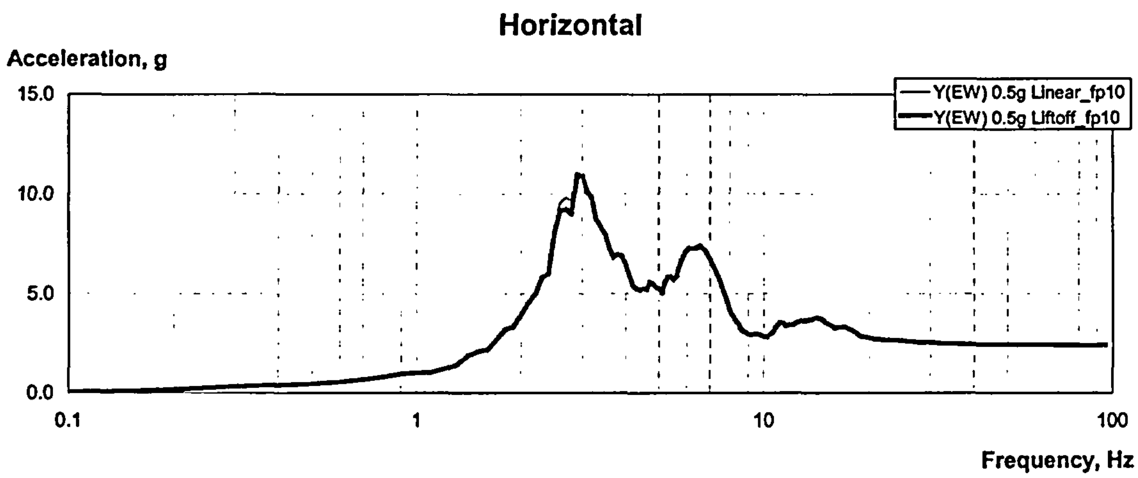


Figure 19A.2-8-7: Floor Response Spectra of ASB Node at EL. 295.23'



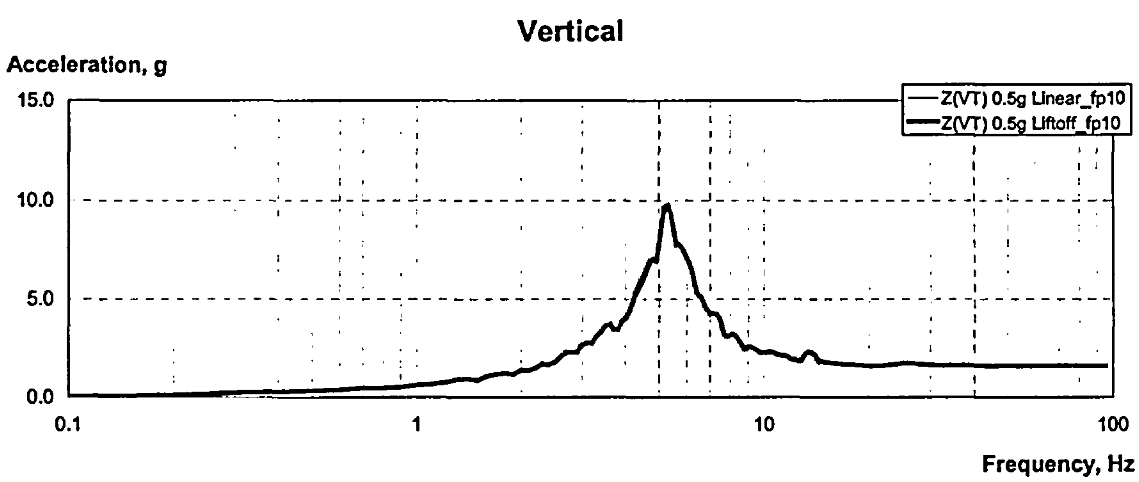


Figure 19A.2-8-8: Floor Response Spectra of ASB Node at EL. 333.13'

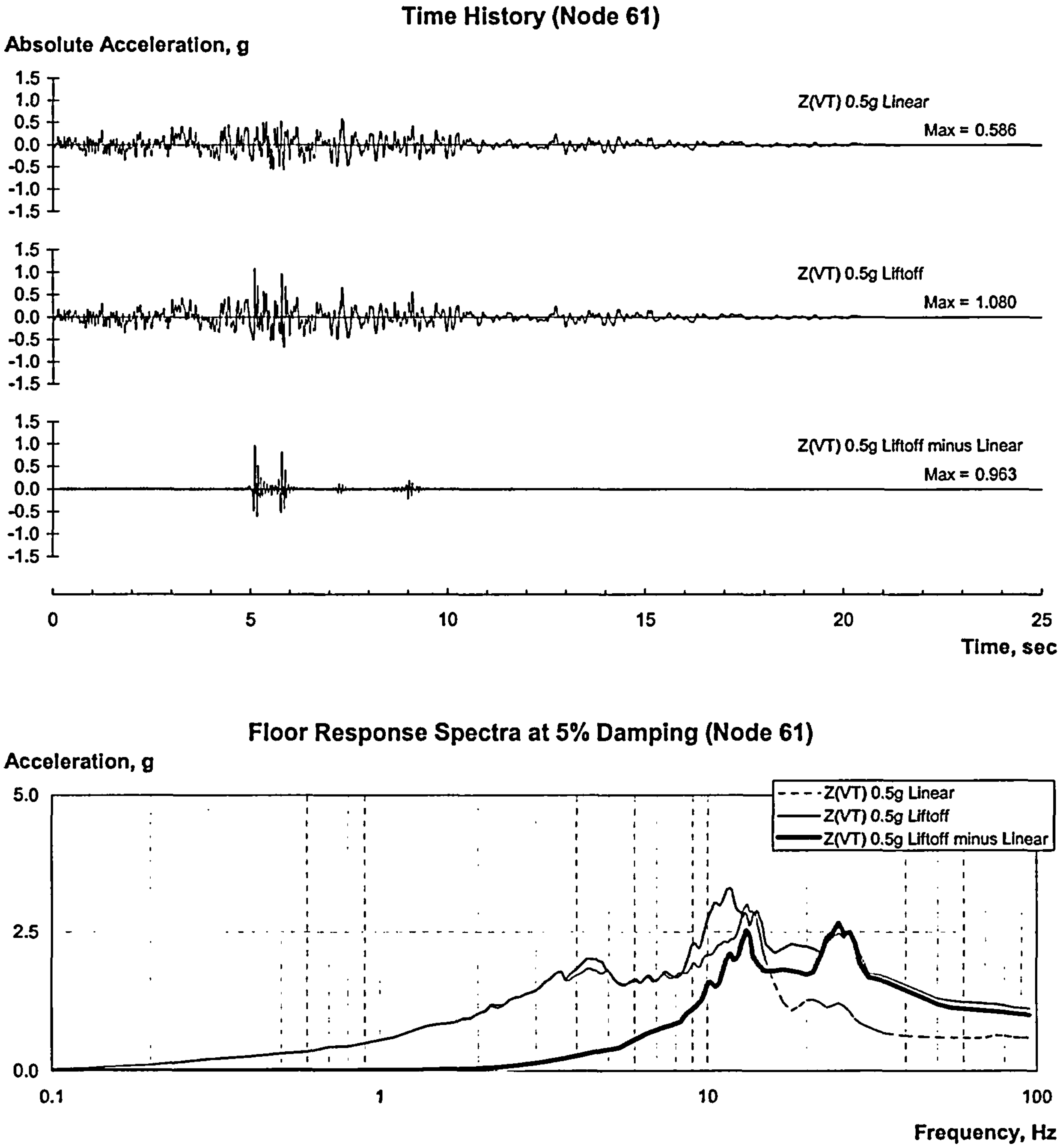
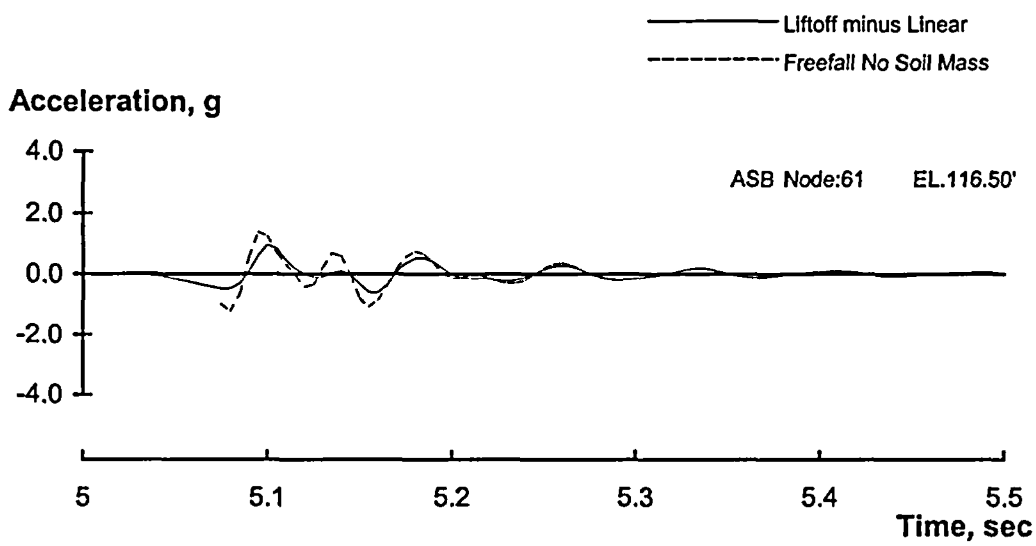


Figure 19A.2-8-9: Vertical Response of ASB Node at EL. 116.50'



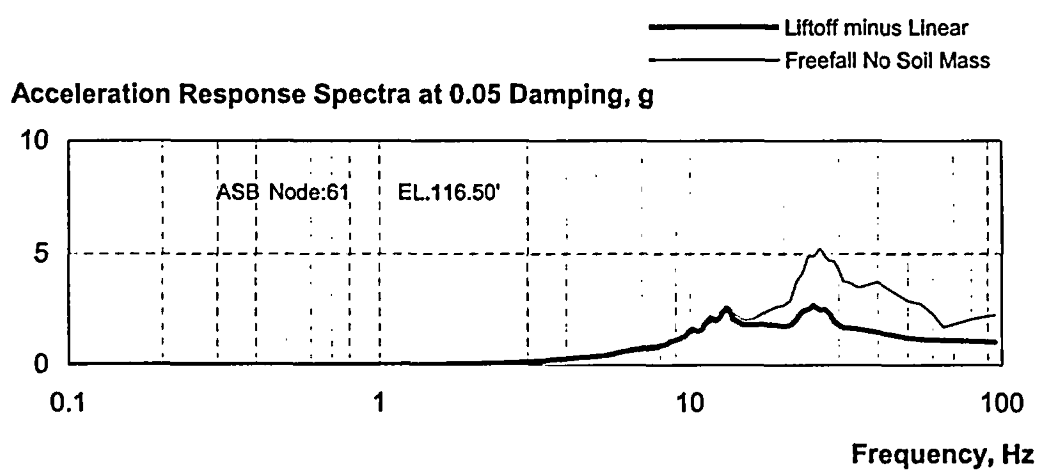


Figure 19A.2-8-10: Vertical Free Fall Response of ASB Node at EL. 116.50'

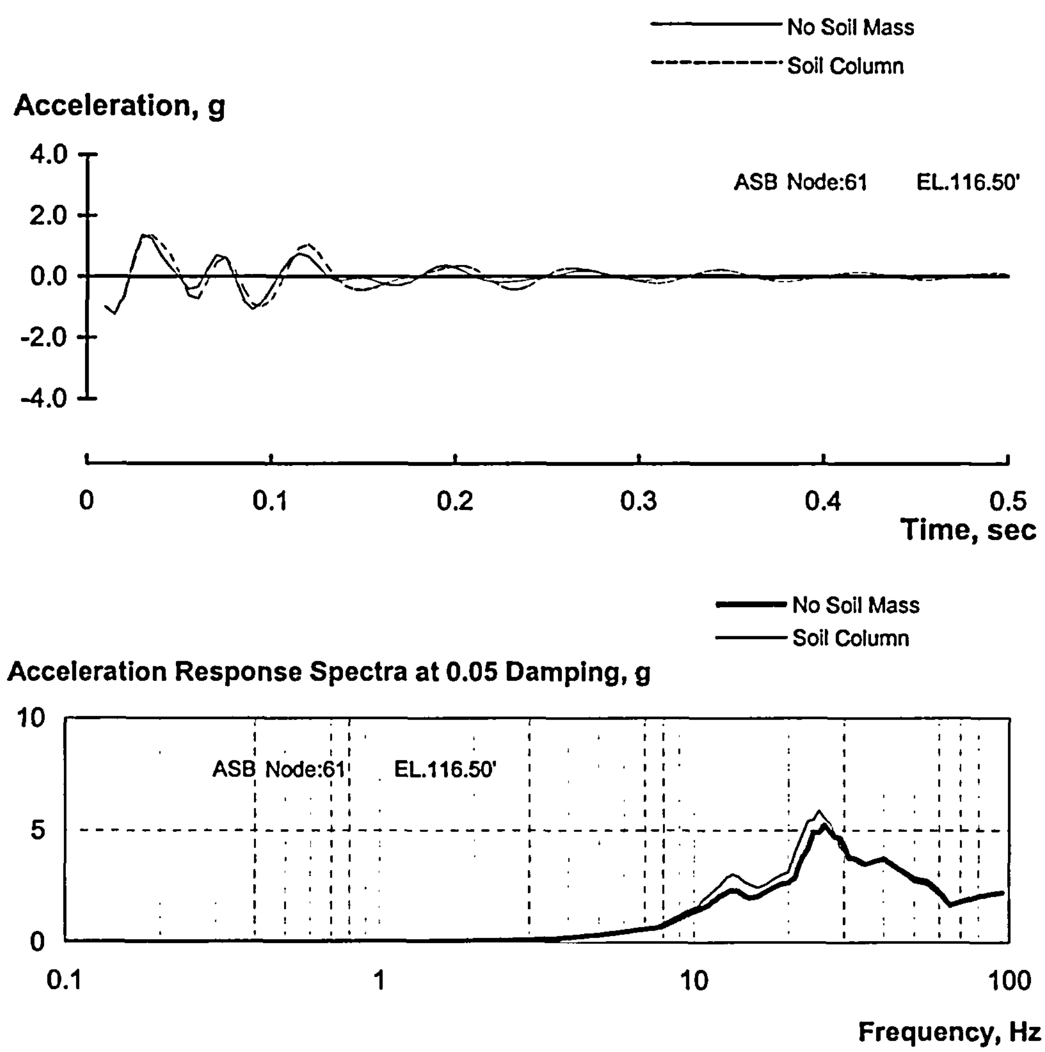
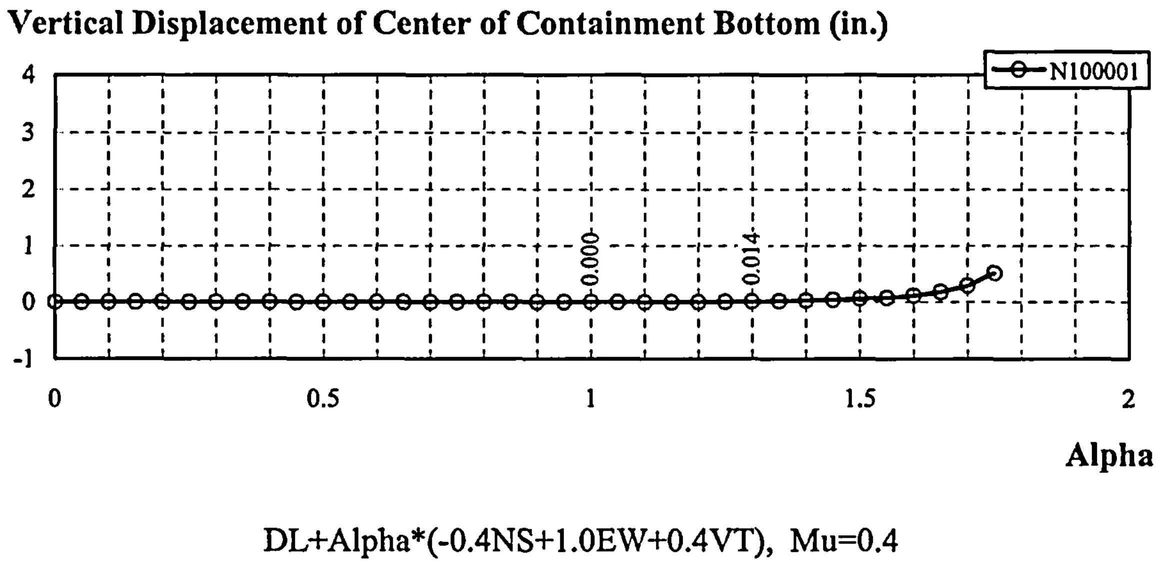


**Figure 19A.2-8-11: Vertical Free Fall Response of ASB Node at EL. 116.50'
Effect of soil mass model**



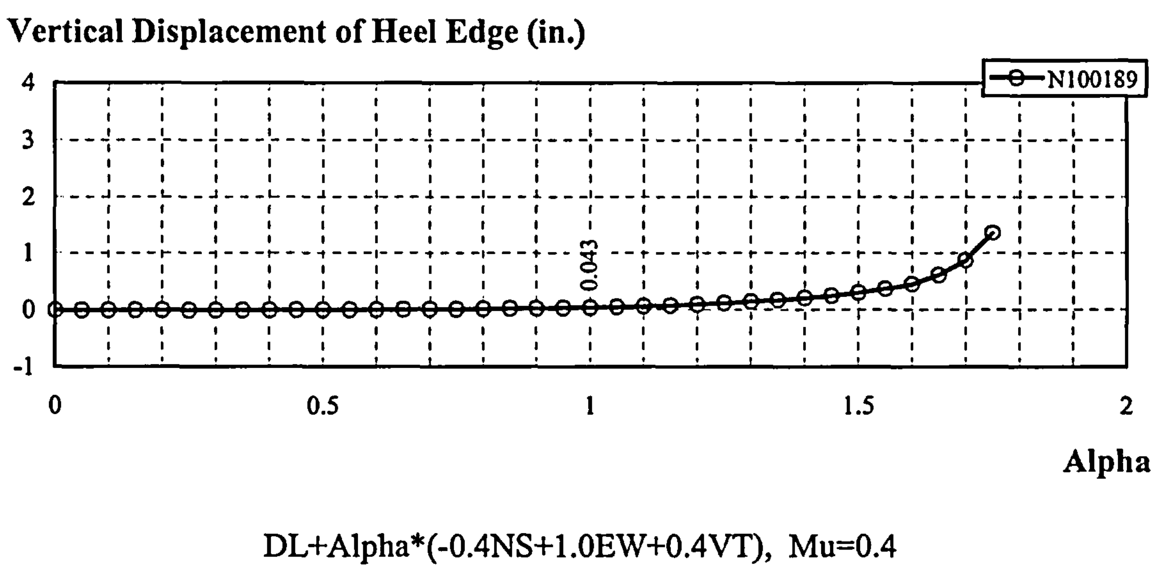
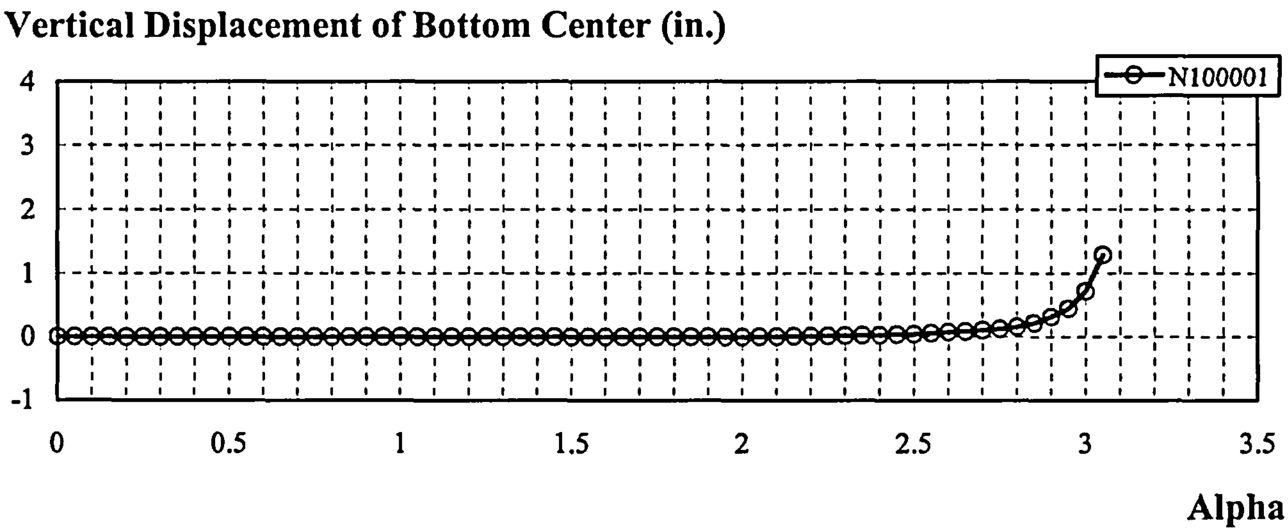


Figure 19A.2-8-12

Vertical Displacements for DL + Alpha x SSE applied to Containment Internal Structures and Containment Vessel



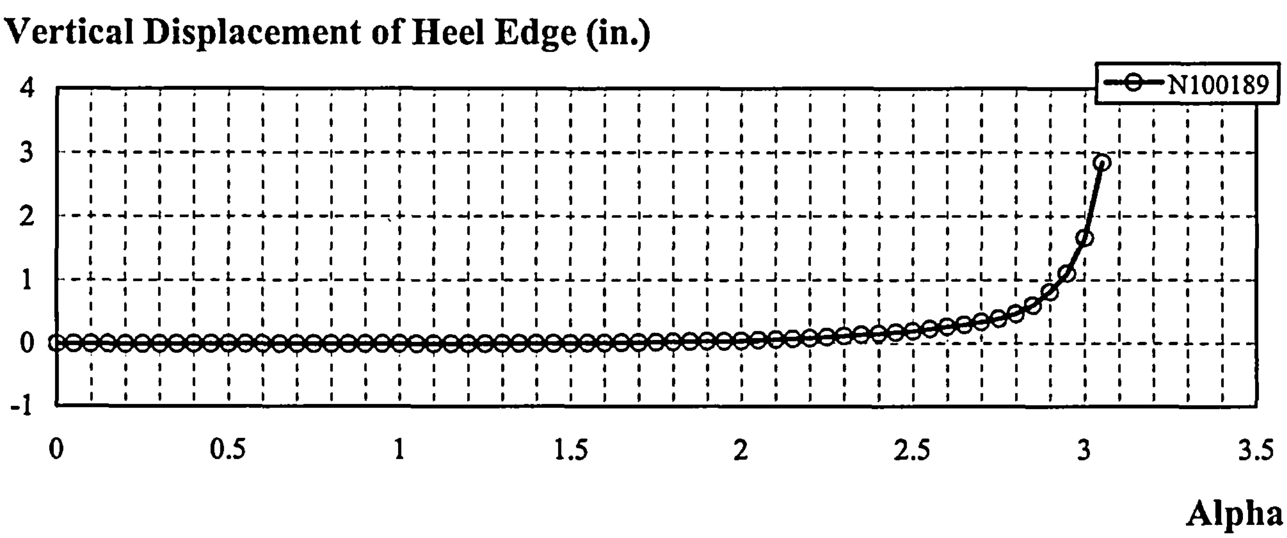


Figure 19A.2-8-13

Vertical Displacements for DL + Alpha x SSE applied to Containment Internal Structures