October 1, 1996 5Q-002

Mr. Charles B. Brinkman, Director
Nuclear Systems Licensing
ABB Combustion Engineering, Inc.
Post Office Box 500
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING CENPD-137, SUPPLEMENT 2-P

Dear Mr. Brinkman:

ABB Combustion Engineering (ABB CE) letter LD-96-017, dated May 23, 1996, submitted CENPD-137, Supplement 2-P "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model" for staff review. Enclosed is a request for additional information on the submittal. The request identifies items of concern which must be resolved for the staff to complete its review.

You requested that the submittal be exempt from mandatory public disclosure. While the staff has not completed its review of your request in accordance with the requirements of 10 CFR 2.790, your submittal is being withheld from public disclosure pending the staff's final determination.

If you have any questions regarding this matter, you can contact me at (301) 415-3139.

Sincerely,

Original Signed By:

Stewart L. Magruder, Project Manager
Generic Issues and Environmental Projects Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: As stated

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

October 1, 1996

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Generic Issues and Environmental
  Projects Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc w/encl: See next page

ABB Combustion Engineering, Inc.

cc: Mr. Ian C. Rickard, Director
Operations Licensing
ABB Combustion Engineering Nuclear Operations
Post Office Box 500
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering, Inc.
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

REQUEST FOR ADDITIONAL INFORMATION
CENPD-137, Supplement 2-P

An acceptance review was previously performed by SCIENTECH for the subject topical report. A number of questions and requests for additional information were made as a part of the acceptance review. ABB-CE provided responses to ten questions in Reference 1.

The following additional information is necessary for the detailed review of the subject topical report which describes proposed revisions to the ABB-CE SBLOCA evaluation model. The revised model, referred to by ABB-CE as S2M, is being reviewed for compliance with the requirements of 10CFR50.46 and Appendix K for CE PWR design plants.

1. The ORNL THTF small break LOCA heat transfer tests include rod surface temperatures up to about 1400°F. For licensing applications, the rod surface temperature can reach 2200°F. Since there is a lack of relevant test data in this range, ABB-CE must establish applicability of the proposed model in the temperature range of interest. The following two analyses is a way to establish applicability of the proposed model:

   a. ABB-CE should perform a sensitivity study for a limiting small break in the manner described below to demonstrate the margins in the proposed model. The limiting small break case should be run with the presently approved model (S1M) with the metal-water reaction model turned off and the required 1.2 multiplier on the decay heat. The power level should be set to achieve PCT's in the range of 2100 to 2200°F. Two variations of this case should then be compared to this base case. First, the case should be rerun with a decay heat multiplier of 1.0 as the only change. Second, the base case should be rerun using the S2M methodology as the only change. PCT comparisons should be shown for the three cases. The metal-water reaction should be turned off in all cases to avoid the run away response that this model introduces at high clad temperatures, so that the differences due to the model revisions can be clearly seen and compared to the known conservatism in the decay heat model.

   b. Benchmark the revised model against the best available data closest to the temperature range of interest. We believe that the high PCT data from the following reference is appropriate for that benchmark.

      EPRI-NP-1692, Vol. 1 and Vol. 2, "Heat Transfer Above the Two-Phase Mixture Level Under Core Uncovery Conditions in a 336-Rod Bundle", January 1981.

2. In the ABB response to question 9 of the acceptance review, ABB-CE argued that there was little difference in calculated PCT when the path length for radiation was changed from the hydraulic diameter to 0.85 times the hydraulic diameter, the value

used by ORNL (p 51 of NUREG/CR-2052). The results presented by ABB-CE showed that there was a very small difference in results when 0.85 times the hydraulic diameter was used instead of the hydraulic diameter. Since the use of a 0.85 multiplier is conservative and also in agreement with published reference sources, ABB-CE should commit to using 0.85 times the hydraulic diameter whenever any future changes are made to the PARCH/REM code.

3. Data presented in Figure 2-5 is used to determine steam emissivity at steam temperatures below 1700 F in the PARCH/REM model. The data presented in this figure is for gas (CO_2) and water vapor mixture at a pressure of 1 atm and a P_w (partial pressure of water vapor) approaching zero. A pressure correction factor that is inferred from work by Hottel is used to adjust the emissivity for reactor system pressure conditions. No information is given in the topical report about the impact of using data where the water vapor pressure approaches 0 on steam emissivity. Application of this data may introduce significant error in the emissivity under reactor conditions, particularly when correcting the pressure from 1 atm to reactor conditions (100 atm or more). Please provide justification for the use of this emissivity data for SBLOCA calculations using PARCH/REM. Discuss the sensitivity of steam emissivity on the cladding temperature prediction.

4. It appears that the denominator of the last term on the right side of equation (2-9) on page 2-11 should be $V_{c,i}$ and not $A_{c,i}$.

5. Units should be shown for all of the equations as has been done for equations (2-8) and (2-9). In equation (2-12) are the heat generation rates on a per unit length basis?

6. Please explain the basis for the weighting factor WF used for the linear interpolation of cladding and steam temperatures in equation (2-17).

7. In the PARCH/REM section of Appendix A, how is it determined which flag value for vectors 101-121 is appropriate for a given node? Is this a change from S1M?

8. On page 3-2 it is stated that the calculations did not require the analysis of the forced convection portion of the transient which is calculated by the STRIKIN-II computer code. Figure 1-1 shows that initial fuel and cladding temperatures for PARCH/REM are obtained from STRIKIN-II. How were initial fuel and cladding temperatures obtained for the calculations in Sections 3.2 through 3.4?

REFERENCES

1. ABB Combustion Engineering, "Response to NRC Acceptance Review of CENPD-137, Supplement 2-P", Enclosure 1 to letter LD-96-034, I. C. Rickard to USNRC, August 9, 1996.

ADDITIONAL REQUEST
CENPD-137, SUPPLEMENT 2-P

9. The enclosed non-proprietary submittal by Framatome Technologies Incorporated (FTI) considers a concern regarding the capability of some pressurized water reactor plants to meet the requirements of 10 CFR 50.46(b) for some small break loss of coolant accident (SBLOCA) scenarios. The scenarios of concern involve a problem with reactor coolant pump loop-seal clearing and are sensitive to the orientation of the break. The FTI report provides additional information to describe the concern and associated phenomena.

   ABB CE should address this issue for ABB CE designed (or fueled) plants to demonstrate that ABB CE SBLOCA evaluation models adequately address 10 CFR 50.46(b) requirements for those plants.

FRAMATOME
TECHNOLOGIES

Integrated Nuclear Services

JHT/96-46
July 15, 1996

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Supplementary Information to FTI's Response to NRC's Request for Additional Information on BAW-10168, Volume II, Revision 2, October 1992; RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants.

Reference: J. H. Taylor to Document Control Desk, "Response to NRC's Request for Additional Information on BAW-10168, Volume II, Revision 2, October 1992; RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," JHT/94-171, October 28, 1994.

Gentleman:

The reference transmitted FTI's response to an NRC request for additional information on topical report BAW-10168, Revision 2. The attachment provides supplemental information to the referenced response. The material enclosed herein is considered non-proprietary to Framatome Technologies.

Very truly yours,

J. H. Taylor, Manager
Licensing Services

cc: Frank R. Orr, NRC
R. B. Borsum
L. W. Ward, INEL - DC
C. P. Fineman, INEL - ID

3315 Old Forest Road, P.O. Box 10935, Lynchburg, VA 24506-0935
Telephone: 804-832-3000 Fax: 804-832-3663

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Break Discharge Coefficients: For SBLOCA, the leak flow requirements of 10CFR50.46 Appendix K have generally been interpreted as use of the Moody discharge correlation with a C_d of 1.0 for the entire two-phase flow regime. However, In BAW-10168 Revision 1, Volume II, Section 4.3.2.4, FTI proposed the use of realistic break discharge coefficients for SBLOCA calculations. Comparisons between the Moody discharge correlation and experimental data show that Moody overpredicts the leak flow rate for void fractions of 70 percent (corresponds to a quality of 10 percent at a pressure of 1000 psi) or greater. To account for this deficiency and better predict system depressurization. FTI's method used a C_d of 0.7 for void fractions of 70 percent or greater. For subcooled, superheated, and saturated discharges up to void fractions of 70 percent, a C_d of 1.0 was still used. FTI's break discharge methodology was NRC-approved based on our qualitative evaluation of the approach and with a request for a quantitative evaluation before or with its first application. FTI provided the NRC-requested evaluations with and in response to requests for additional information on Revision 2 of BAW-10168, Volume II.

After consultation with NRC personnel, it became clear that FTI's discharge model, while having a sound technical basis, would be considered as non-standard, requiring a substantial additional licensing effort. We have concluded that the expenditure of such an effort would currently not be productive. In point of fact, for most SBLOCAs the use of either method would produce comparable trends and results, since little time is spent at leak void fractions where significant differences are noted between Moody and test data. Therefore, FTI is modifying its SBLOCA break flow model to reflect the common interpretation of Appendix K. A discharge coefficient of 1.0 will be used regardless of leak flow quality--subcooled, saturated, or superheated. Discharge correlations--Extended Henry-Fauske (subcooled), Moody (saturated), and Murdock-Bauman (superheated)--will remain unchanged. This, coupled with a break spectrum, complies with the intent and requirements of Appendix K for SBLOCA.

This switch in methodology will not invalidate the studies and benchmarks performed in support of Revision 2 nor will FTI totally abandon the use of its more accurate modeling technique. FTI will reanalyze SBLOCA cases having clad temperatures in excess of 1800 F using its variable C_d model. Reductions in the rate of system depressurization occurring during the "core boildown" (or high void phase of the transient), resulting from the use of the variable C_d method, can adversely impact ECC injection, core inventory, and possible lead to clad temperature increases above those predicted using the normal Appendix K technique. Analyzing high temperature SBLOCA transients using both Appendix K and our variable C_d methods will assure that the PCT is not underpredicted. SBLOCA transients below 1800 F are not highly susceptible to large clad temperature changes resulting from items such as the incidence of rupture and its accompanying inside/outside metal-water energy addition; the reverse becoming true as temperatures climb above 1800 F. At and above 1800 F, the energy contribution from the metal-water reaction is becoming increasingly significant. For those cases just below 1800 F, a reasonable safety margin of at least 400 F to the PCT criterion is provided. Hence, 1800 F is a logical transition point between analyzing a SBLOCA transient using only the Appendix K method and analyzing the case using both methods.

In summary, FTI will use a discharge coefficient of 1.0 for the entire two-phase leak flow regime. All other aspects of our break modeling will remain unchanged. This methodology complies with the intent and requirements of Appendix K for SBLOCA. For SBLOCA transients predicting clad temperatures above 1800 F using the Appendix K technique, FTI will also analyze such cases using its variable C_d method. 1800 F will be the established transition point. Analyzing such cases with both techniques assures that the PCT will be conservatively predicted.

Partial Loop Seal Clearing: In response to questions regarding partial loop seal clearing, several additional SBLOCA cases were run using the plant model shown in Figure 1. Break sizes were varied from 1.6 to 2.0-inch to study the transition from no loop seal clearing to the clearing of the broken loop. It was found that RELAP5/MOD2-B&W predicts this transition for breaks between 1.9 and 2.0-inches. The liquid levels in the broken loop pump suction piping for these two cases are shown in Figures 2 and 3. Figure 4 shows the core liquid levels for the two cases. From Figure 4 it can be observed that the minimum core liquid levels of about 9.0-ft occur at about 1600 seconds and increase thereafter. For the 2.0-inch break, the core liquid level is about 10.0-ft at the time of loop seal clearing. The loop seal spillunder elevation corresponds to 3.0-ft height from the bottom of the core.

The steam velocity in the upside pump suction piping for the 2-inch break is shown in Figure 5. Once the steam venting process initiates, the head imbalance in the loop seal accelerates the steam flow and can be expected to reach a terminal velocity sufficient to clear the loop seal. For the 2-inch beak the terminal steam velocity in the upside pump suction piping reaches about 10.0 ft/s at the time of loop seal clearing as shown in Figure 5. Tuomisto and Kajanto[1] show that the loop will clear completely for steam velocity greater than 6.2 ft/s (1.9 m/s) at 870 psia (60 bar). This is based on the flooding criterion for large diameter vertical pipes, Kutateladze Number Ku (See Equation 5 in Reference 1) equals 3.2. This flooding criterion is defined as a zero downward flow of falling film on the tube surfaces. They also show that, at pressures above about 145 psia (10 bar), vertical flooding is the limiting mechanism for loop seal clearing rather than the droplet entrainment from the stratified liquid in the horizontal section of the loop seal. For the 2.0-inch break case, the system pressure is about 1000 psia and therefore the loop will clear for steam velocities lower than 6.2 ft/s. The 1.9-inch break case in ROSA (see response to Question 14) demonstrates the loop seal clearing mechanism discussed above. For these break sizes, it is possible to accumulate some of the liquid in the loop seal once the initial acceleration of steam is complete as observed in the test. This liquid fall back is also observed in the RELAP5 simulation of the 1.95-inch break case which is discussed at the end of this section.

Figure 3 shows that the liquid level in the upside of the loop seal section starts to decrease after about 1700 seconds. The void fractions in Nodes 255, 260, and 265 are shown in Figures 6 through 8, respectively. From these figures it can be seen that the liquid level decrease in the loop seal upside section is caused by the increase in void fraction in the pump volume (Node 260). Steam venting from the loop seal occurs only after about 2200 seconds as shown in Figure 6. The pump discharge piping on the other hand is highly voided after about 750 seconds due to the steam flow from the upper head spray nozzles into the downcomer. At about 1600 seconds the break junction void fraction increases rapidly from zero to a highly voided state and the flow in the cold

leg starts to oscillate. Injection of the cold ECC water into the highly voided cold leg and the break node amplify these oscillations. This results in a flow of steam from the cold leg into the pump volume. Note that in the broken loop, up until loop seal clearing, the HPI water is injected in to the Node 276 (a vertical node), and the CCI water is injected into the cold leg. The equilibrium option is selected in Node 276, making Node 276 a major source of oscillations. Stratified flow is expected in the pump discharge piping and RELAP5 allows only small condensation when the flow is stratified. The voiding of the pump node prior to loop seal clearing is discussed further in the next section.

To further study the possibility of predicting partial loop seal clearing, a 1.95-inch break case was run. The broken loop also cleared for this case. However, some liquid remained in the upside section and in the pump node, possibly as a liquid film on the pipe walls that fell back after the high steam flow period ended. This water eventually accumulated in Node 255 as shown in Figure 9. All other nodes in the loop seal were almost completely voided. The liquid did not fall into node 250, which represents the lowermost portion of the U-bend. This is consistent with the discussion in Reference 1.

Pump Noding Sensitivity Study

The broken loop pump suction noding for the base model is shown in Figure 10. To reduce early loop seal clearing, Node 248, representing the lower portion of the downside piping, was set at a small node height, 1 foot. The bottom of Node 248 coincides with the spill under elevation of the loop seal. Node 250 represents the horizontal portion of the U-bend and the height of this node is the radius of the pipe. Node 260 represents the pump. The height of Node 260 is 5.81 ft which is the actual height of the pump up to the centerline of the discharge piping. In RELAP5, the pump volume also uses the high mixing flow regime, and, therefore, slug flow (Wilson drag) is not used in this node, even though it is a vertical node.

The early voiding of the pump node for the 2.0-inch break case, as discussed in the previous section, may have been caused by the height of Node 260. To study the sensitivity of pump node size, the base input model was modified by dividing the pump volume into three nodes (259-1, 259-2, and 260) as shown in Figure 11. Node 260 still represents the pump. In this case the 2.0-inch break case did not clear the loop seal. For a 2.1-inch break case, the loop seal cleared after about 3300 seconds. Collapsed liquid levels in the loop seal and core and the void fractions in the loop seal nodes, pump node, and the pump discharge node of the broken loop are shown in Figures 12 through 22. From these figures the following observations can be made. Steam venting through the loop seal starts after Node 245 is highly voided. This occurs at about 1400 seconds. The void fraction in Node 259-1, which is part of the actual pump, is close to the void fraction in Node 258. Node 260 is highly voided and the void fraction in node 259-2 is somewhere between the values for Nodes 259-1 and 260. The void distribution in the upside U-bend, including the pump volume, is improved over that in the base calculation. The venting of steam causes the liquid level in the upside of the U-bend to decrease, reducing the core level depression. Figures 12, 14, and 15 show liquid level oscillations on the order of 1.0 foot in the down side of the U-bend from about 1500 seconds until the time of loop seal clearing, about 3300 seconds. The oscillations are mainly caused by the condensation of steam on the cold ECC water

injected into the cold legs. Rothe, Wallis, and Thrall[2] discussed the pressure and flow oscillations due to the condensation of steam on ECC water in the cold legs. CE[3] and Westinghouse 1/14 scale tests[2] (See Table X in Reference 2) both show condensation induced pressure oscillations on the order of 10 to 20 psi. Therefore, the RELAP5 calculated 1.0 foot oscillations are reasonable.

Conclusion

From this study the following conclusions are made. The transition from no loop seal clearing to clearing of one loop occurs within a narrow range of break sizes. Condensation-induced oscillations causes steam venting through the loop seal before the liquid level in the downside section of the loop seal reaches the spillunder elevation. This substantially reduces the possibility of core uncovery at the time of loop seal clearing for these break sizes. The core never uncoverd for the break sizes studied here.

The revised pump noding will be used in SBLOCA EM. However, this model change does not impact previous EM studies and benchmarks.

References

1. H. Tuomisto and P. Kajanto, "Two-Phase Flow in a Full Scale Facility," Nuc. Engg. And Design 107, pp 295-305, 1988.

2. P. H. Rothe, G. B. Wallis, and D. E. Thrall, Cold Leg ECC Flow Oscillations, EPRI NP-282, November 1976.

3. W. E. Burchill, P. A. Lowe, and J. R. Brodrik, Steam-Water Mixing Test program Task D: Formal Report for Task B and Final Report for the Steam Relief Phases of the Test Program, CENPD-101, AEC-C00-2244-1, October 1973.

:F Proprietary

Figure 5.9-2

RELAP5 Small Break LOCA Model

Sequoyah Noding for Primary Loops with Model 51 SGs

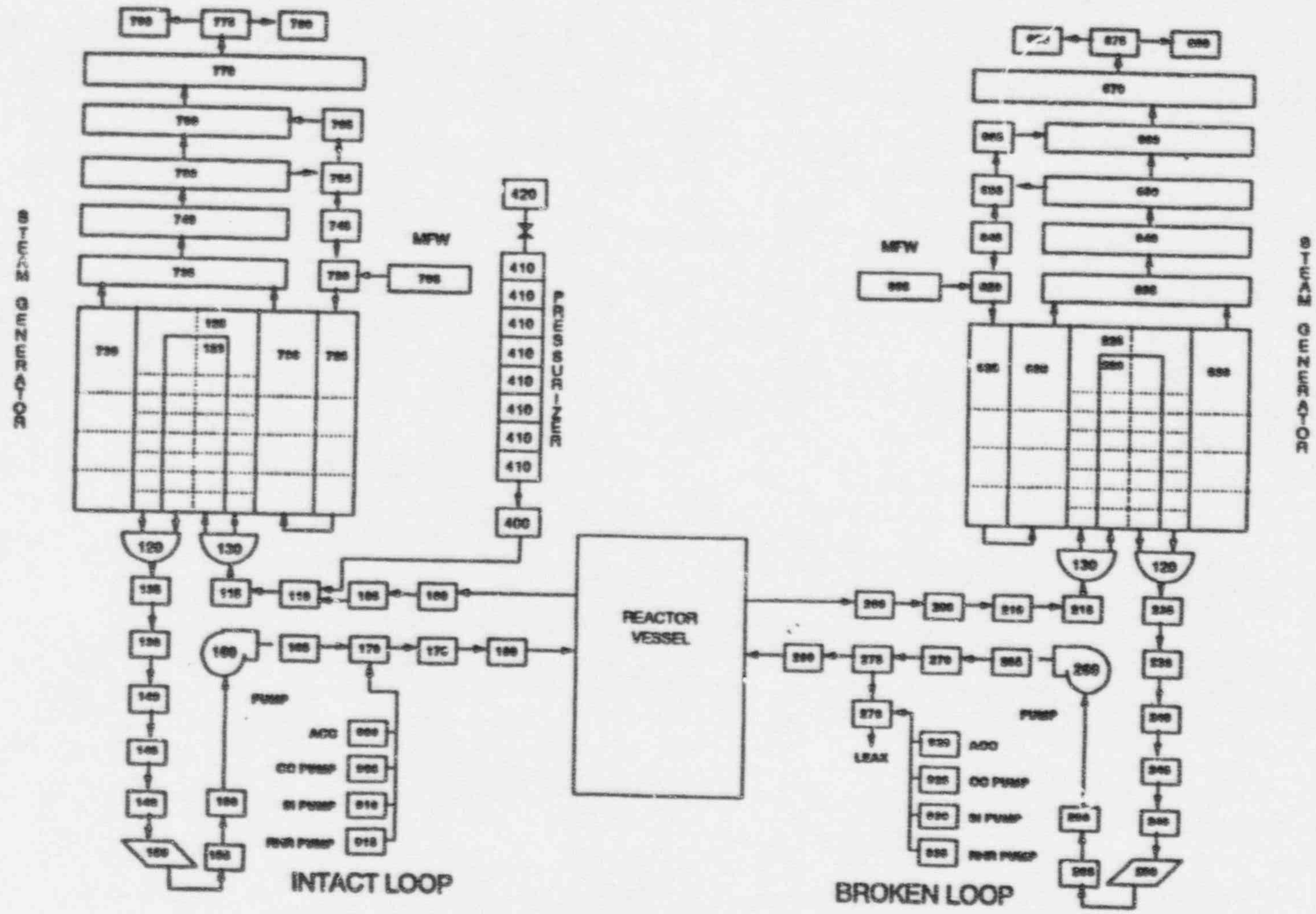


Fig-1

5

R5/2 1.9 INCH PD BREAK
RELAP5/MOD2 Ver 20.0HP

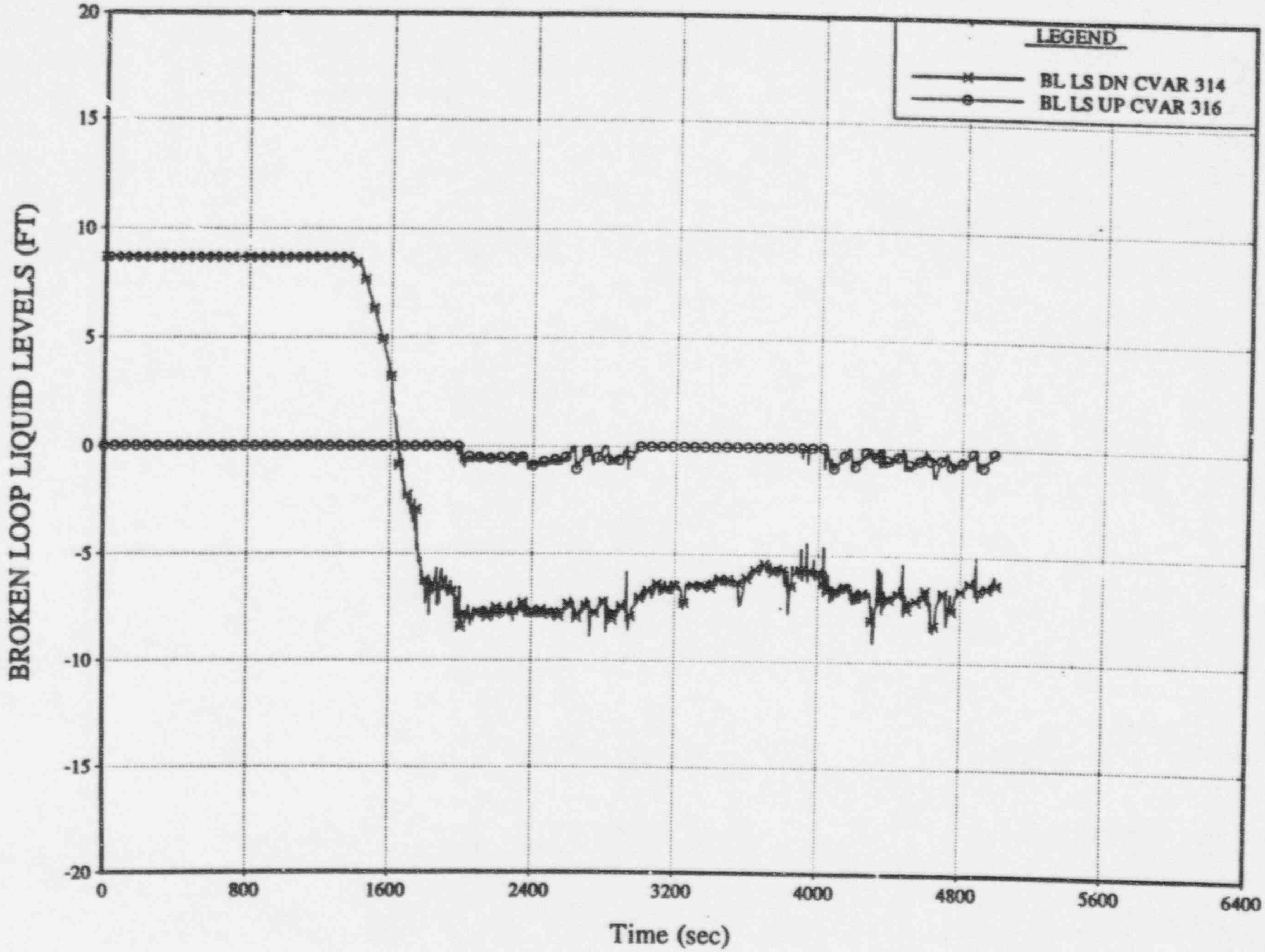


Fig-2

Fig 2

R5/2 2.0 [illegible] INCH PD BREAKS
RELAP5/MOD2 Ver 20.0HP

Fig-3

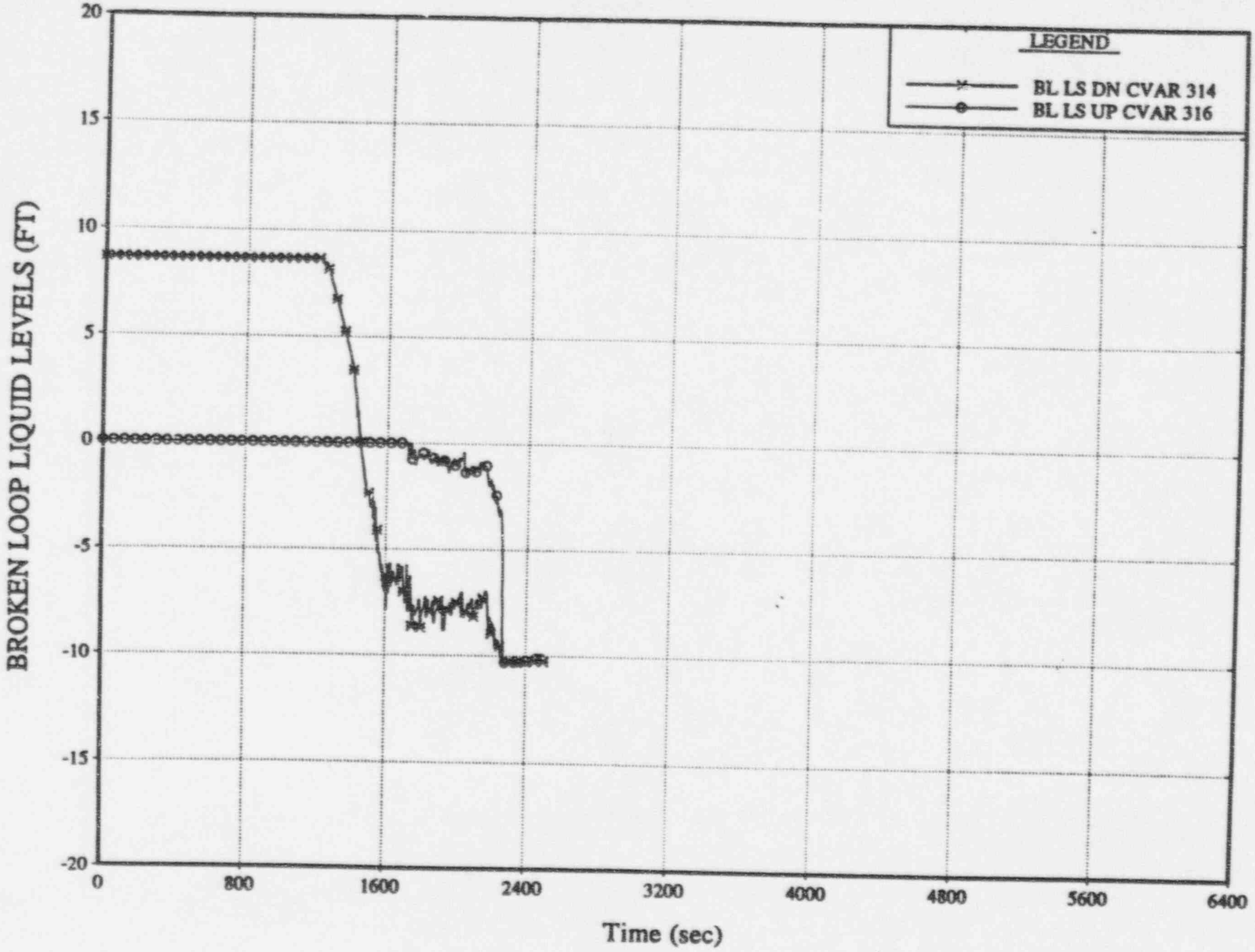


Fig. 3

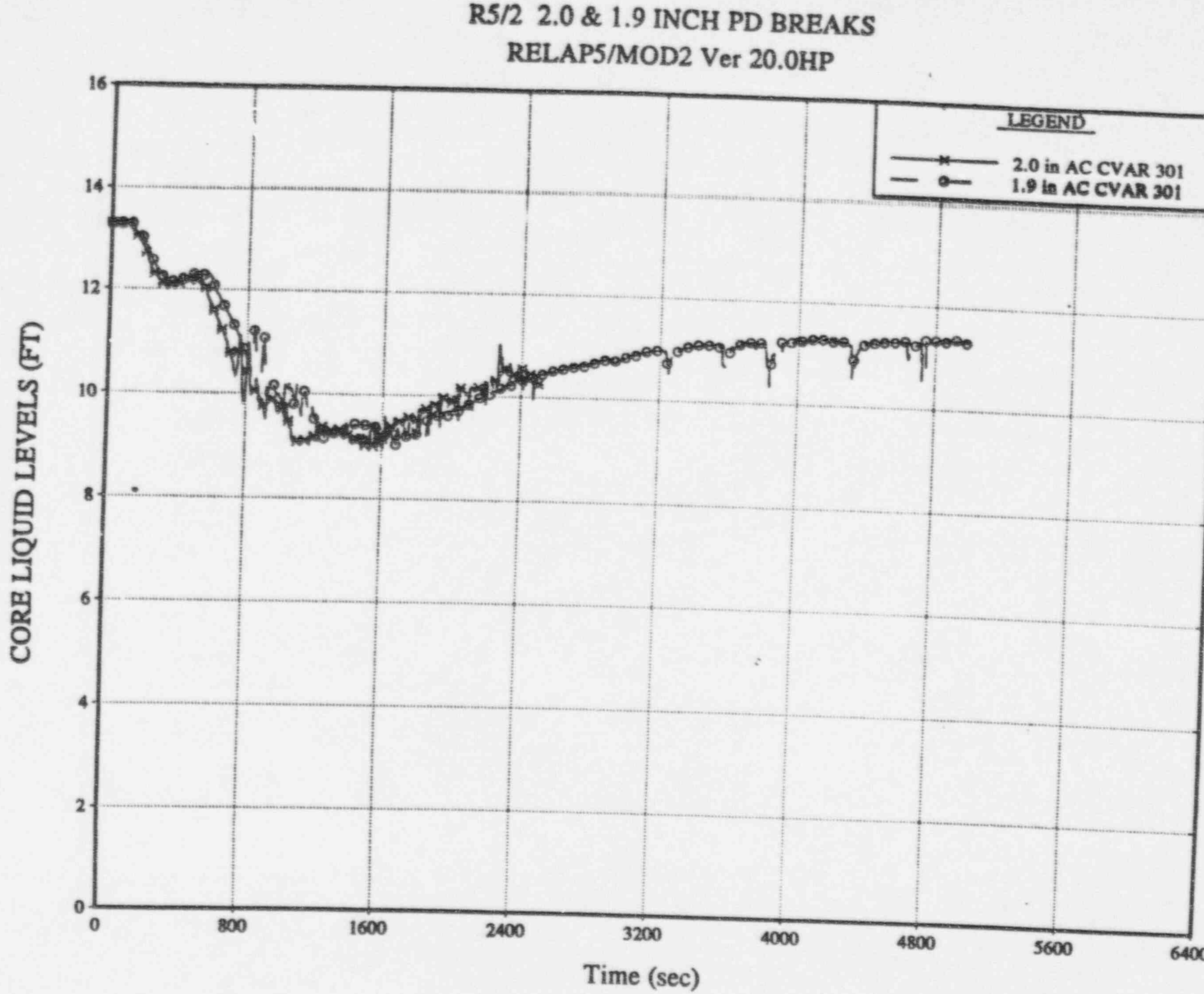

R5/2 2.0 & 1.9 INCH PD BREAKS
RELAP5/MOD2 Ver 20.0HP
LEGEND
2.0 in AC CVAR 301
1.9 in AC CVAR 301
CORE LIQUID LEVELS (FT)
Time (sec)
0
2
4
6
8
10
12
14
16
800
1600
2400
3200
4000
4800
5600
6400

Fig - 4

Fig 4

R5/2 2.0 INCH PD BREAK
RELAP5/MOD2 Ver 20.0HP

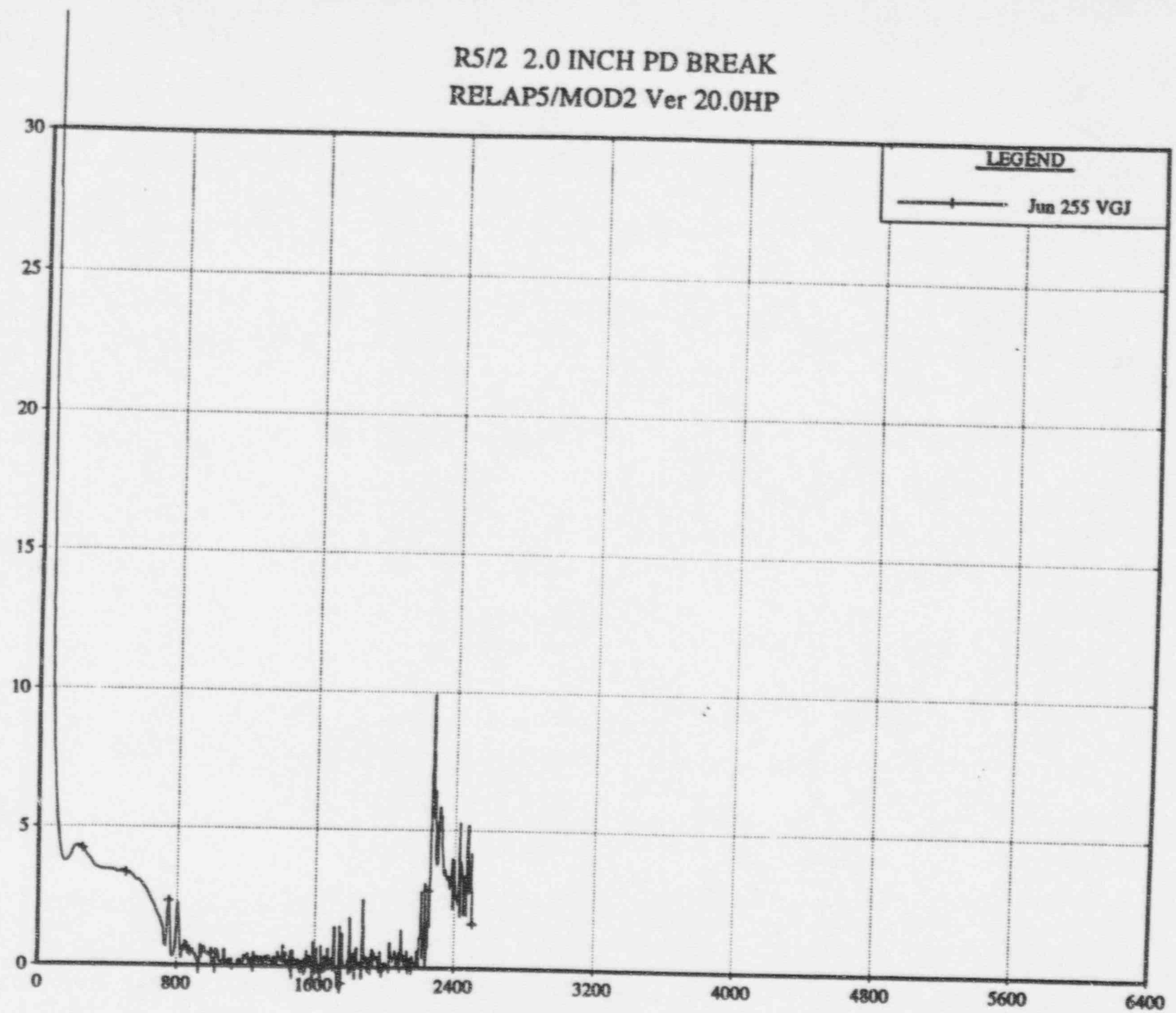


Fig 5

Fig-5

6

SQN 2.0 in SBLOCA Case, sqnsb2b.in

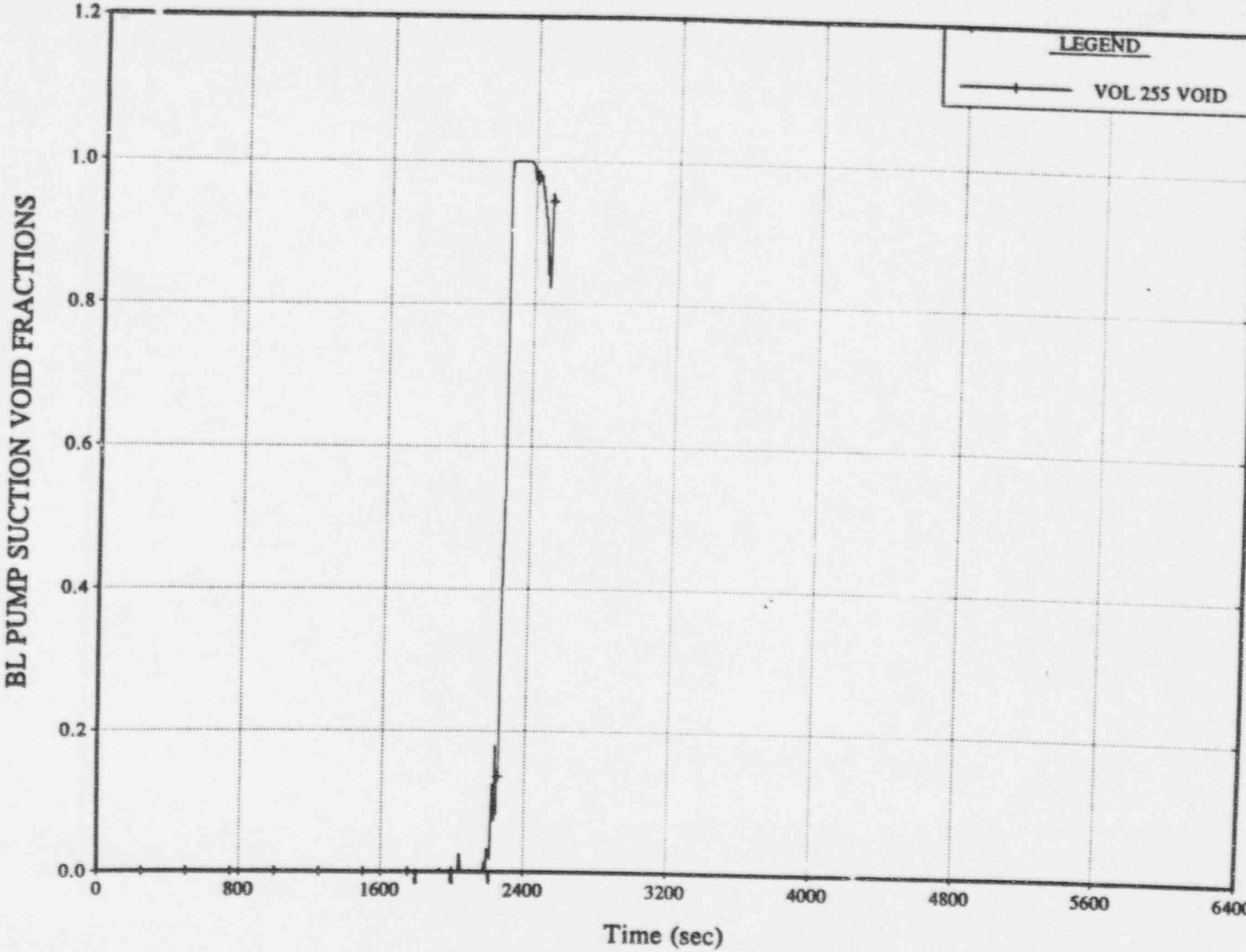
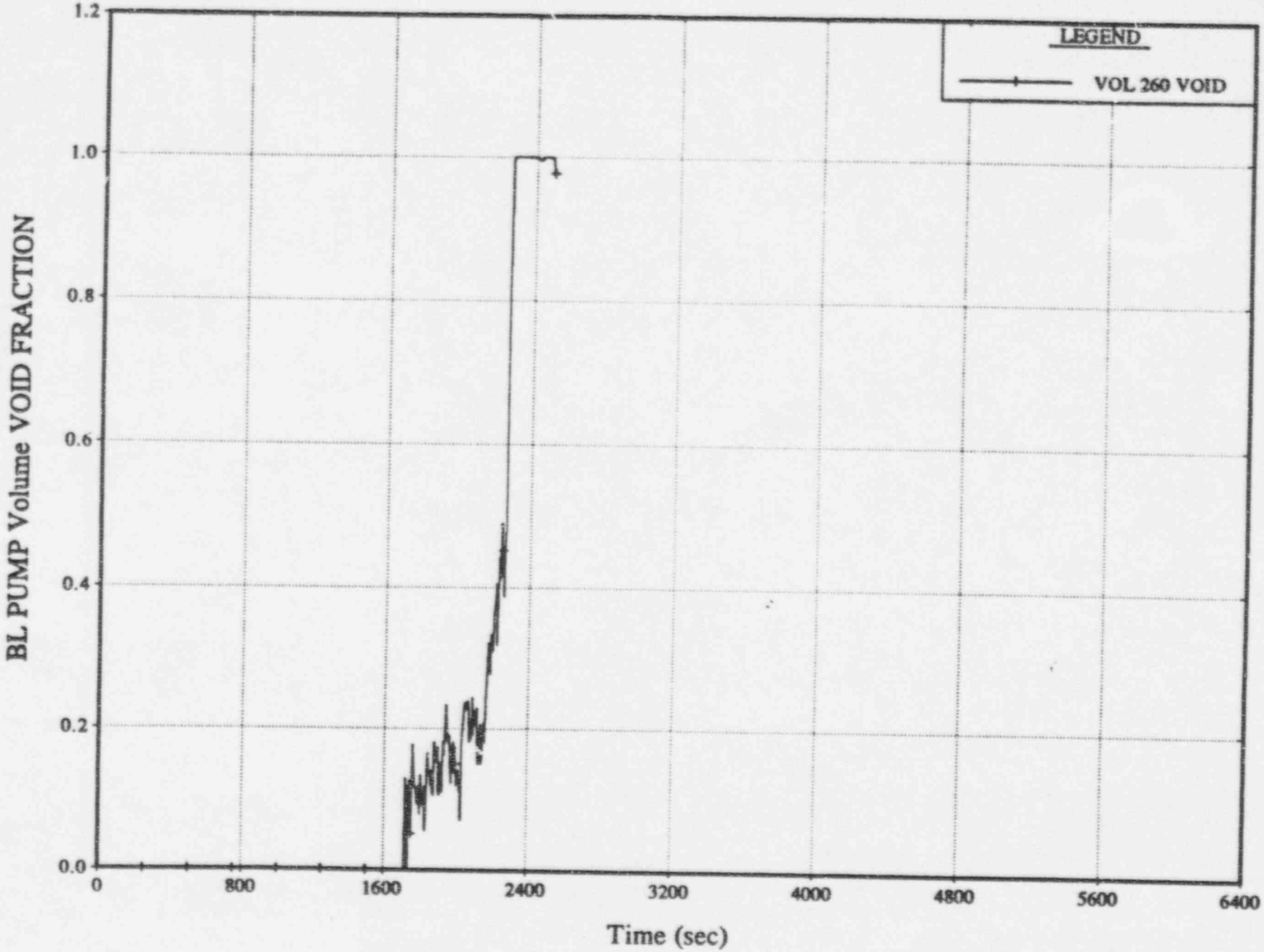


Fig 6

FAID VOIDG001

SQN 2.0 in SBLOCA Case, sqnsb2b.in



11 Fig - 7

Fig. 7

SQN 2.0 in SBLOCA Case, sqnsb2b.in

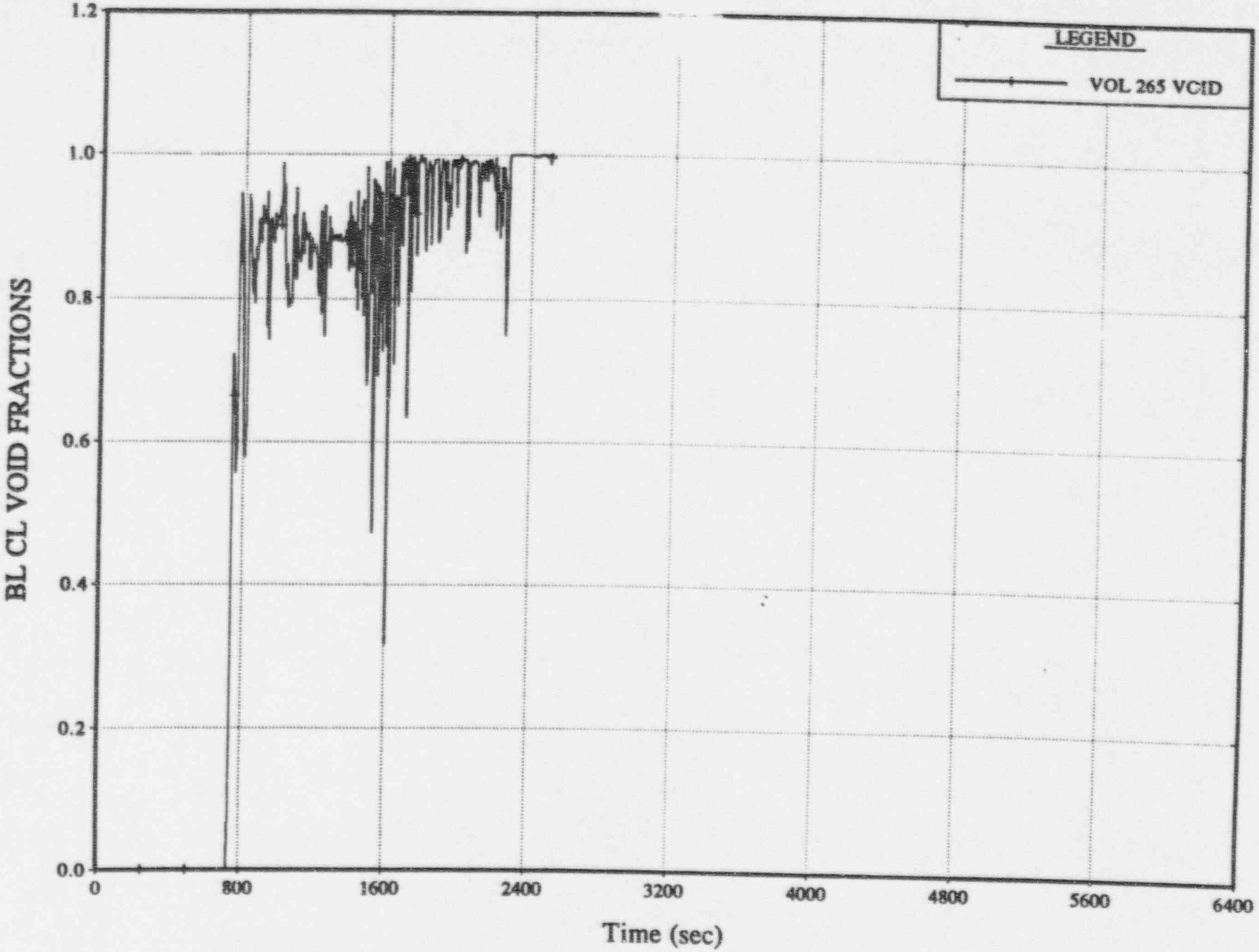


Fig - 8 - 12

Fig 8

FAID VOIDG005

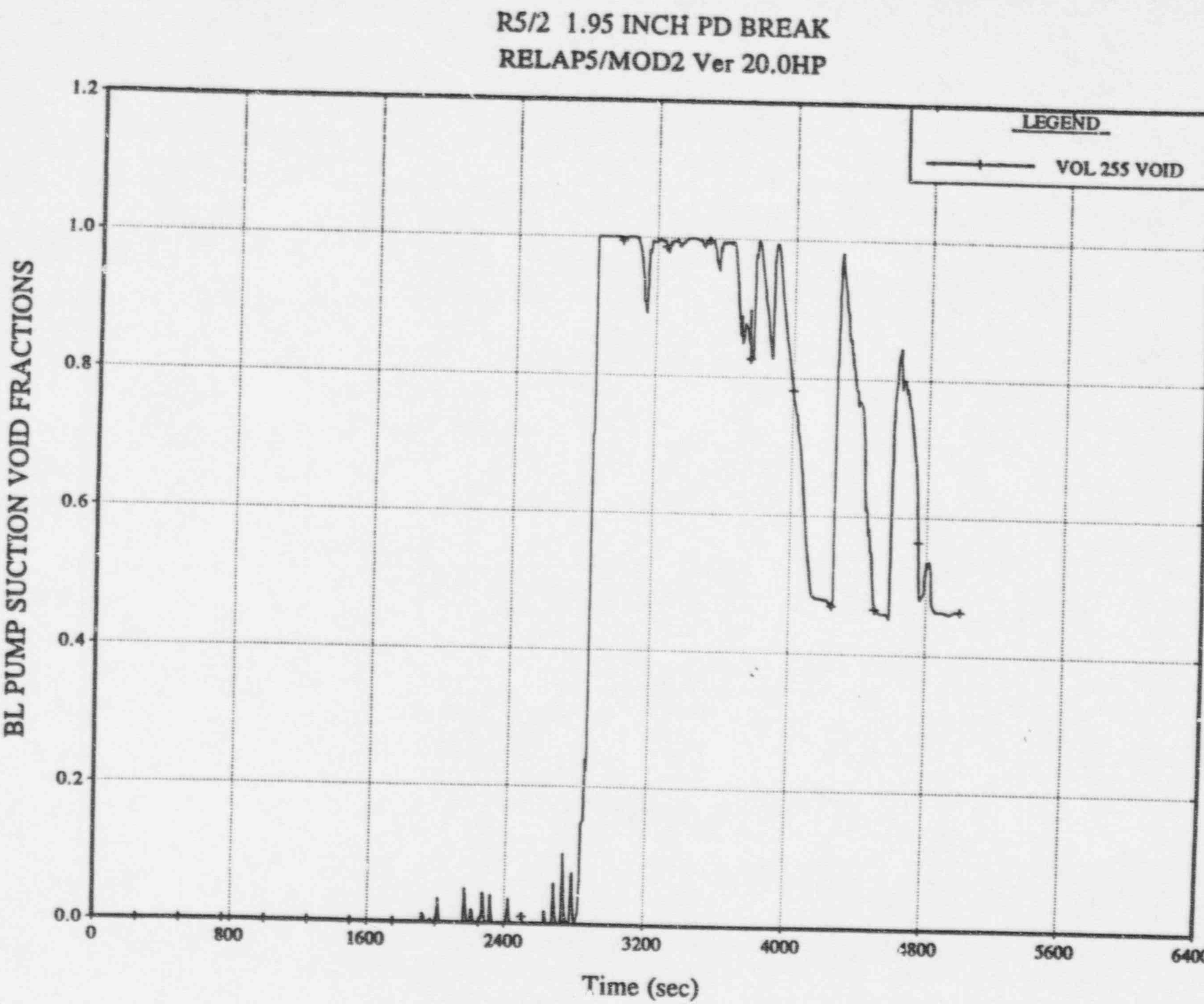

R5/2 1.95 INCH PD BREAK
RELAP5/MOD2 Ver 20.0HP
LEGEND
VOL 255 VOID
BL PUMP SUCTION VOID FRACTIONS
0.0
0.2
0.4
0.6
0.8
1.0
1.2
0
800
1600
2400
3200
4000
4800
5600
6400
Time (sec)

Fig 9

Fig-9 13

Figure 10: Typical FTI SBLOCA 4-Loop RSG Pump Suction Noding Details

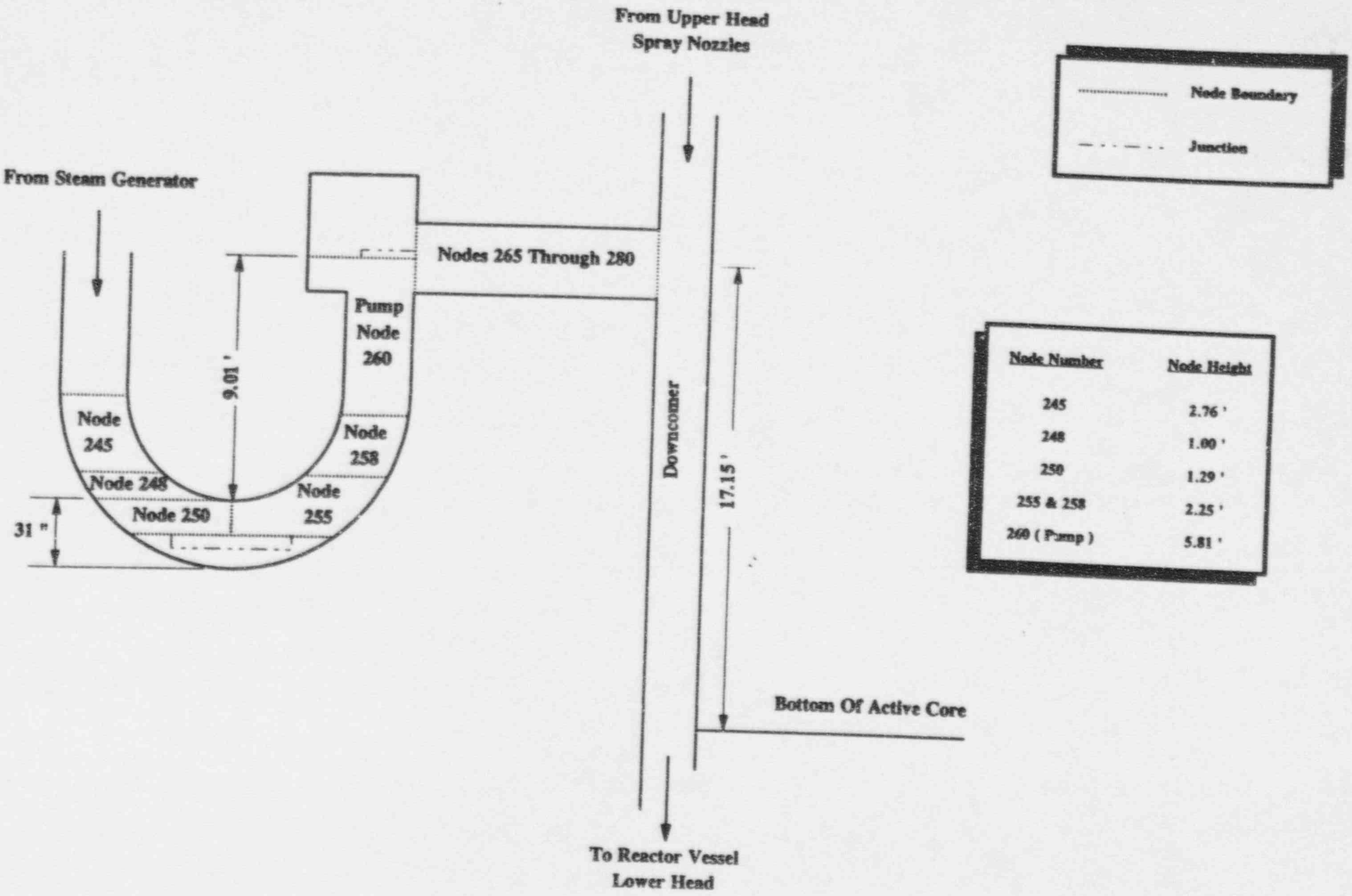
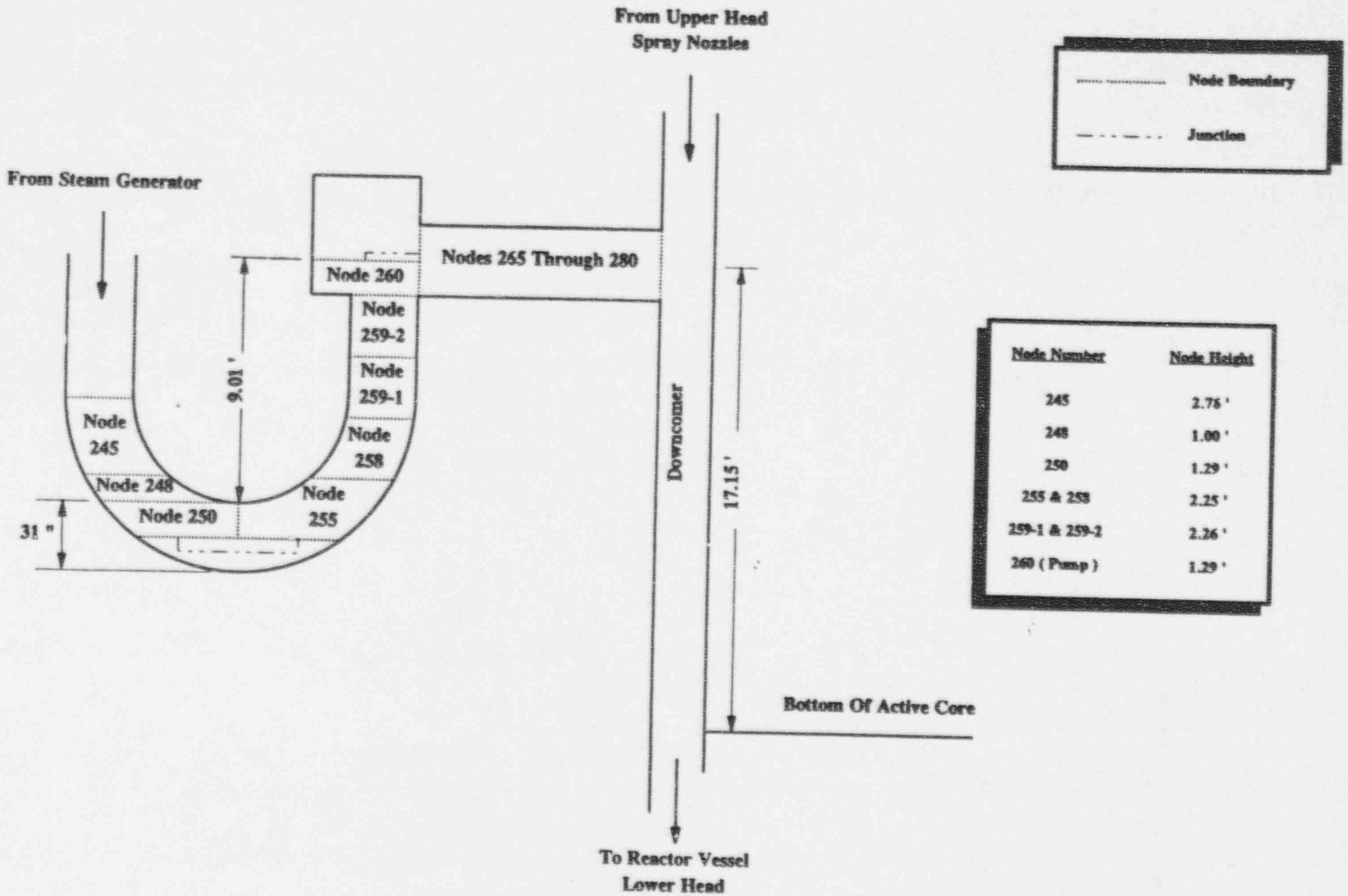


Fig-10

14

Figure 11: Revised FTI SBLOCA 4-Loop RSG Pump Suction Noding Details



Node Number	Node Height
245	2.76 '
248	1.00 '
250	1.29 '
255 & 258	2.25 '
259-1 & 259-2	2.26 '
260 (Pump)	1.29 '

Fig-11

15

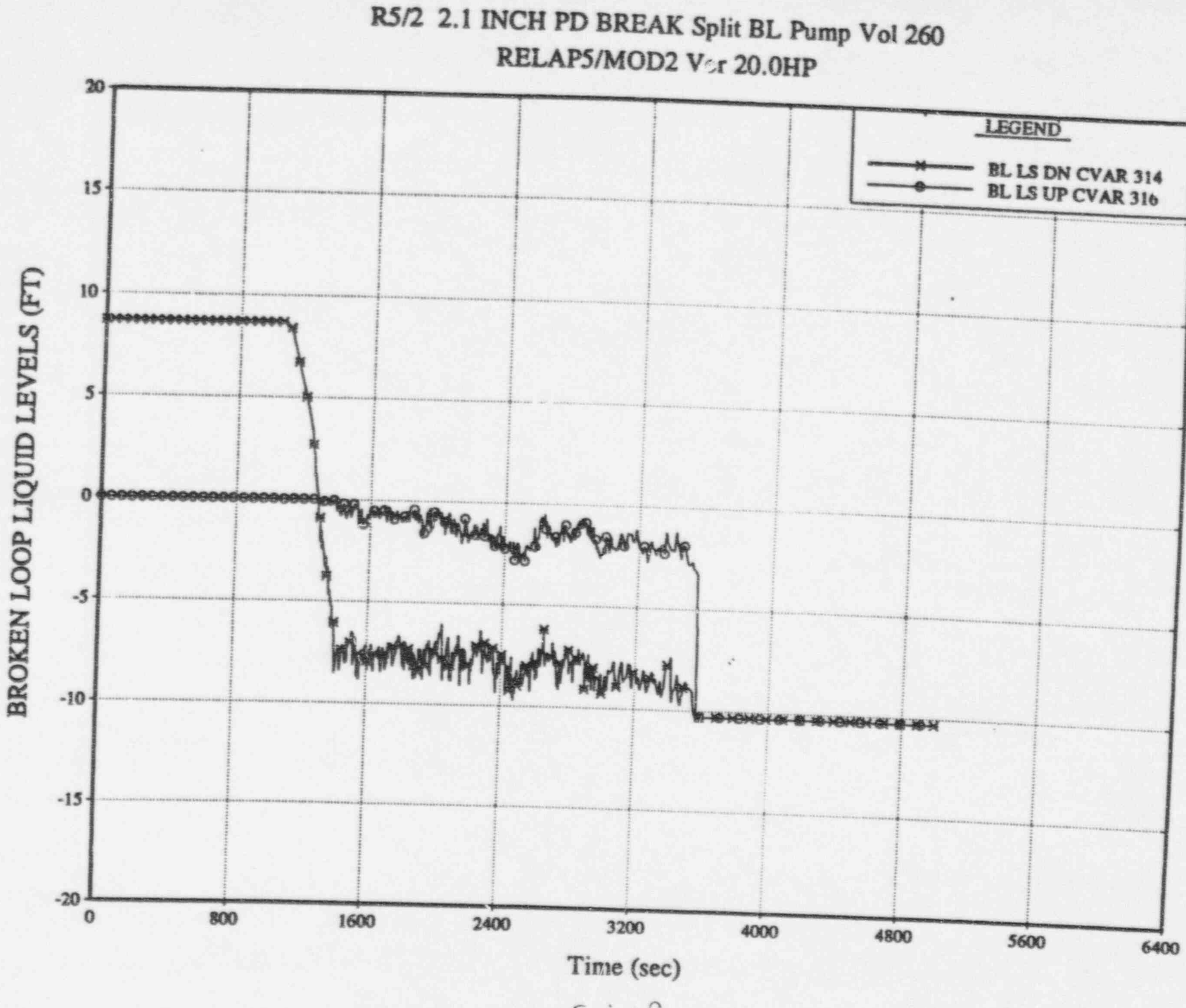


Fig-12

Fig 12

16

TATB VWRO CNTRL040

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

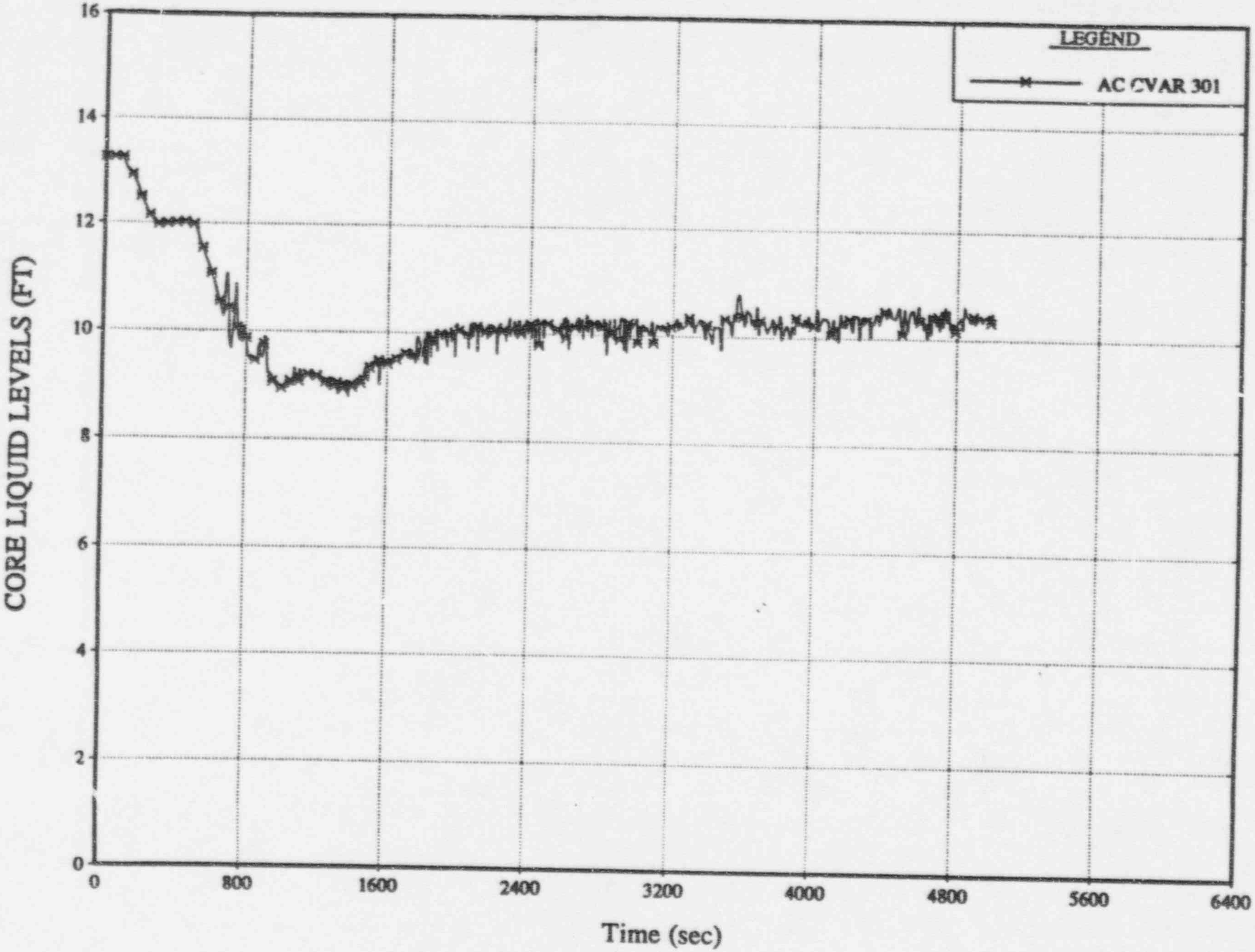


Fig. 13

Fig-13

17

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

Fig - 14

18

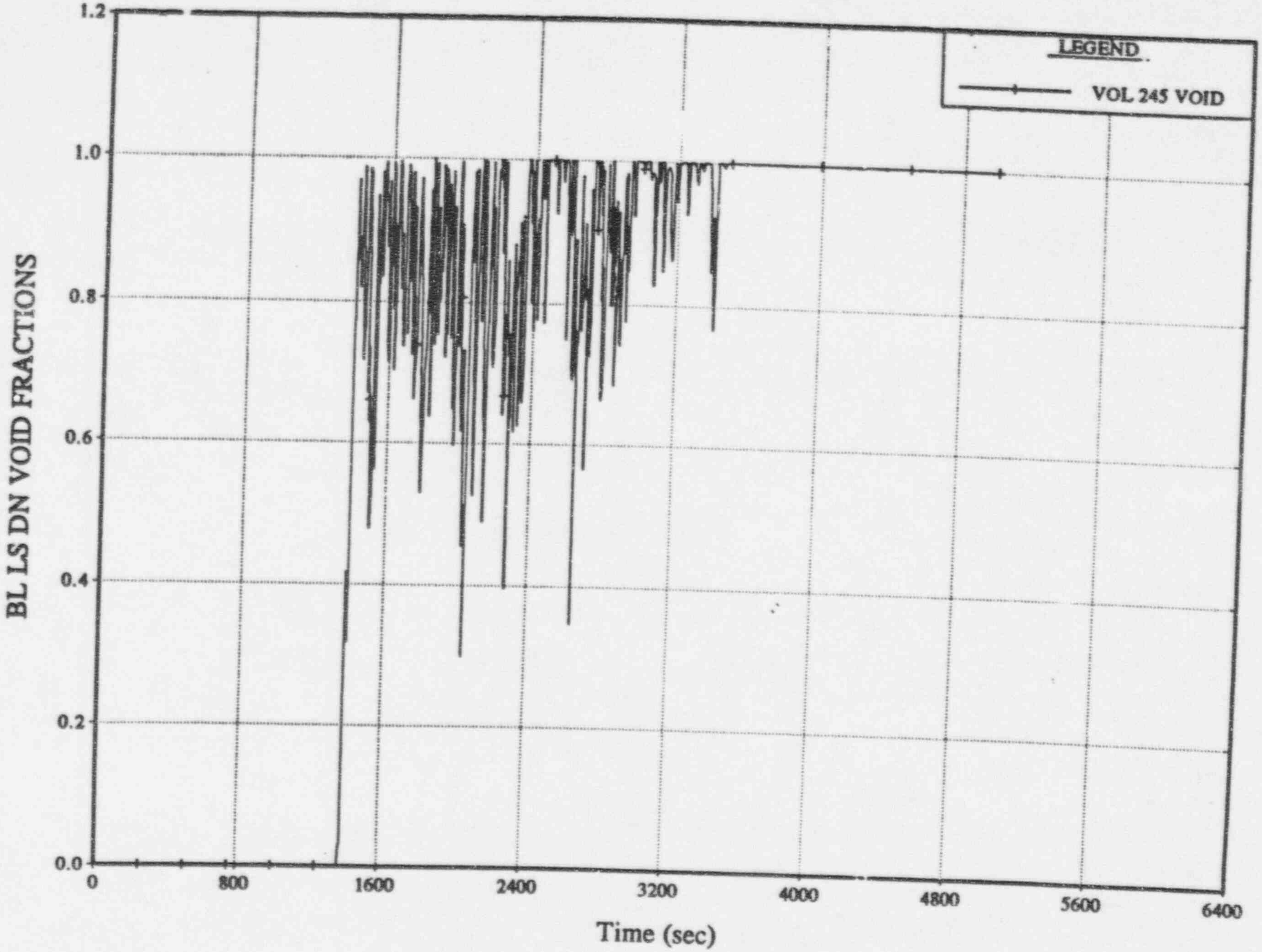


Fig 14

TATR VWRO VOIDG007

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

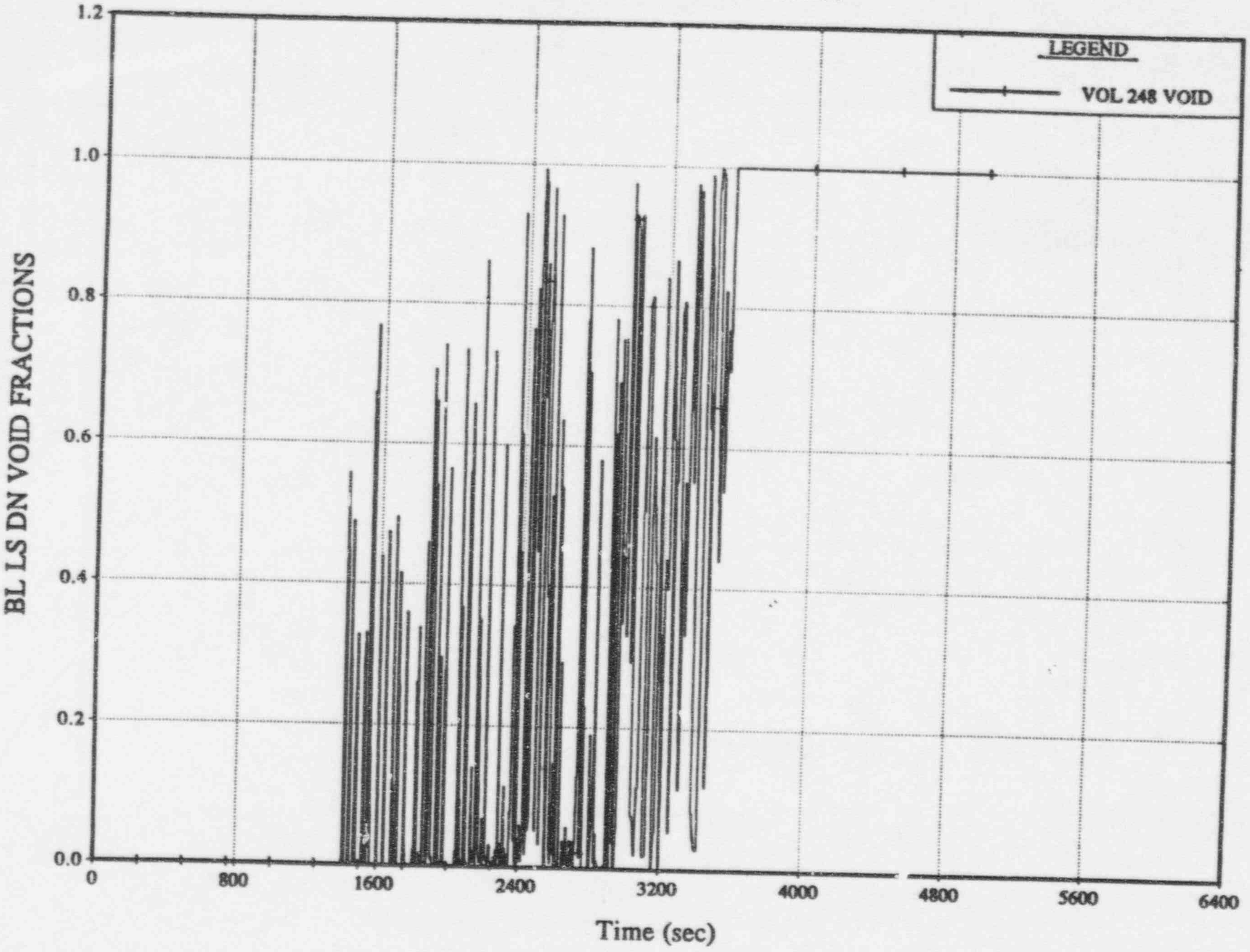


Fig 15

Fig -15

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

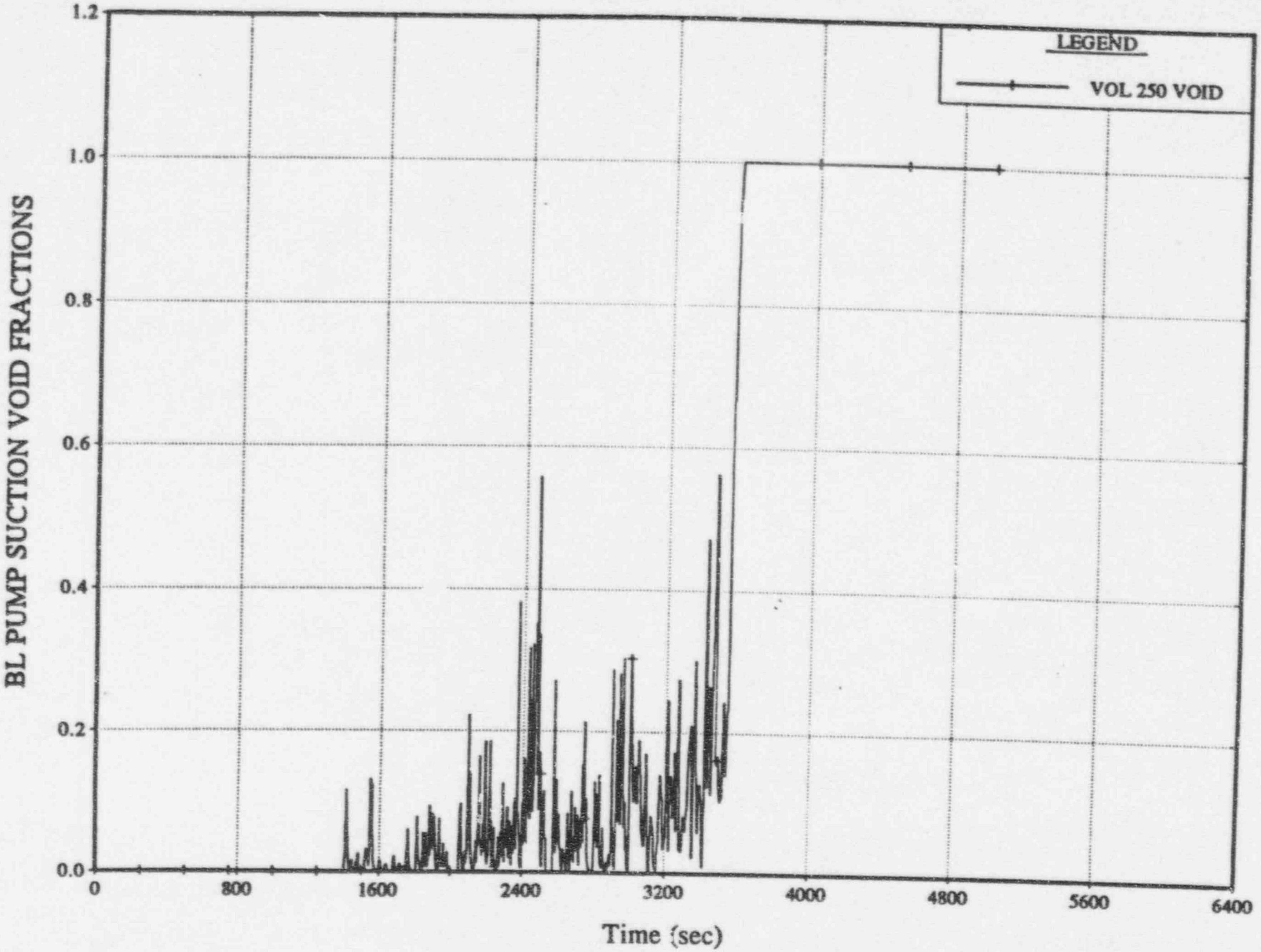


Fig 16

Fig-16

TATB VWRO VOIDG004

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

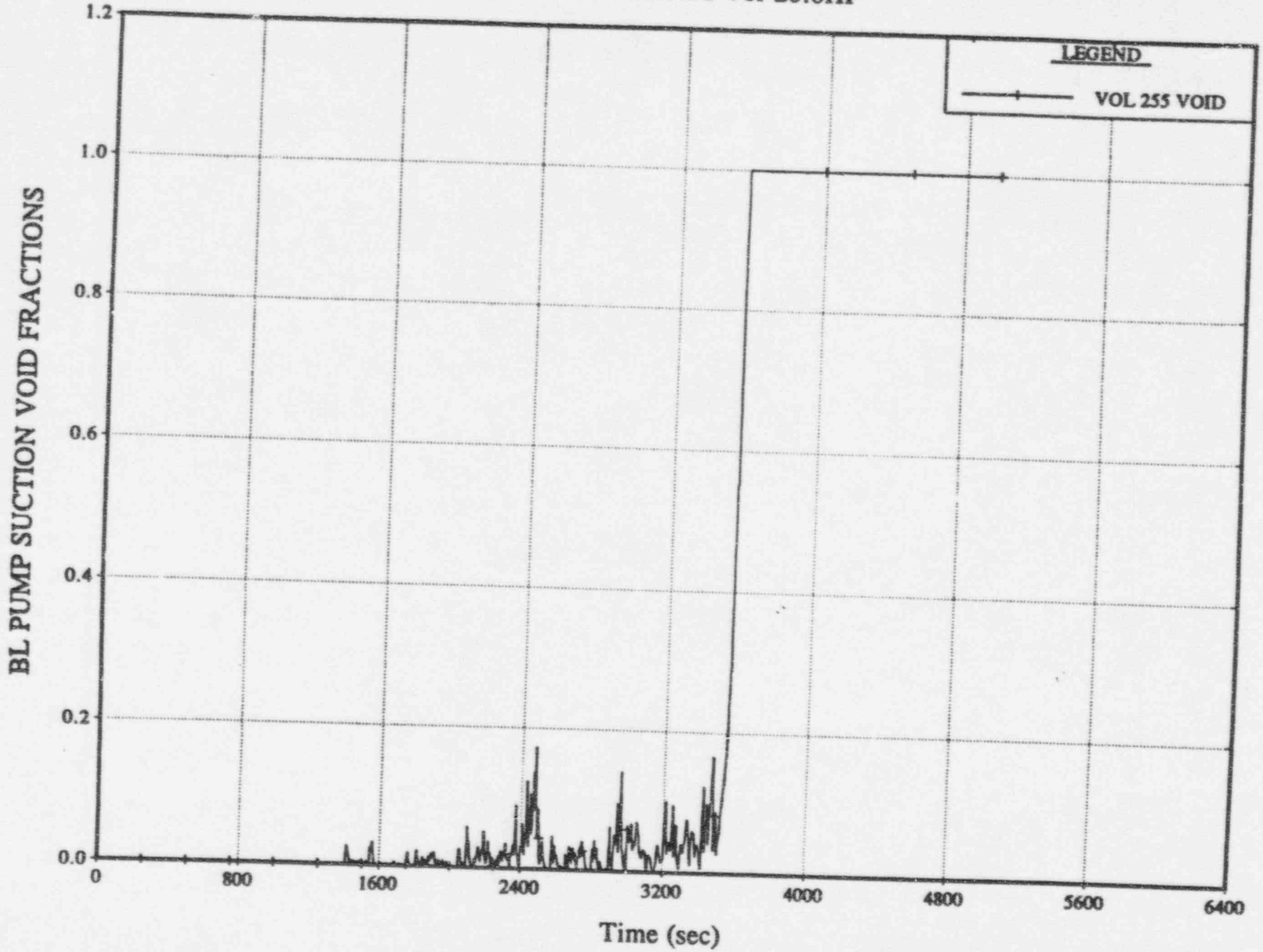


Fig-17

F.. 17

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260

RELAP5/MOD2 Ver 20.0HP

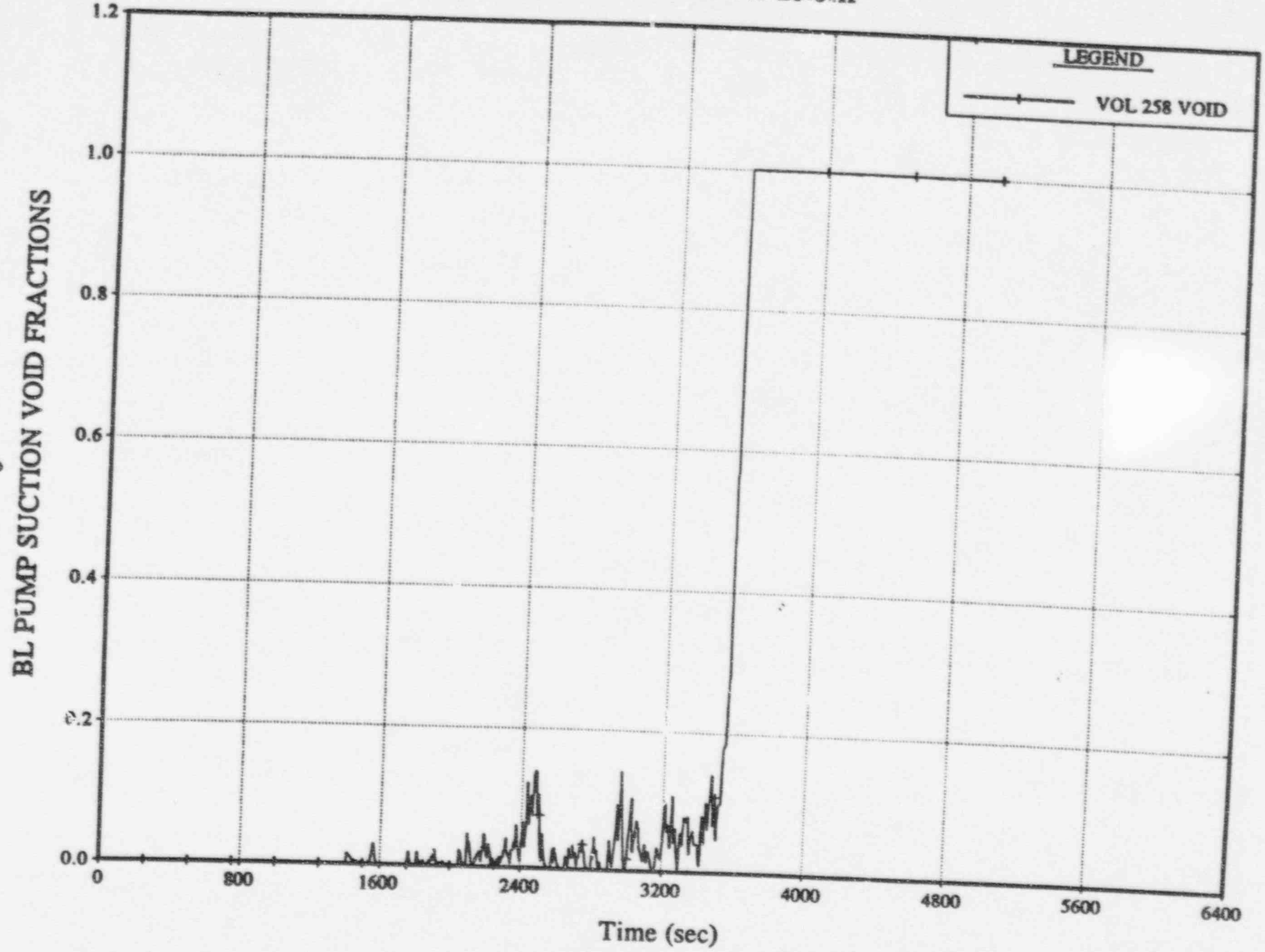


Fig 18

Fig 18

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TATB VWRO_VOIDG002

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

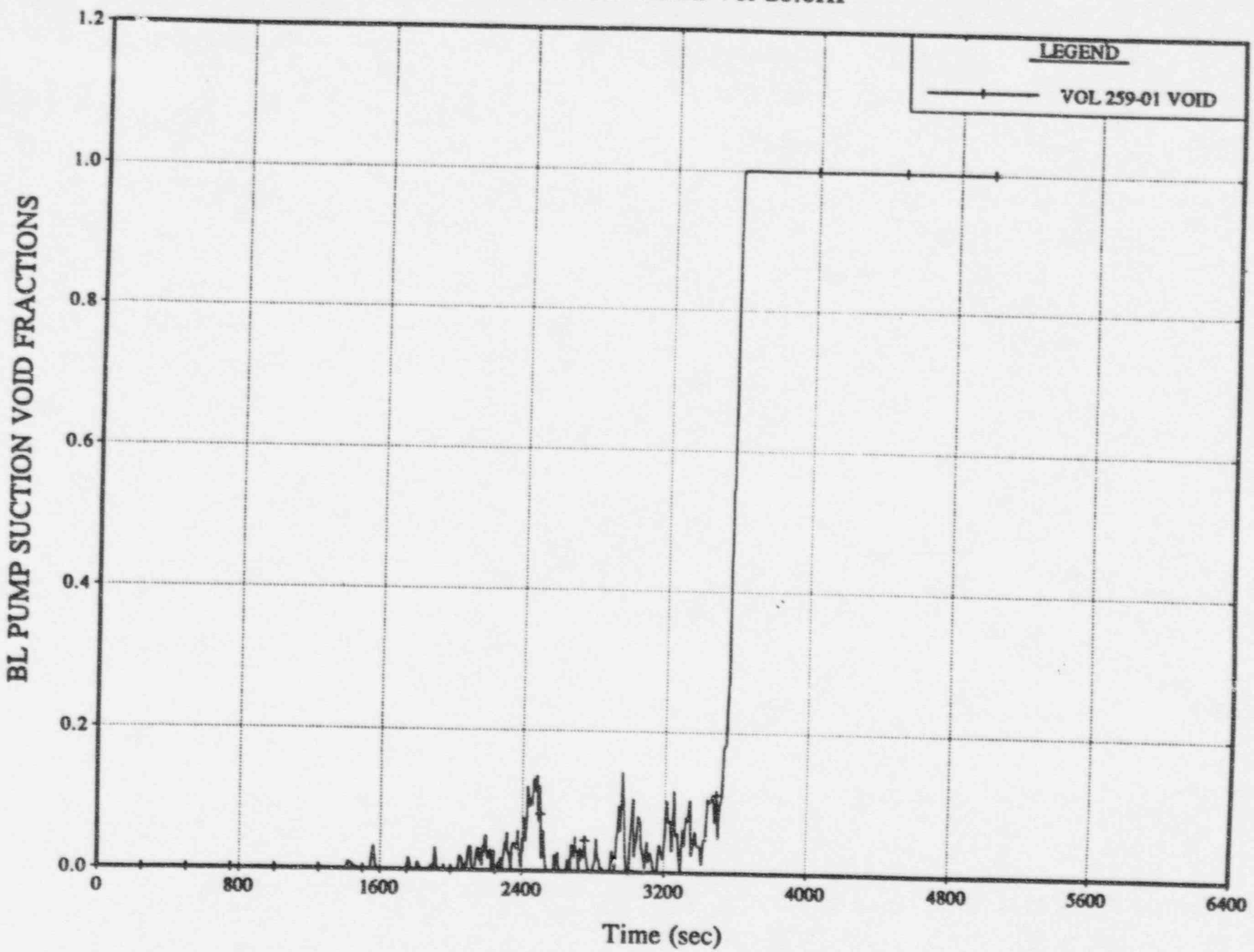

Fig-18

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R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

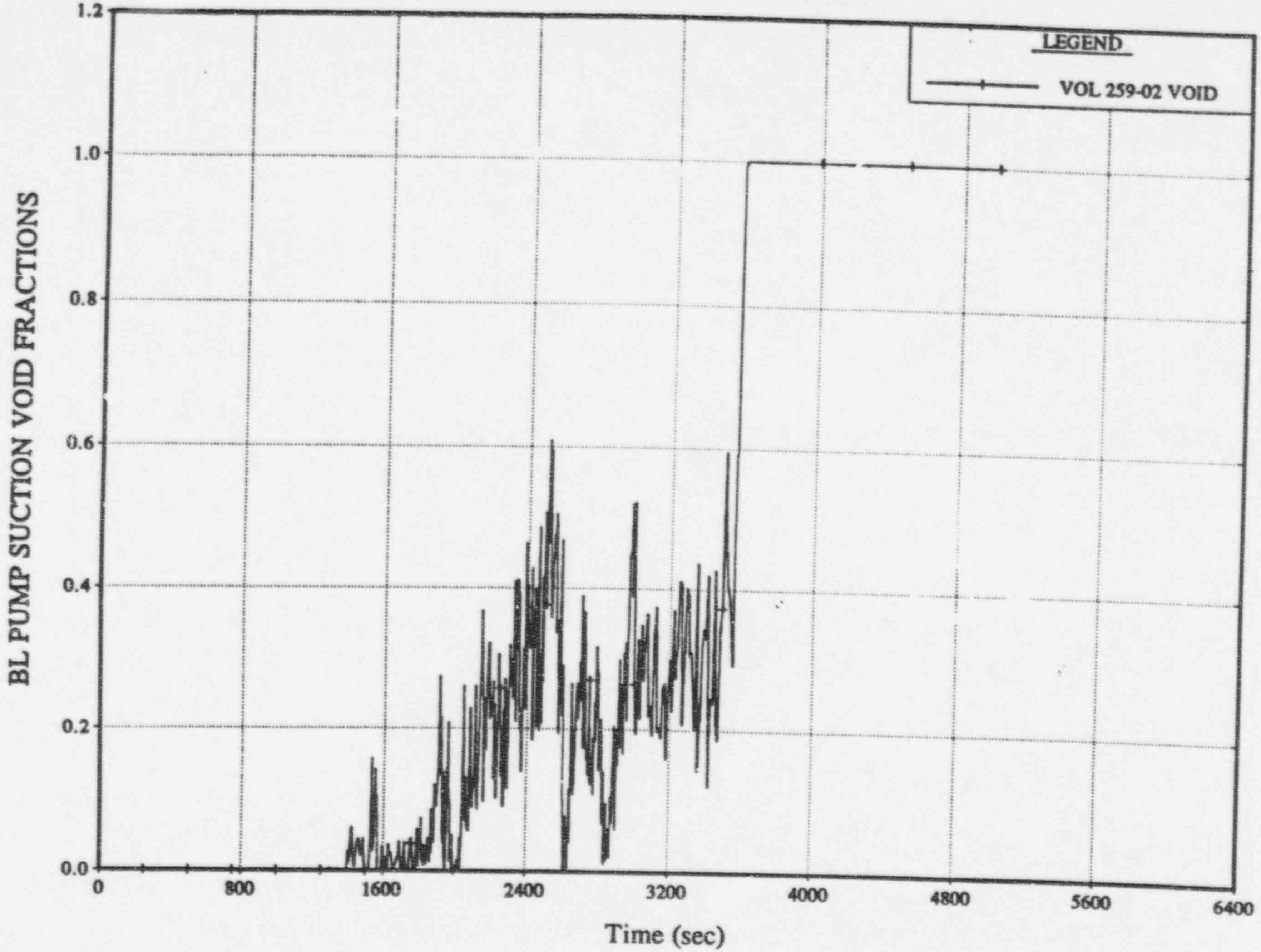

Fig 26

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R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

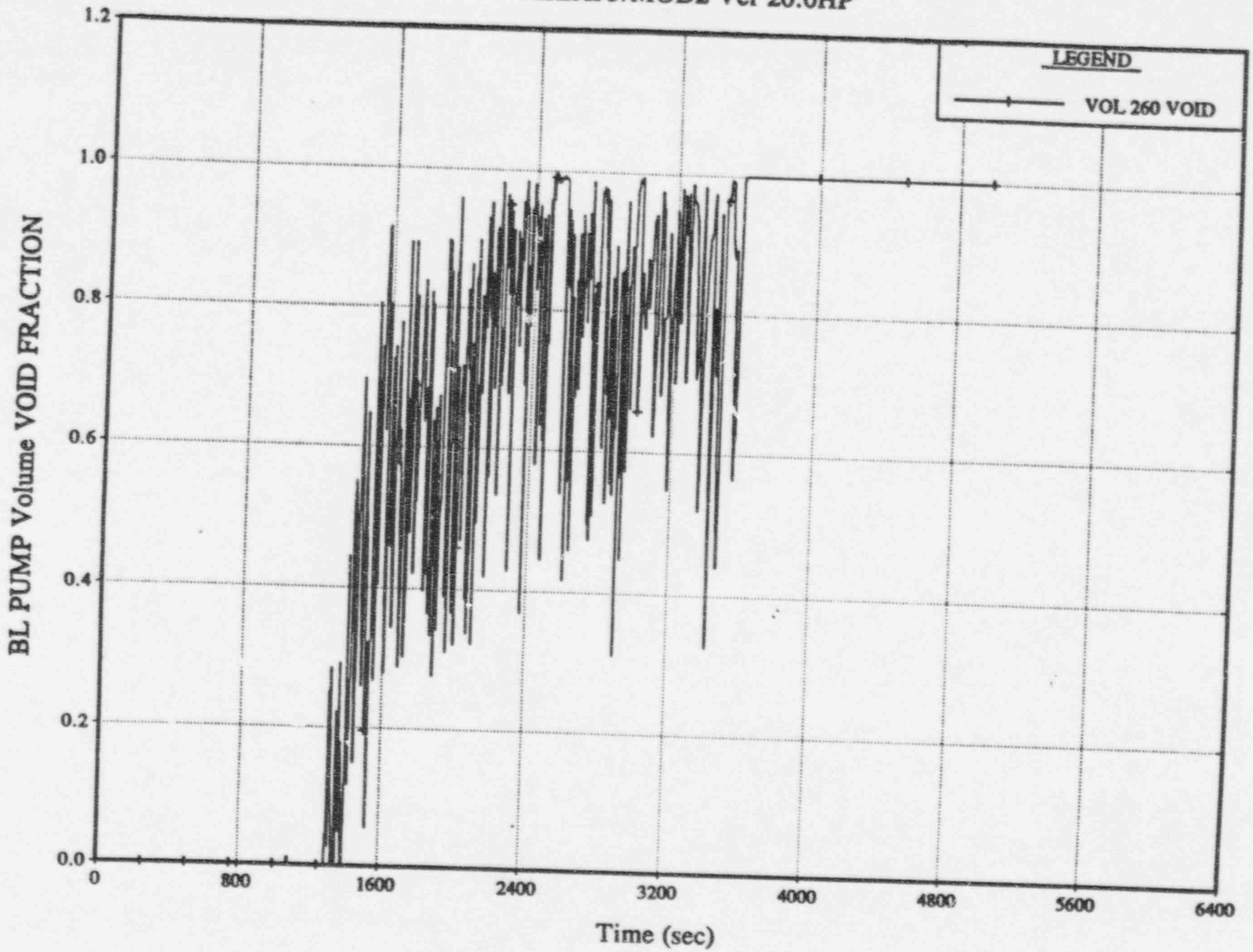

Fig 21

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP

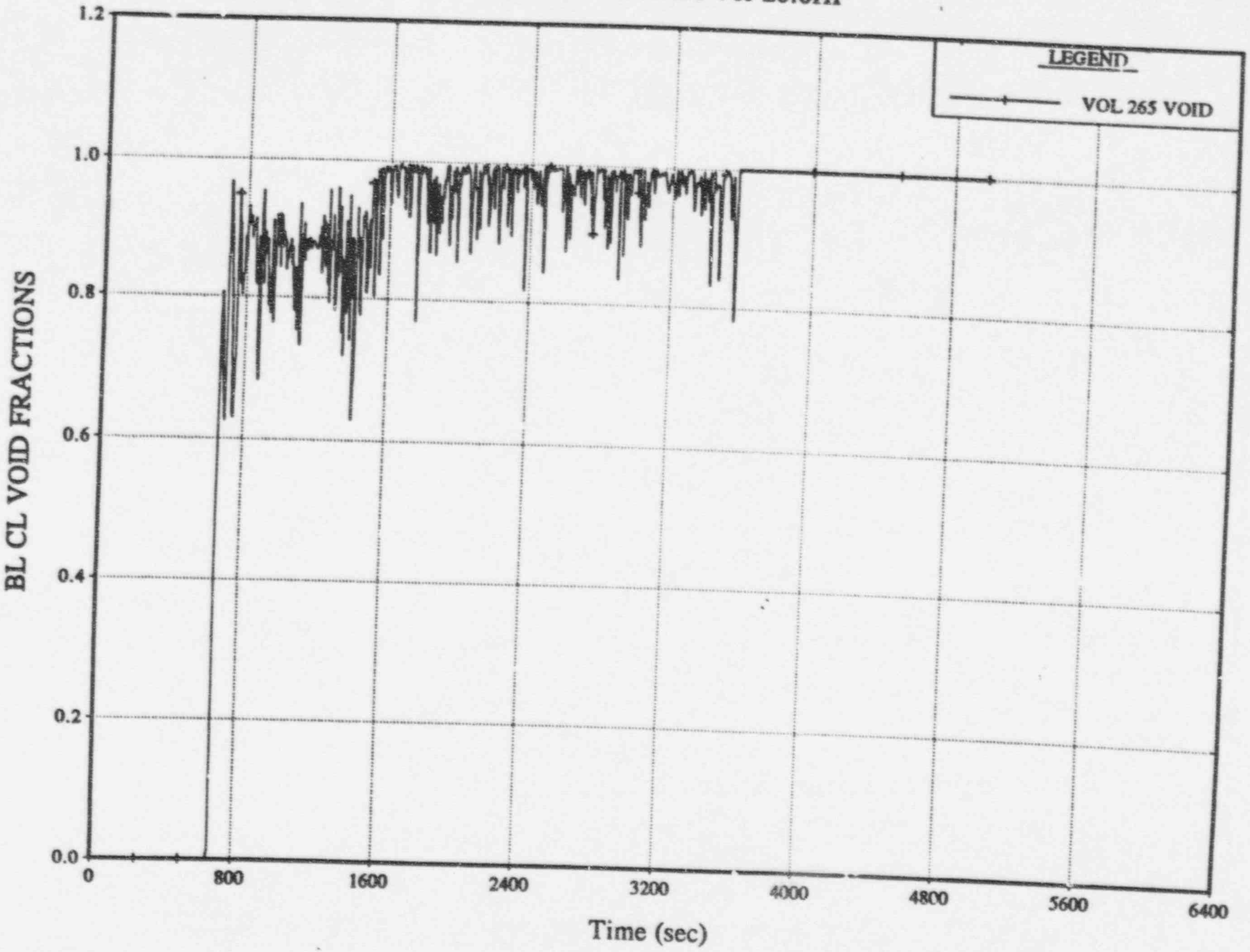

Fig 22

Fig 22

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Break Orientation: The break orientation, for SBLOCA studies, is placed at the bottom of the cold leg piping, between the ECCS injection location and the reactor vessel, since this configuration poses the greatest challenge to the ECCS in providing sufficient coolant flow to maintain core cooling. With the break so situated, ECCS entering the RCS through the injection nozzle in the broken cold leg must pass over the break prior to penetrating the reactor vessel. Unless the pump discharge piping is already full, the emergency coolant will be passed out of the break, unable to provide core cooling. This limits the effective ECCS flow, during critical cooling times, to that injected into the remaining loops (intact loops). For that reason, most plants have limits on the amount of injection that can be delivered to any one loop or leg during SBLOCA. A typical limit is that no more than 70 percent of the total ECCS flow can be delivered to any one injection nozzle.

The issues involved with the evolution of SBLOCA transients having alternate break orientations are primarily concerned with the longer term management of the accident than with the measurement of the capability of the ECCS system to provide sufficient and timely injection. The investigation of an SBLOCA scenario with the break at the top of the pump discharge piping is illustrative. For the first period of the transient--reactor trip, ECCS initiation, and loop draining through loop seal clearing--the LOCA is essentially the same irrespective of the break orientation, top, side, or bottom. The pump discharge piping is essentially full of water. Plant pressure is controlled by a balance between the volumetric discharge through the break, the vapor generation in the core, and condensation in the steam generator, if that is needed. Plant inventory is being lost rapidly and a liquid level imbalance is being setup between the downcomer and the core in order to achieve loop seal clearing. Loop seal clearing, when it occurs, is self advancing and rapid. At the end of loop seal clearing, one or more loops have been cleared of liquid; the liquid is retained in the core and downcomer. The downcomer core level imbalance is reduced to that necessary to drive steam to the break. This process, though dependent on break size, is independent of break orientation; it occurs in essentially the way same for bottom, top, and side breaks. Some arguments exist that side and top breaks offer less potential for liquid diversion to the break during loop seal clearing and, thus, arrive at a stable cleared configuration with higher vessel inventories than do bottom breaks. That effect, however, is difficult to demonstrate.

Following loop seal clearing, the ECCS system is challenged as to its ability to supply water at a rate sufficient to replace the water that is being boiled off in the core. In the critical cases, with a single failure of one of the high pressure injection systems (HPIs) and the break located at the bottom of the discharge piping, the ECCS cannot immediately keep pace with core boiling. The system is then in a boildown mode. The inventory in the reactor vessel continuously decreases until the decay heat drops or the ECCS flow increases (because of system depressurization) to the point of achieving a match with the core boiling. If the imbalance is sufficient, the core may uncover, exposing its upper regions to steam cooling before the match occurs. Modeling this phase of the transient with a bottom break is limiting because top or side breaks have effective ECC flows, that are up to 40 percent higher. Thus, for the initial system response and the determination of the adequacy of the ECCS, the bottom break is clearly the conservative choice.

After this initial period, some differences in the modes of accident recovery do occur. Following the acceptable match of decay heat and ECCS flow, the decay heating will continue to decrease at a slow rate; the system pressure may also continue to slowly decrease. This will create excess ECCS and the reactor vessel will start to refill. The rate is dependent on the particulars of the accident and can vary from a reasonable refill rate to an extremely slow one. Eventually the downcomer will be refilled with ECCS water backing up into the discharge piping. At this point, the behavior of the bottom, and side and top breaks starts to differ. For bottom breaks, the liquid backing up into the discharge piping will result in a fluid quality change at the break such that the break discharge is sufficient to remove excess injection. The downcomer remains full; the core, being hydrostatically balanced against the downcomer, is well covered and nothing of significance occurs for an extended period of time. For a side or top break, the break flow cannot respond to the rising system water level and the excess ECCS eventually spills over into the pump suction piping. Whether the loop seals reform or not and the consequences of that happening depend on many factors including operator action to manage the accident.

That the plant is safe and can be managed acceptably during recovery is, in FTI's view, a concern for the plant Emergency Operating Procedures (EOPs) or other devices that control the eventual recovery of the plant. The initial response of the ECCS, its adequate sizing, and the establishment of long-term cooling have, by this phase of the accident, been established. That is the purpose of 10CFR50.46. The eventual recovery from the accident, the evaluation of the multiplicity of operator actions, and their affect on the RCS and core are operational matters. Furthermore, these evaluations should be conducted with realistic boundary conditions such that expected and probable plant behavior is described; aberrant, supposedly conservative assumptions, should not be used. Still an investigation into the possibilities can be useful in determining if any role remains for LOCA analysis past the initial ECCS response.

There are four main factors that determine the continued course of an SBLOCA for side and top breaks. Actually, even a bottom break will eventually evolve to the same configuration as side and top breaks since the break flow cannot be adjusted infinitely, but their development requires an extremely long time period. For our purposes, it is sufficient to consider just the top or side break. The main factors are:

a. The amount of steam flow possible through the upper head spray nozzles (UHSNs).

   This vent path, if it supports the core steaming rate, can eliminate the need for steam venting through the loops. Because core steaming is dependant on decay heat, the UHSNs increase in significance as time progresses.

b. The amount of steam or water that can be passed through the reactor vessel fit up leakage.

   Hot side to cold side leakage is another vent path capable of eliminating or reducing the need for loop venting. This mechanism responds with time in two ways. First, decay heat decreases with time reducing the amount of steam to be vented and, secondly, the RCS nominal temperature also decreases with time, increasing the fitting gaps and improving vent capability. Care should be exercised in applying leakage credit during partial core

uncovery since the steam in the upper head will be superheated, tending to heat the metal structures and reduce the gaps.

c. Whether the mechanism for filling the suction lines evolves gradually or it is a spontaneous development.

   If the means for spilling water into the suction piping is the decrease in decay heat, the build up of excess injection will occur slowly and the accumulation of water in the suction piping will be gradual. The potential for blockage will be imposed gradually and at times beyond which loop venting may not be needed. If, however, the increase in spillage is rapid, as may occur because of the return to service of a failed injection system, the potential for blockage can occur with reasonable rapidity.

d. The amount of steam flowing through the loops that is not condensed in the steam generators.

   This of course is the most direct factor of concern in evaluating the effect of re-closure of the loop seals. An important consideration is the degree of management credited. If the steam generator pressure control is conducted as intended by the EOPs, the plant will evolve to a reflux mode with no need for loop venting except where spontaneous increases in injection flow occur (item c).

Depending on the plant, the UHSNs can eliminate any concern over a secondary loop seal clearing process. All Westinghouse plants, classified as T_{cold} upper head plants, have reasonably large UHSNs. McGuire/Catawba and Sequoyah are examples of such plants. An examination of the Sequoyah calculations for a 1.9-inch break shows that the process of loop seal clearing is interrupted at about 2,000 seconds by the development of a head imbalance between the downcomer and the core that is large enough to support sufficient steam flow through the UHSNs to eliminate the need for loop venting. For this break and breaks of smaller cross-sectional areas, the loops never clear and, after achieving a minimum suction piping downside level, the suction piping will gradually refill. Because the core swell factor (mixture level divided by the collapsed level) is approximately proportional to core steam generation and the differential pressure required for flow through the UHSNs is proportional to the square of the rate of steam generation, the elevation head difference between the core and the downcomer will decrease more rapidly than the swell height difference as decay heat drops. The core mixture level actually increases with time, assuring continued core cooling. Therefore, for breaks that do not require loop seal clearing during the initial system response, no need for clearing will develop later in the accident. Further, for larger breaks that do require loop seal clearing, the ability to flow sufficient steam through the UHSNs will develop with time, also eliminating the need for loop steam venting. Thus, for T_{cold} upper head plants, because the UHSNs have substantial capability for steam venting, no concern over the refilling of the loop seals with time exists.

For T_{hot} upper head plants, the UHSNs are not sufficient to vent a meaningful amount of steam. Such plants can be bounded by considering the results of excess ECCS for a theoretical plant, absent UHSNs and internals leakage. To this end, an evaluation has been conducted for a plant

without UHSNs or internals leakage and for which no operator actions have been taken to manage the accident. The analysis comprises an examination of the potential condition of the RCS following a 2-inch diameter break in the side or top of the cold leg just after loop seal clearing, 1½ hours into the accident, and at six hours into the accident. In each case, sufficient time has elapsed for the suction piping to have been refilled to the extent predicted. The plant is considered to be in a transient mode for the evaluation of the conditions post-loop seal clearing and in a quasi-steady-state for the evaluations at 1½ and six hours. The spectrum of conditions considered are one and four loops venting and one or two HPIs providing makeup. No injection is arbitrarily lost or spilled from the system. The timing of loop seal clearing was obtained from available spectrum calculations performed with the evaluation model. The timing may differ slightly for a top break with two HPIs, but that is not a significant simplification.

One key in understanding the analysis is to realize that a transport mechanism for the core energy must exist. Either the core is boiling and steam is being used to transport energy to the break or the RCS is basically water solid and experiencing natural circulation. A water solid configuration at six hours is possible, if the operator has followed the EOPs and depressurized the steam generators. However, there is no concern for loop seal blockage in a circulating system so that case will not be considered further. Because steam is the transport mechanism, the core is boiling and the flow rate of water to the core can be determined by balancing the heads between the suction riser section and the core given that the inlet enthalpy is specified. For this evaluation, the core inlet enthalpy was assumed to be the injection enthalpy and a level credit was taken for the difference in the downcomer liquid density and the core average liquid density. An analogous assumption, that the core inlet is saturated, can be made with no density difference applied between the core and the downcomer. Either approach achieves essentially the same core mixture level. One depresses the core collapsed level less, while the other generates a higher mixture swell. Steam generated in the core passes through one or four loops and is mixed with liquid in the pump suction riser section at the spill under. Here, excess ECCS subcooling condenses steam to the extent possible and any remaining non-condensed steam is bubbled up through the riser section to the break. For the post-loop seal clearing analysis, the pressure is taken from the reference RELAP5 calculation. For the extended time evaluations, the pressure is determined from the break model (Moody or Extended Henry-Fauske) and the consideration of mass and energy equilibrium for the RCS. For the single HPI cases, the break requires steam and water to be in equilibrium and only that steam flow (the break steam) was used to lighten (decreased density) the riser section. For the two HPI cases, the HPI sensible heat was sufficient to absorb all of the core heat and no break steam flow occurred. In these cases, the condensation process in the bottom of the riser section was assumed to take place in an exponential pattern over the bottom four feet of the riser section. Forty-eight percent of the steam was condensed in the first one-half foot, eighty percent was condensed by 1½ feet, and all the steam was condensed by four feet.

The table presents the results obtained for liquid collapsed levels in the riser sections of the venting suction piping and the reactor core. The table also indicates whether or not the core is covered by the boiling mixture. As can be seen from the table, the core is essentially covered with a boiling mixture for all cases. The one HPI, four-loop venting case has a core mixture height of 11.9 feet at six hours, which is considered essentially covered. Extending these results

to greater times will eventually demonstrate core uncovery. However, operator action in conjunction with the EOPs has been delayed for over 5 hours for these analyses. Because such action will mitigate the consequences of these transients, it is not necessary to consider the response of the system for longer times.

The evaluations provided are appropriate if the processes described and credited are not erratic. That may not be true for the condensation process in the riser sections. At that location, with steam being forced into subcooled water, water cannon or water hammer effects may be produced. In that event, the system can be expected to vary about the nominal conditions derived here. Core mixture levels will be both higher and lower than those indicated, but, because the core heating at these times is not rapid, the core overall should be well cooled. Again, if the operator follows the EOPs, the potential for these conditions will be removed early in the event.

In summary, FTI maintains that the decision to run 10CFR50.46 calculations for breaks at the bottom of the piping is appropriate. These breaks clearly offer the greatest challenge to the emergency core cooling systems. SBLOCA transients may evolve differently for top and side breaks than for bottom breaks, but the evolution is essentially independent of the ECCS. Further, the differences occur during the period of accident management that is the purview of the Emergency Operating Procedures and they should not be evaluated with the required EM conservatisms. Not withstanding these considerations, FTI has considered the evolution of top and side breaks. For T_{cold} upper head plants, the evolution of the transient has been shown to produce a smooth increase of core coolant level with sustained and continuous core coverage after a possible initial uncovery. For T_{hot} upper head plants, inter-vessel leakage around the hot leg nozzles serves the same purpose as UHSNs for the T_{cold} plants, making long-term cooling a smooth process with no core uncovery.

Additionally, top breaks were evaluated out to 6 hours for a plant without UHSNs or inter-vessel leakage. It was shown that, at least on the average, the core will be continuously covered. It was demonstrated that the transient can progress past six hours without experiencing serious core uncovery, requiring many additional hours to produce significant core uncovery. Because the potential to require loop venting in the long term is limited (UHSNs and inter-vessel leakage effects) and because the EOPs typically recommend operations to depressurize the plant early in the transient, thereby refilling the plant and mitigating any need for loop venting, FTI believes that any consideration of times beyond those presented to be the proper subject of operational procedures and not suited for consideration under 10CFR50.46.

Analysis Results for a 2-inch Diameter Pump Discharge Break at the Top of the Pipe

Time hours	Decay Heat %	HPIs Operating	Loops Venting	Pump Suction Riser Collapsed Level feet	Core Collapsed Level feet	Core Mixture Level feet
0.5	2.0	1	1	2.4	14.6	12+
			4	5.5	11.5	12+
		2	1	3.7	13.3	12+
			4	6.2	10.8	12+
1.5	1.5	1	1	4.4	12.6	12+
			4	6.6	10.5	12+
		2	1	7.4	11	12+
			4	8.2	10.2	12+
6.0	1.0	1	1	5.9	11.1	12+
			4	7.3	9.7	11.9
		2	1	7.6	10.4	12+
			4	8.2	9.8	12+

Cross Flow Resistance and Core Modeling: In our 3/28/96 telecon, questions were raised as to the basis for the crossflow modeling used within the core. The modeling is outlined in Section 4.3.2.5 of volume II of the RSG evaluation mode' report, BAW-10168, Revision 2. Basically, the model is a 20 axial region core, radially divided into a single assembly hot channel and the remainder of the core. Each volume in the core model is connected vertically and horizontally. Vertical resistance is based on core design factors which in turn are based on flow tests for the fuel assemblies. Correlations for the prediction of lateral resistances vary substantially. A k-factor value of 2, based on the interface area between adjacent fuel assemblies, has been selected for the evaluation model. This value produces reasonable results that agree with experimental expectations for SBLOCA. The value, however, does not appear to be unique and either smaller or larger values would also appear to produce valid results. The B&W-designed plant RELAP5 small break evaluation model uses a value of 200 for the base crossflow resistance and does not produce substantially differing predictions. (There are indications, however, that the higher resistance used in the B&W-designed plant SBLOCA model may have a stableizing influence on the calculation.)

Two adjustments are imposed on the basic resistance in order to assure conservative SBLOCA predictions. For the top half of the core, the flow resistance from the average channel to the hot channel is increased by a factor of 10 (flow resistance from the hot channel to the average channel is not increased). This has little effect on the behavior of the core mixture or the core flows below the mixture level. However, above the mixture in the steam cooling region, provided the core has uncovered, the increased resistance limits any tendency to flow steam from the average to the hot channel. It is expected that steam will flow from the hot channel to the average because of the higher vapor generation in the hot channel. Because flow diversion out of the hot channel is a conservatism, that flow is not impeded. However, flow reversion back to the hot channel would have the effect of reducing the hot channel vapor temperature and increasing cooling. Although some flow reversion is expected, the resistance within the model is increased so as to limit the effect. The factor is only applied to the upper half of the core because, on a practical basis, it is not possible to predict acceptable cladding temperatures if the top half of the core is uncovered for an extended period. This modeling adjustment, then, is taken to help assure a conservative evaluation.

For reasons similar to the increased crossflow resistance, the hot channel outlet reverse flow resistance was increased to a k-factor of 200 based on the assembly flow area. It was envisioned that this would reduce the tendency for liquid fall back into the hot channel by encouraging liquid to flow into the average channel and then crossflow to the hot channel. The effectiveness of the high reverse flow resistance, however, is mitigated by the need for the hydraulic solution to achieve a pressure balance between the inlet and outlet plenums. As the flows and void fractions develop axially within the core, the hot channel maintains a slightly increased voiding because of its higher vapor generation rates. This leads to an apparent pressure imbalance between the two columns (hot and average channels) as the core exit is approached. To adjust for this imbalance, the solution allows negative liquid flow into the uppermost volume of the hot channel creating a lower void fraction for that volume. The reduced voiding in the upper volume balances the channel pressures. Note should be taken that the upper two volumes of the hot or average

channels do not represent nuclearly heated regions of the plant. These volumes model the upper unpowered segments of the fuel pins (the fuel pin upper plenum and interior springs) and the upper nozzle of the fuel assembly. Thus, the flow and the void reduction do not occur within the core active region. The resultant negative flow from the upper plenum to the hot channel exit volume only occurs when the upper plenum contains some mixture. Model prediction problems are not created because once the inner vessel mixture level falls into the core region the pressure balance is maintained by a slightly increased mixture level in the hot channel. This higher mixture level in the hot channel is physically real and well modeled. Observations of the core mixture level predictions for the hot and average channel discussed below demonstrate the credibility of the solution. The increased resistance has been maintained in the model as a hedge against possible core reverse flow. The resistance does not work as a flow diversion under stagnate conditions but is likely to divert flow away from the hot channel under flow conditions. This would be a meaningful conservatism if SBLOCA were to involve any substantial period of reverse core flow. Although no such period can be identified, the only reverse core flow phases are those occurring during the loop stagnate phases of loop seal clearing and core boildown, the increased resistance factor has been kept as a precaution.

That the hot channel and average channel mixture heights evolve reasonably during a core uncovery can be observed in the attached figures. These figures display the axial void distributions of the hot and average channels as they developed for a 3-inch pump discharge break in a Westinghouse-designed 4-loop plant over the loop seal clearing period. The figures display void fraction versus axial core elevation from the lower plenum to the upper plenum at the elevation of the outlet nozzles. Each void fraction is displayed axially at the center of the volume from which it is taken and is connected to the void fraction of the adjacent nodes by a straight line. If not recognized, this technique can introduce some confusion, as occurs between the lower plenum and the core. The lower plenum is or is nearly liquid solid throughout the time period of these graphs, but the linear connection to the first core volume, which is legitimately voided, produces a visual impression that the lower plenum contains steam as the bottom of the core is approached. In truth there is a step change in void content between the lower plenum and the core. The same recognition should be made in reviewing the upper plenum void fractions. This, in part, is the reason that the channels, except for the lower plenum to the core, are displayed with connecting lines while the upper plenum volumes are displayed as points. The time at which the figure is captured is displayed just above the figure border. Within the upper plenum, the upper most value is at the elevation of the center of the core outlet nozzles. This volume spans the height of the outlet nozzles. The next lower volume is entirely below the span of the hot leg piping.

Loop seal clearing for the case shown in the figures occurs at approximately 715 seconds. The graphs display the core elevation head/mixture height as the necessary head to clear the loop seal develops on the approach to loop seal clearing and as the core refills after clearing. Graphs are provided at 640, 660, 680, 690, 700, 710, 715, 720, and 800 seconds. By 640 seconds, the clearing process has initiated and the core mixture level has fallen below the nozzle belt as indicated by the void fraction in the upper most volumes. (The upper volume represents the portion of the upper plenum adjacent to the outlet nozzles.) The core is still covered with mixture and the depressed void fraction at the exit to the hot channel can be observed. It can also be

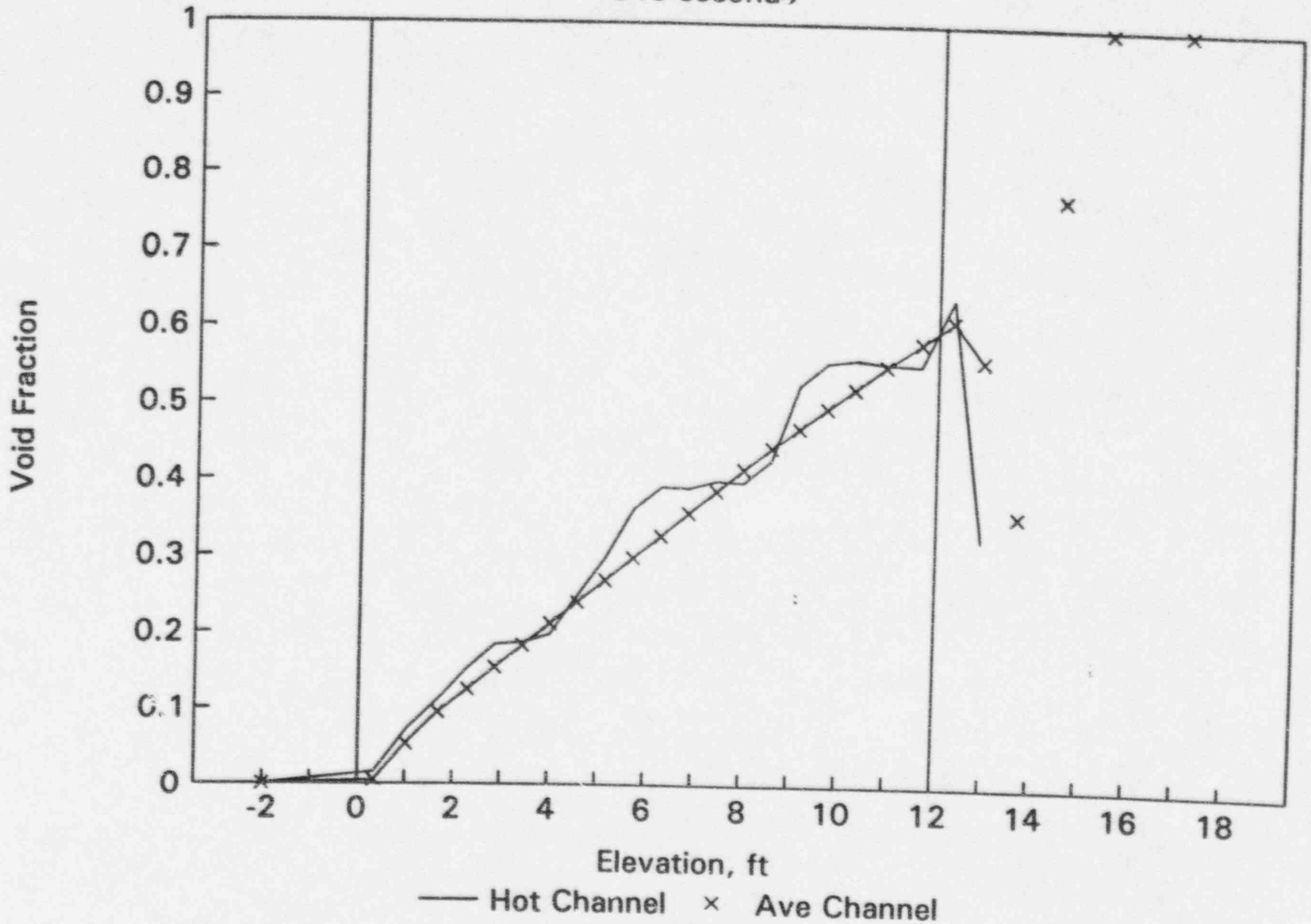
observed that the correspondence in void content between the average channel and the hot channel is quite good. Deviations occur, but the general trend is a slightly higher void content in the hot channel. There is no indication that the lower void content of the hot channel exit volume has propagated downward. By 660 seconds, more of the upper plenum is voided, but the core is still covered and the core void distributions remain reasonable. At 680 seconds, the columns representing the hot and average channels are starting to void. The upper plenum is essentially 100 percent voided. The core heated regions are still covered since the high voiding has not penetrated below the non-heated regions of the fuel assemblies. By this time, before any core heatup, the void fraction for the hot assembly upper region has evolved into agreement with that of the average channel. At 690 seconds, the heated regions of the hot and average channels have started to uncover. Loop seal clearing is now about 25 seconds away. Because the core outlet void fraction is at 90 percent, the cladding temperatures remain near saturation.

At 700 seconds, the two upper volumes of the heated core are showing substantial voiding and the very top heated node may be experiencing some heatup. For the limited uncovery apparent here, mist entrainment from the mixture may be sufficient to prevent core heatup. The hot and average mixture levels are in agreement as the uncovery proceeds. At 710 seconds, the mixture has fallen to its lowest level during loop seal clearing. The hot and average channel mixture levels remain in agreement with the hot channel slightly more voided. At 715 seconds, the loop seal for the broken loop has cleared and the downcomer and core levels are starting to equilibrate creating a core refill. By 720 seconds, the refill has progressed into the upper plenum. The void fraction at the very outlet of the hot channel is again depressed but that was not observed in the partial refill at 715 seconds. Thus, the predictions of the hot channel exit void fraction are consistent with the needs of the transient prediction, attaining the required degree of accuracy under conditions when core uncovery is occurring or eminent. By 800 seconds, the refill is complete and the core boil down phase has been entered. As shown, the refill did not completely fill the vessel. The region just below the out nozzle remains at an elevated void content and the upper plenum at the outlet nozzle elevation is completely voided.

In conclusion, core modeling has been arranged to provide for hot and average channel effects. Specific provisions have been incorporated into the EM to achieve conservative predictions of cladding temperature (crossflow resistance for the upper half of the core). The modeling works well during core uncovery as evidenced by the agreement between the hot and average channel mixture levels. Although a modeling factor does lead to an apparently inconsistent void fraction in an upper unheated volume of the hot channel during those phases of the SBLOCA transient when the upper plenum contains mixture, this difficulty is resolved as the core uncovers and is not present at any time that the calculation is predicting core uncovery or calculating cladding temperature excursions. Therefore, the core modeling approach employed is appropriate for the calculation of small break LOCA simulations.

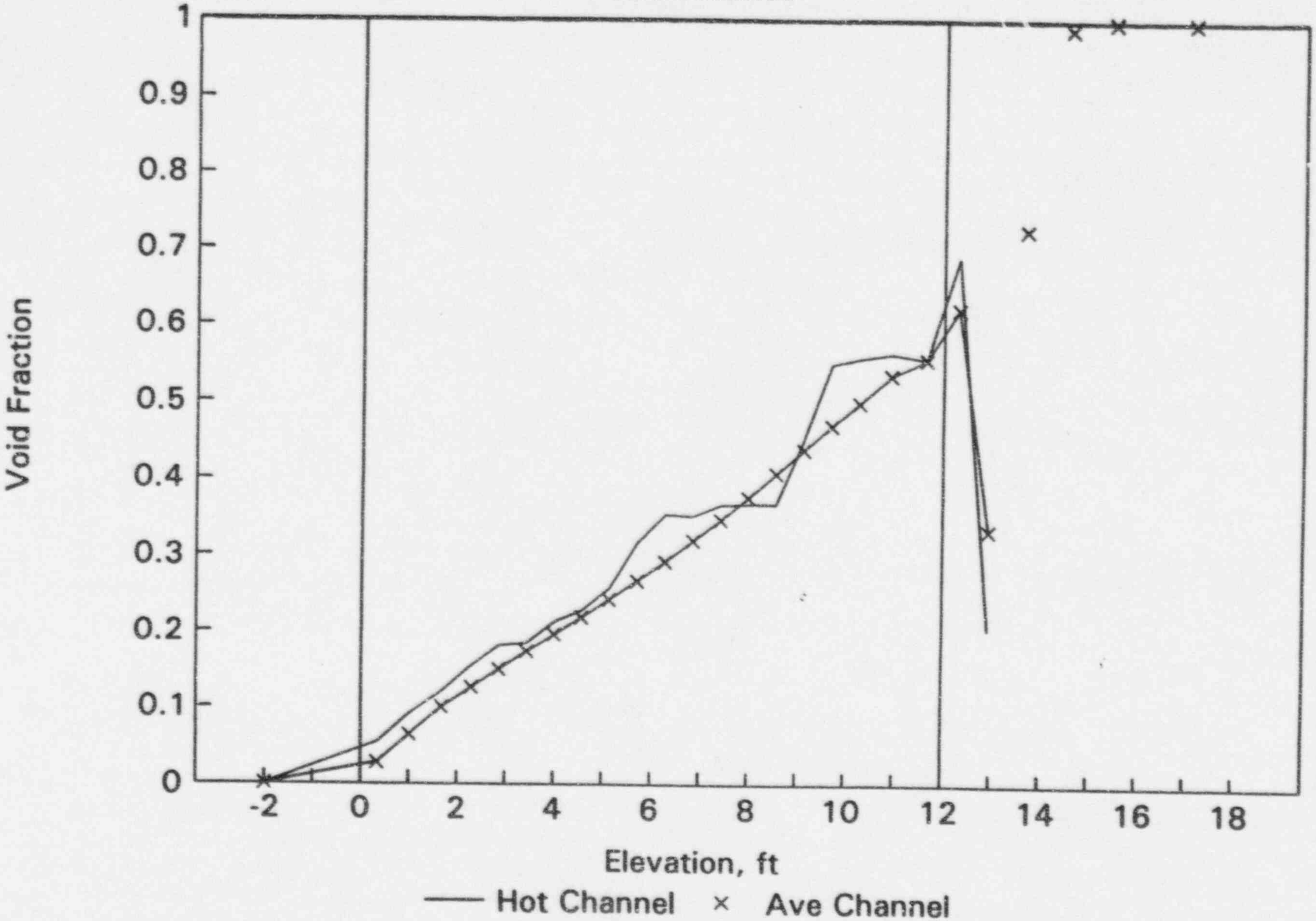
CORE VOID DISTRIBUTION - 3 in Break

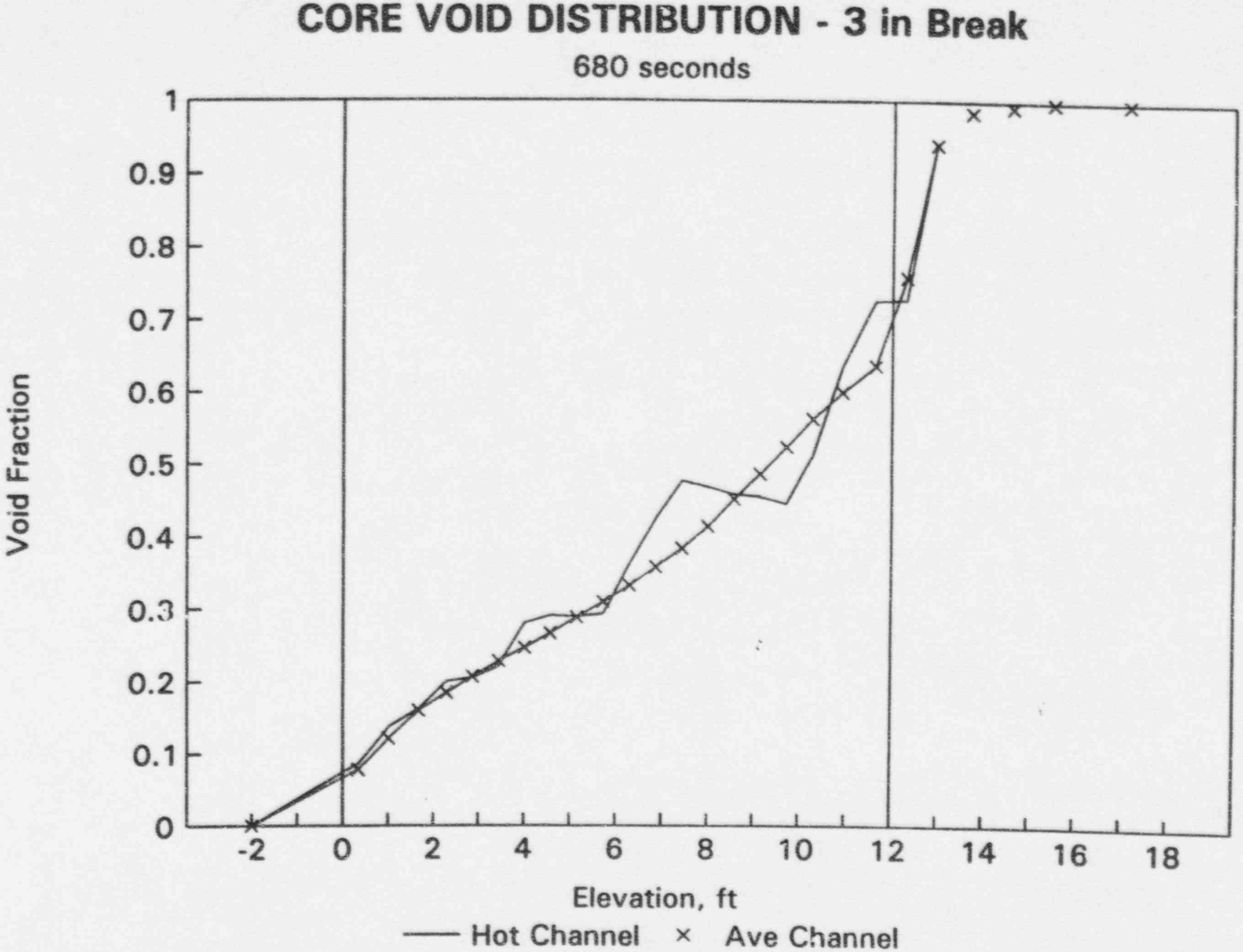
640 seconds



CORE VOID DISTRIBUTION - 3 in Break

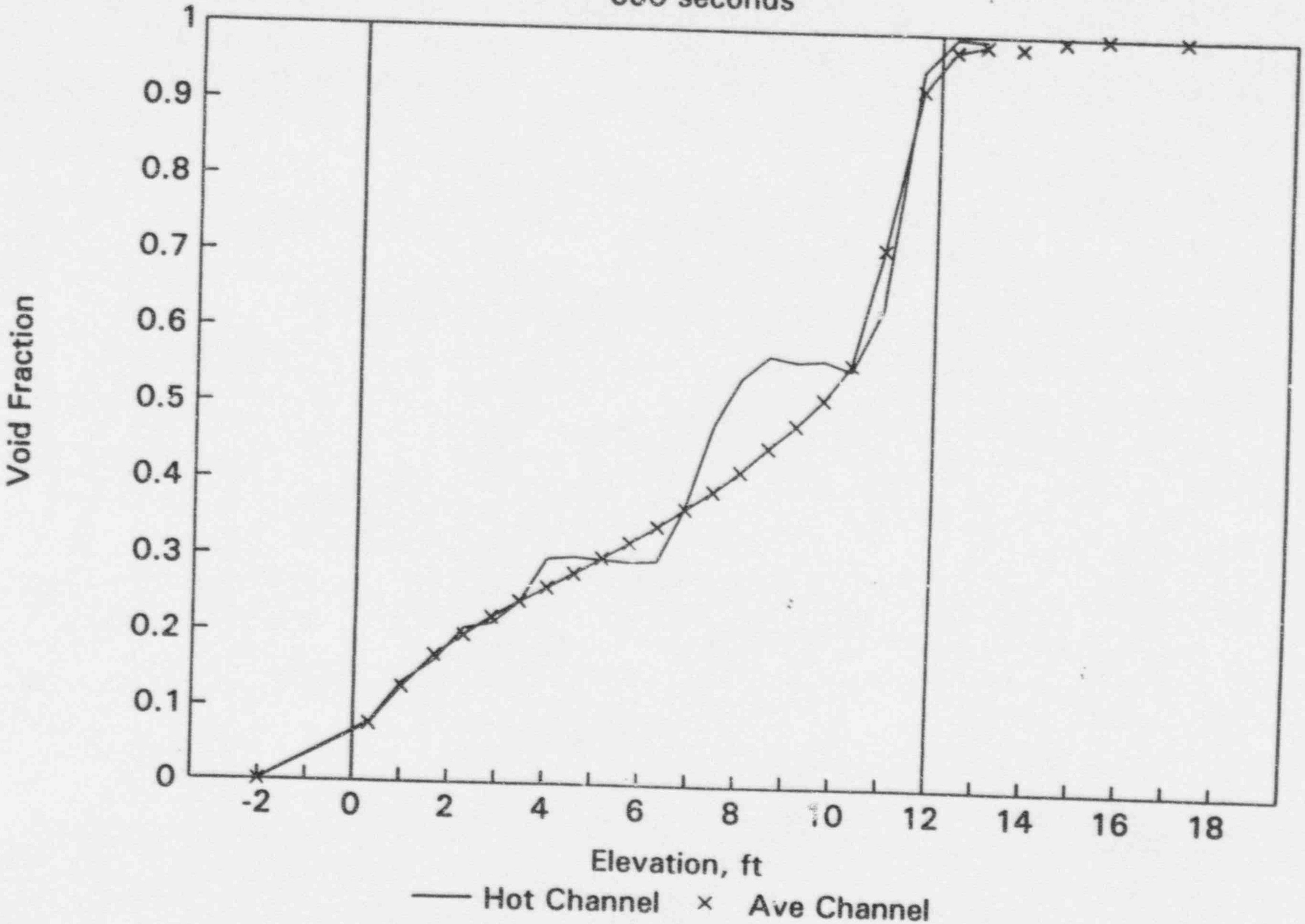
660 seconds




CORE VOID DISTRIBUTION - 3 in Break
680 seconds
Void Fraction
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
-2
0
2
4
6
8
10
12
14
16
18
Elevation, ft
Hot Channel
Ave Channel

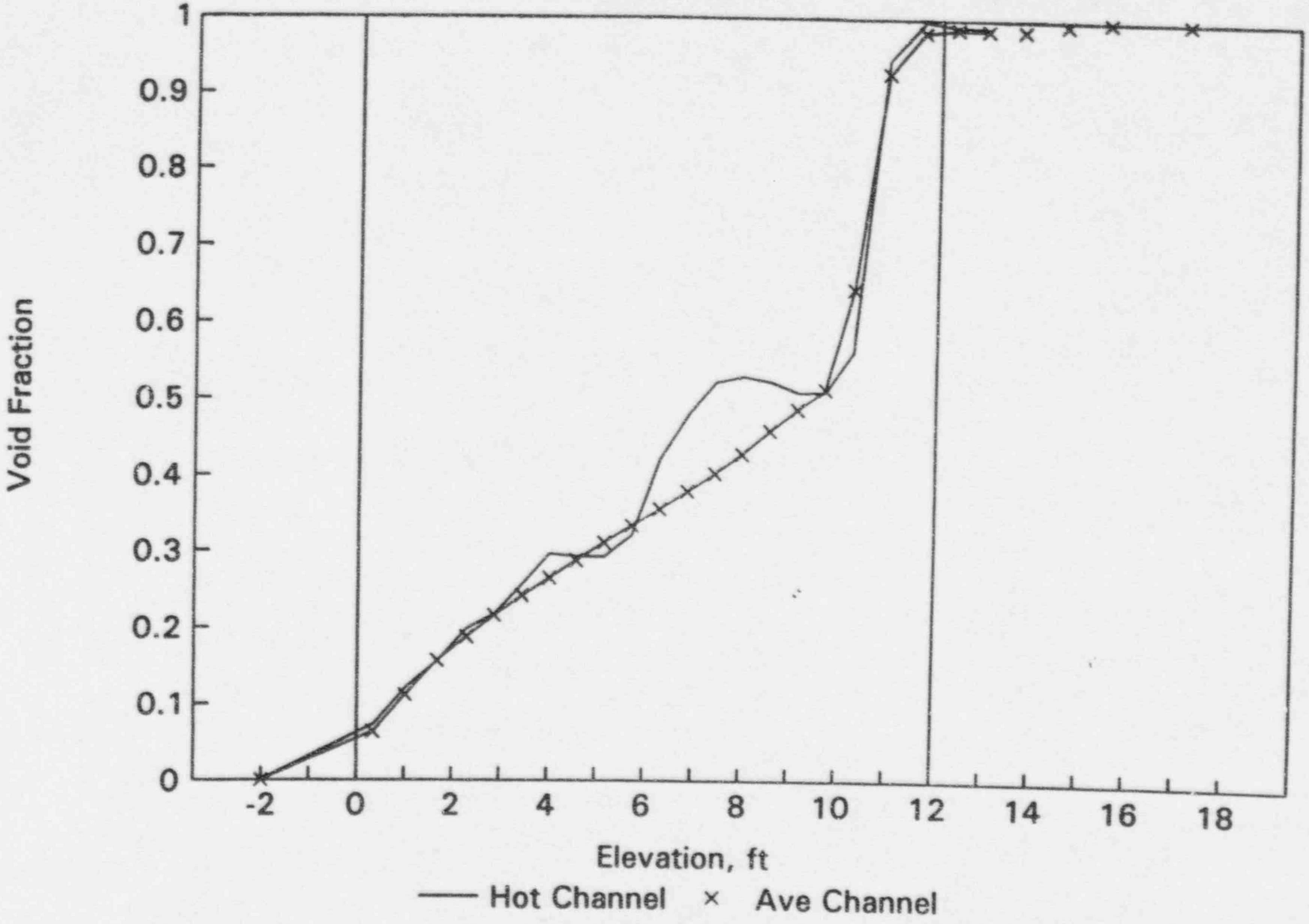
CORE VOID DISTRIBUTION - 3 in Break

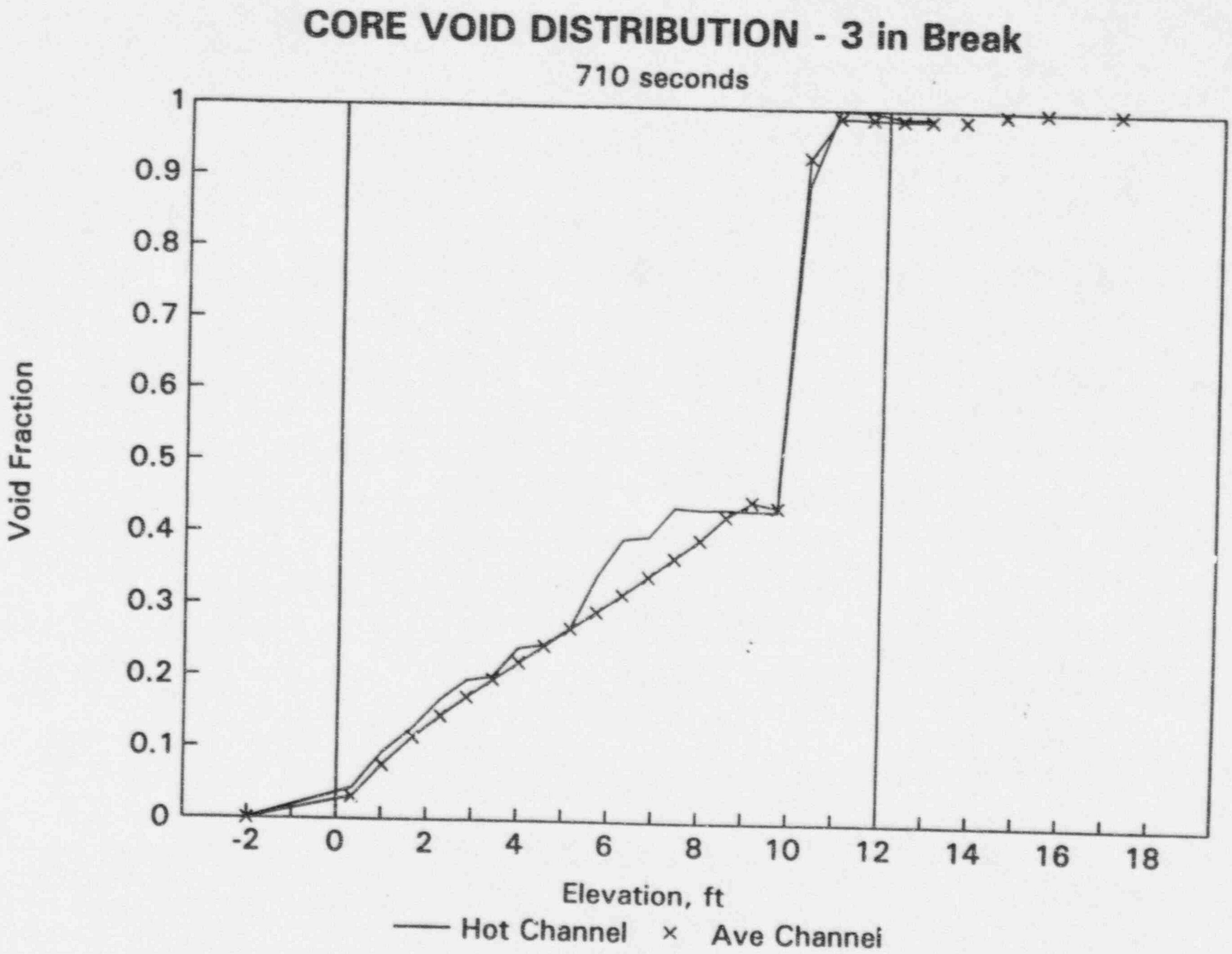
690 seconds



CORE VOID DISTRIBUTION - 3 in Break

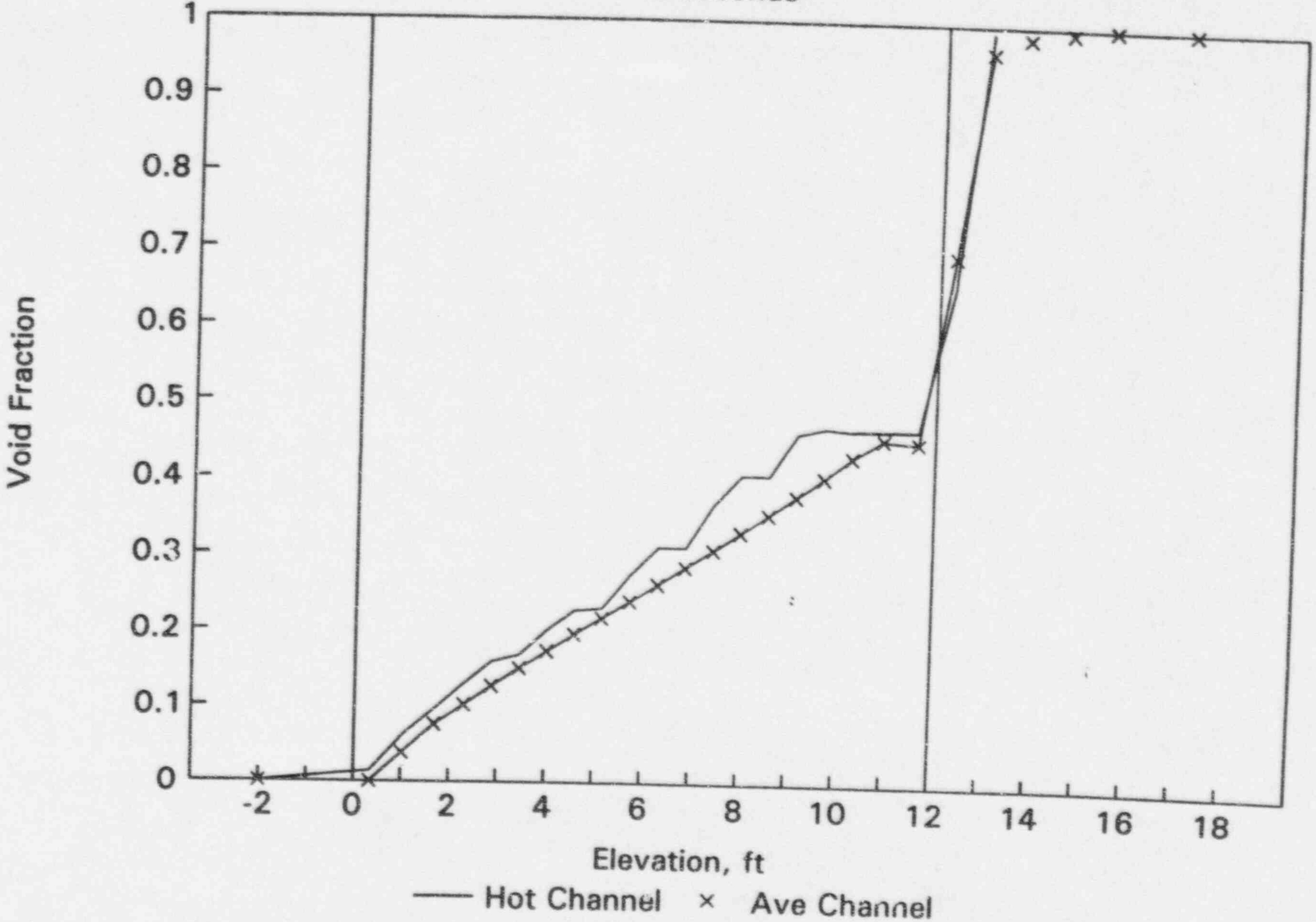
700 seconds




CORE VOID DISTRIBUTION - 3 in Break
710 seconds
Void Fraction
Elevation, ft
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
-2
0
2
4
6
8
10
12
14
16
18
Hot Channel
Ave Channel

CORE VOID DISTRIBUTION - 3 in Break

715 seconds



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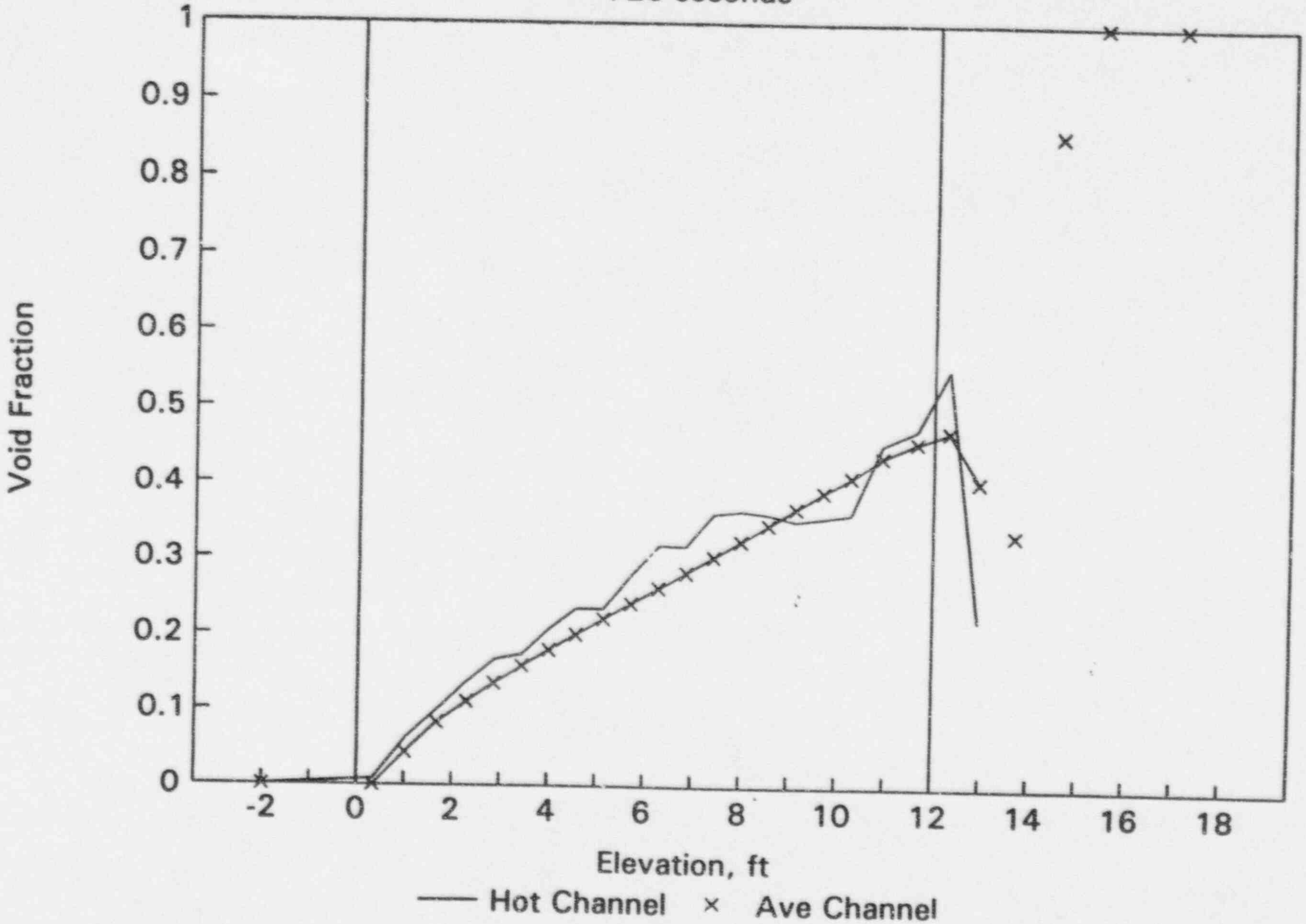
CORE VOID DISTRIBUTION - 3 in Break

720 seconds



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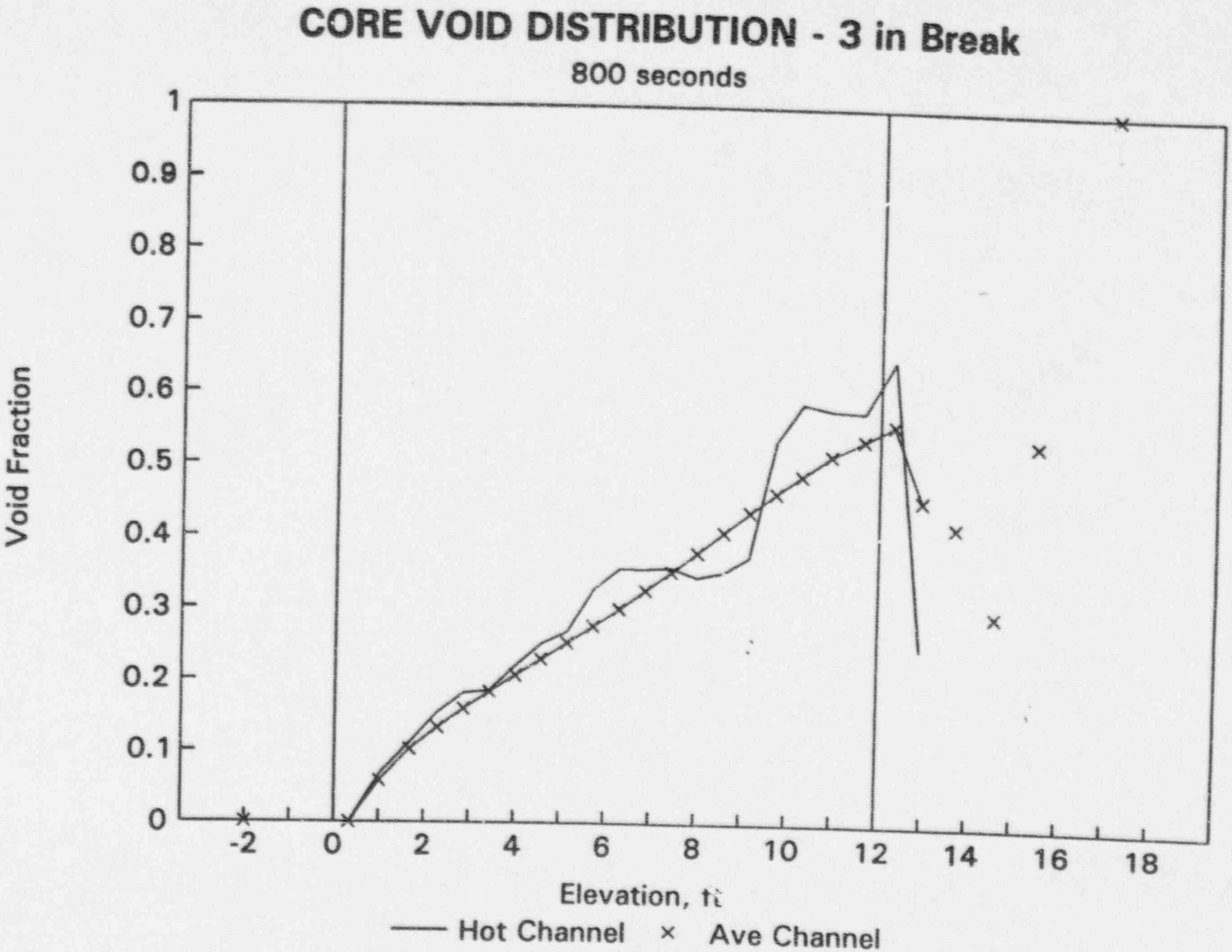


CORE VOID DISTRIBUTION - 3 in Break
800 seconds
Void Fraction
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
-2
0
2
4
6
8
10
12
14
16
18
Elevation, ft
Hot Channel
Ave Channel

44

Supplementary Break Orientation Information:

Range of Upper Head Spray Nozzle Areas:

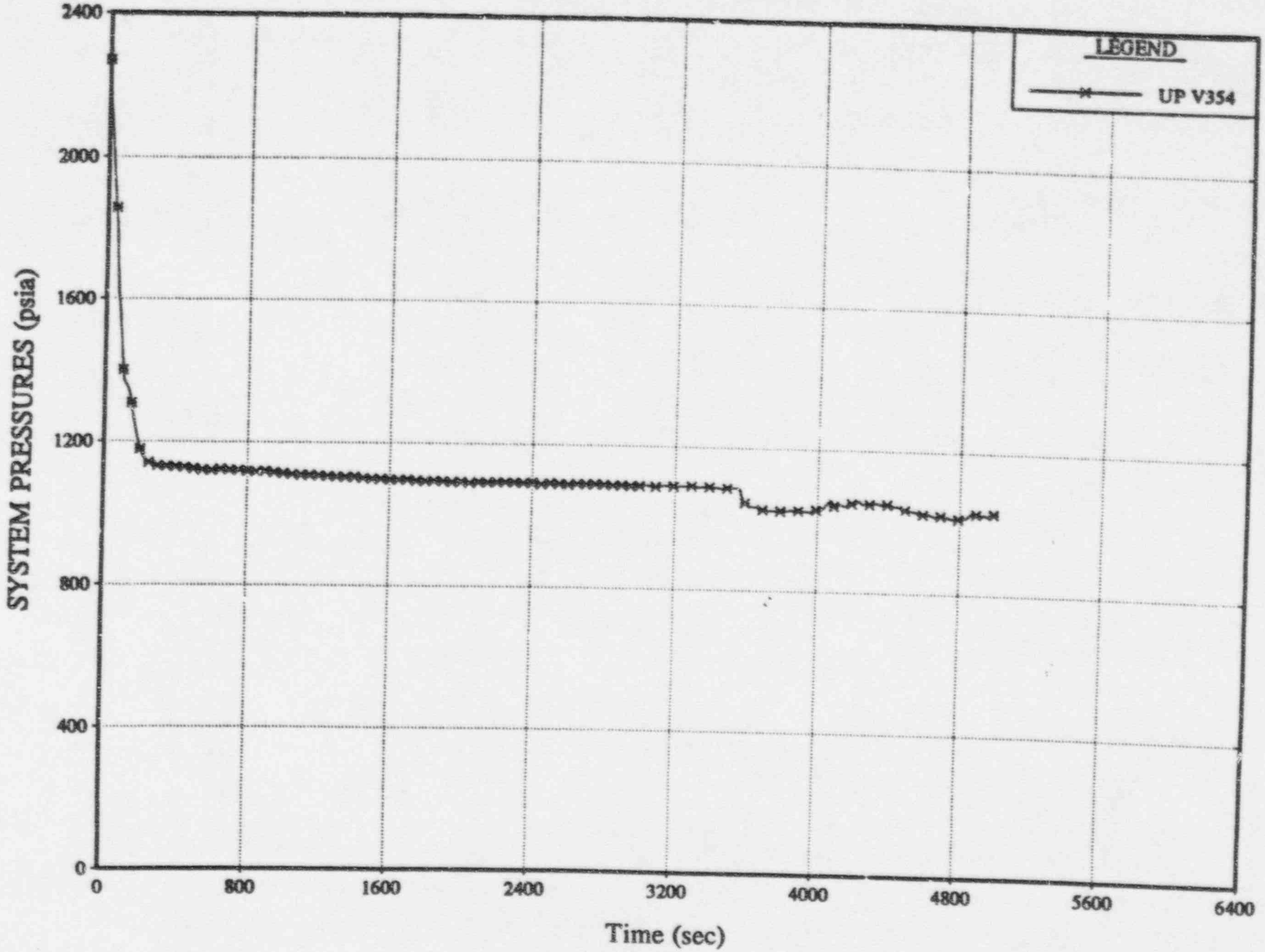
T-hot Plant => ≈ 0.02 ft^2 (Trojan, North Anna, Surry, etc.)
T-cold Plant => ≈ 0.45 ft^2 (McGuire/Catawba, Sequoia, etc.)

Some plants sit in between these limits with areas of 0.2 or 0.3 ft^2.

The inclosed plots are for the 2.1 inch case that was provided in an earlier communication. I felt that with them being part of a larger set they would be more useful. If the specific 2 inch case is important we can reconstruct it and send the same plots. Some of the definitions on the plots are:

UP	Upper Plenum
V	Volume or Node
AC	Average Channel
HC	Hot Channel
CVAR	Control Variable For the case of AC CVAR and HC CVAR the display is a collapsed water level for the core region with 0.0 taken at the bottom of the active region. The reason that the values exceed 12 feet is the inclusion to the two unheated volumes of the fuel assemblies that model the fuel pins above the uranium pellets and the upper nozzle of the fuel assembly.
Jun	Junction or Flow Path
J	Junction or Flow Path
UH SPRAY	Upper Head Spray Nozzle
IL ECC	Intact loops ECCS flow
BL ECC	Broken Loop ECCS flow For this case IL ECC CVAR and BL ECC CVAR are simply the high pressure injections. Had the plant depressurized these control variables would have picked up the accumulators and the low head systems.
HOT CH	Hot channel, HOT CH, CVAR is a control variable that approximates the mixture level in the core hot channel. For the purpose of this CVAR mixture is defined as $\alpha < 0.9$. The control variable samples the α from the bottom to the top in each node of the channel. If α is less than 0.9 the height of the volume is considered mixture once α is greater than 0.9 the control volume is considered as above the mixture and the search stopped.

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP



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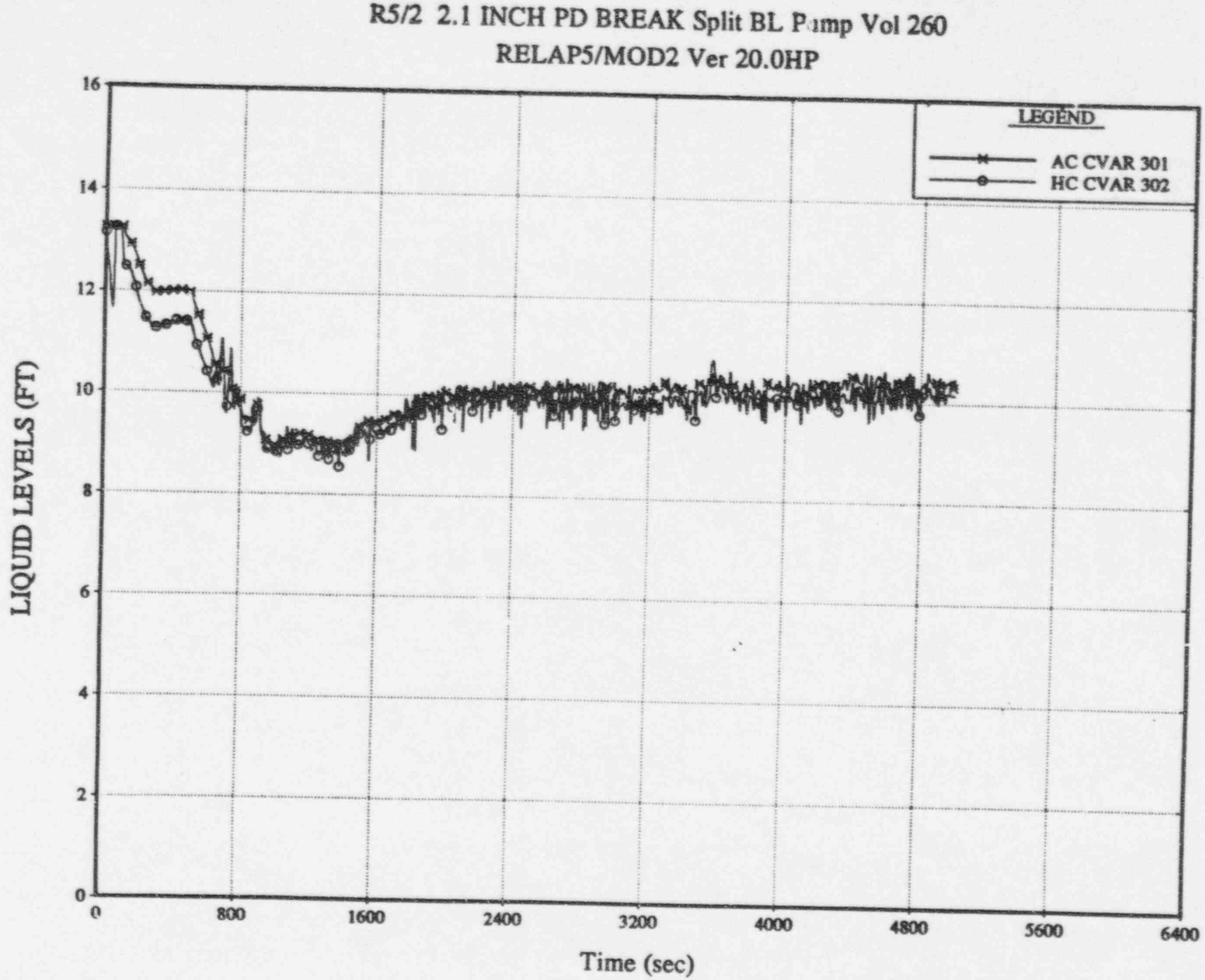
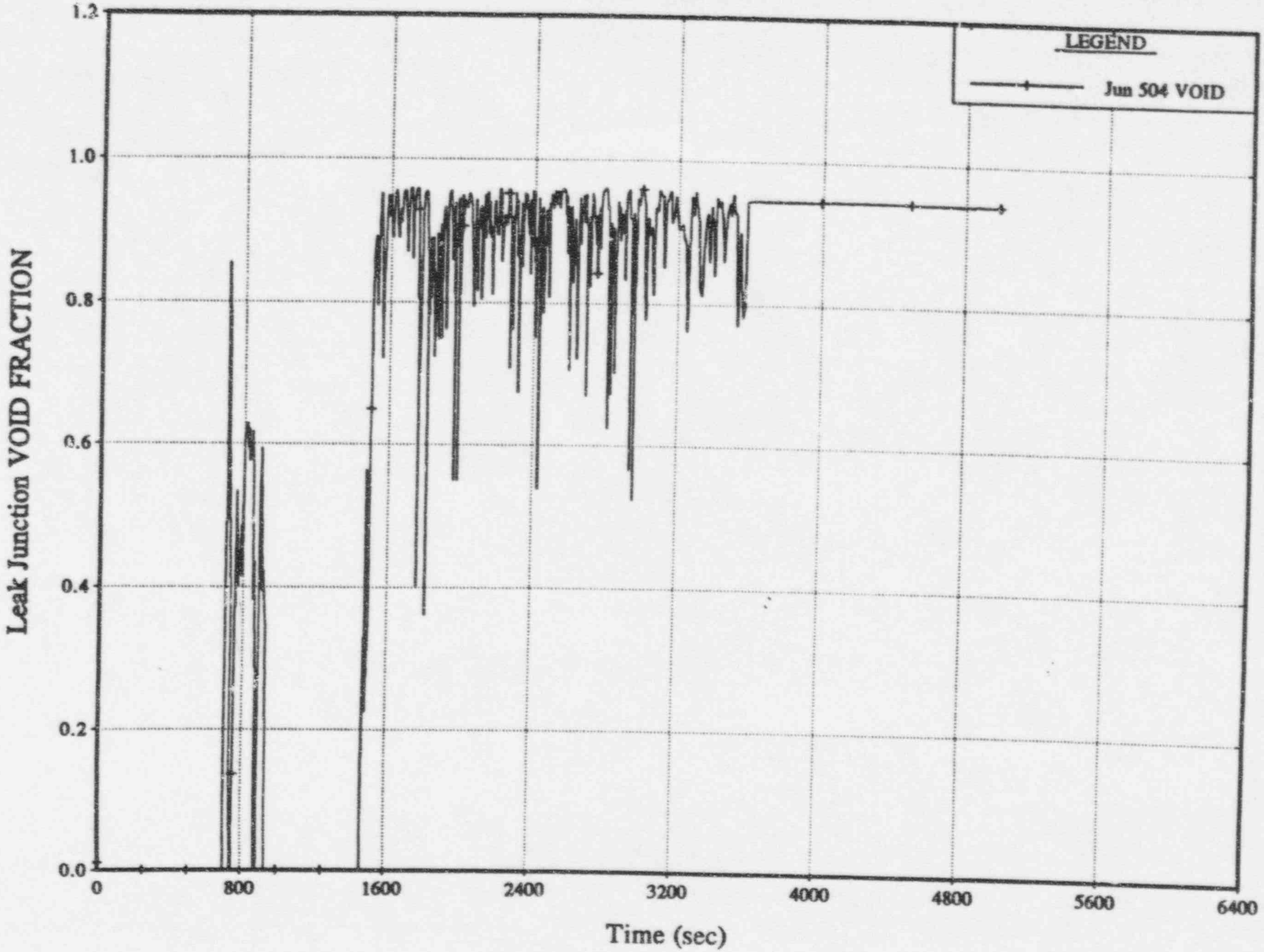
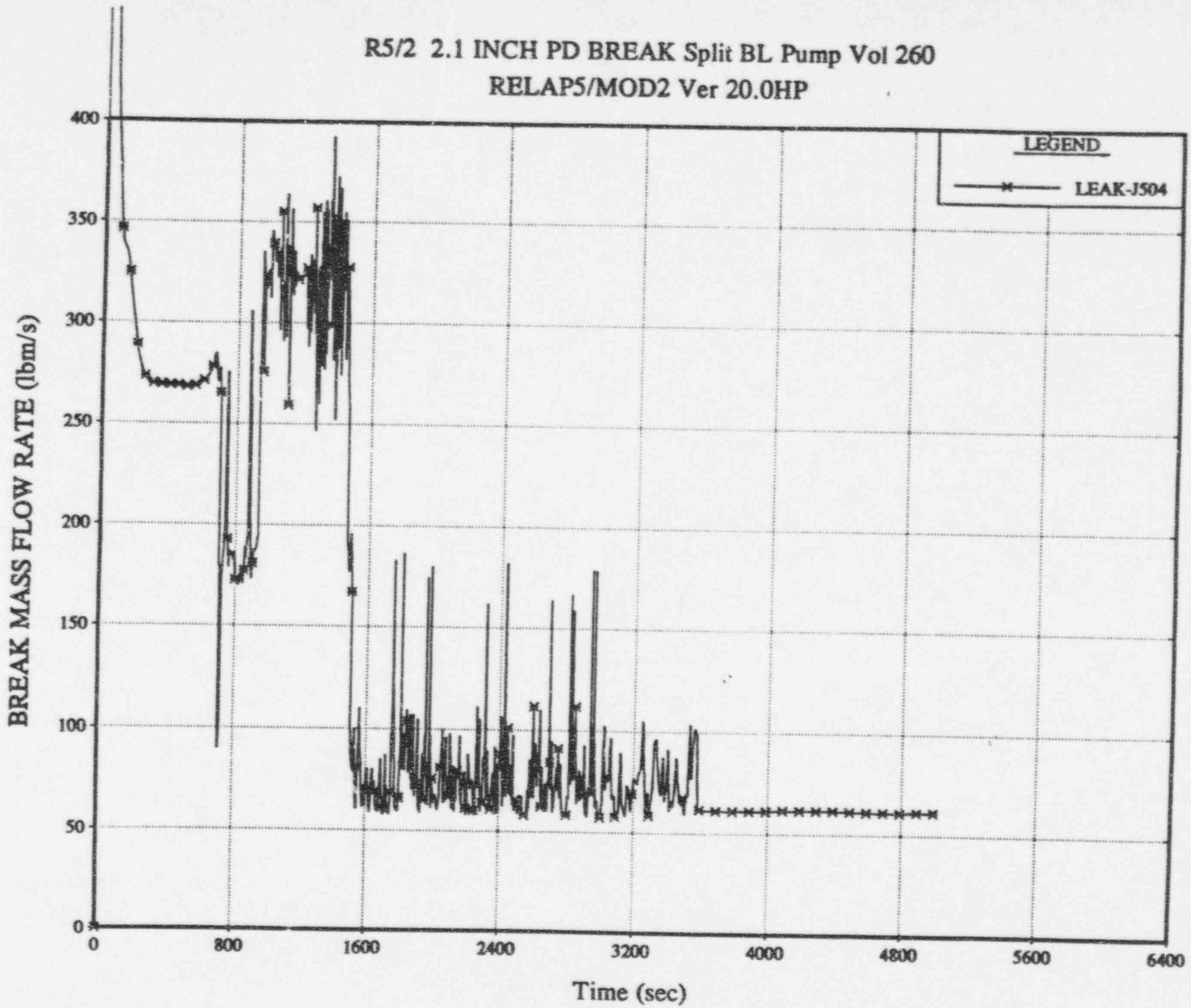

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP
LEGEND
AC CVAR 301
HC CVAR 302
LIQUID LEVELS (FT)
Time (sec)
0
2
4
6
8
10
12
14
16
800
1600
2400
3200
4000
4800
5600
6400

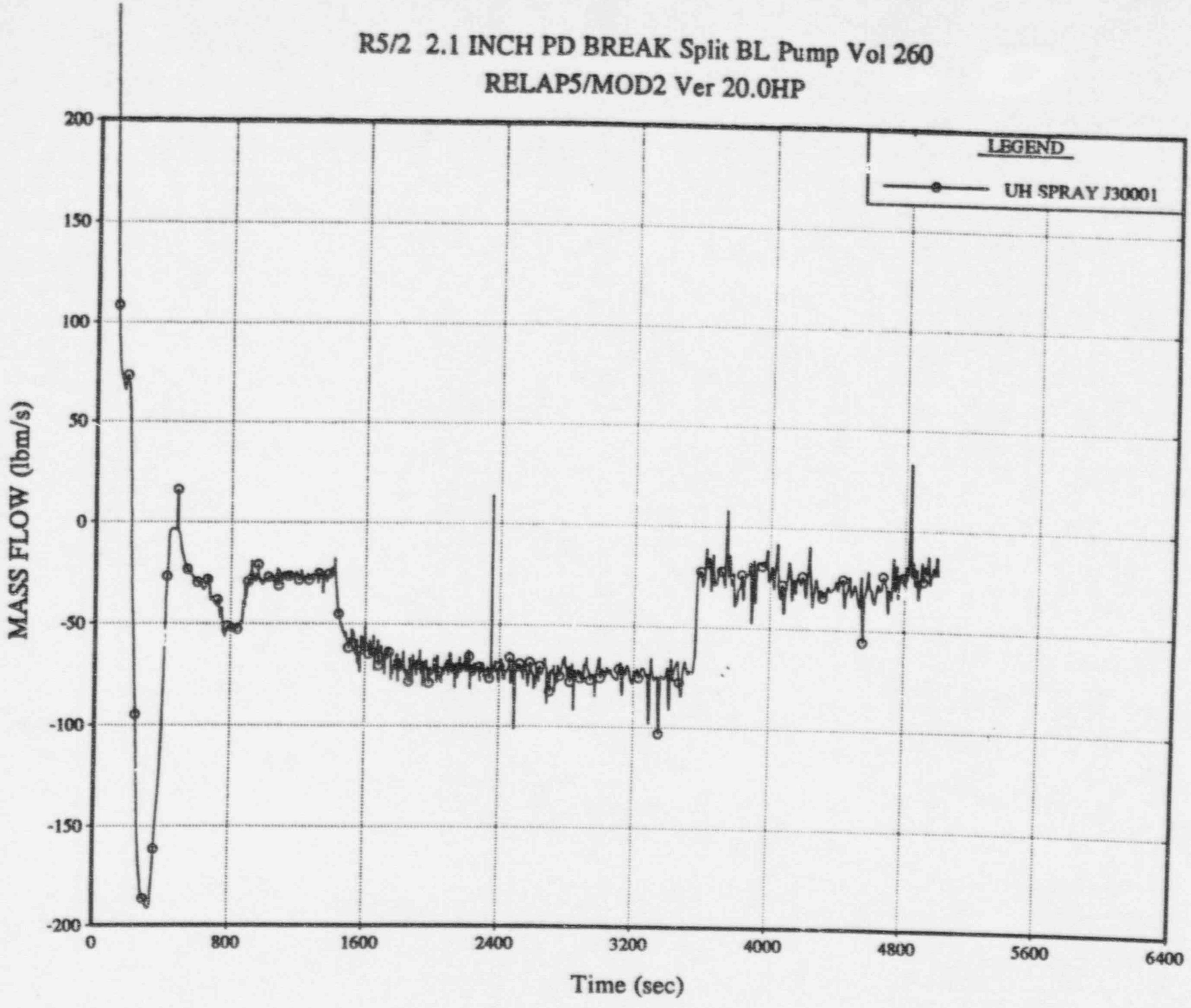
47

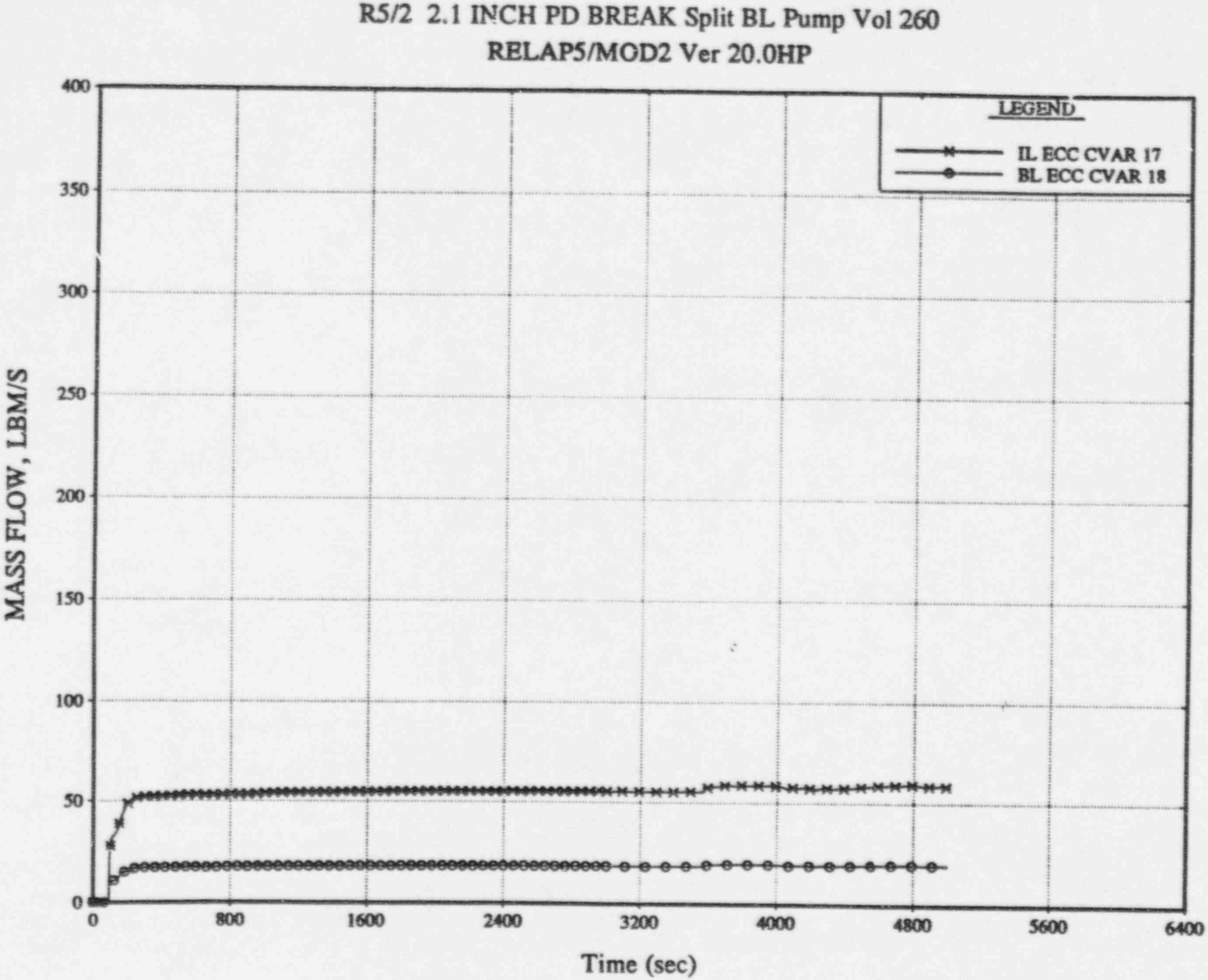
R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP




R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP
LEGEND
LEAK-J504
BREAK MASS FLOW RATE (lbm/s)
Time (sec)
0
50
100
150
200
250
300
350
400
800
1600
2400
3200
4000
4800
5600
6400

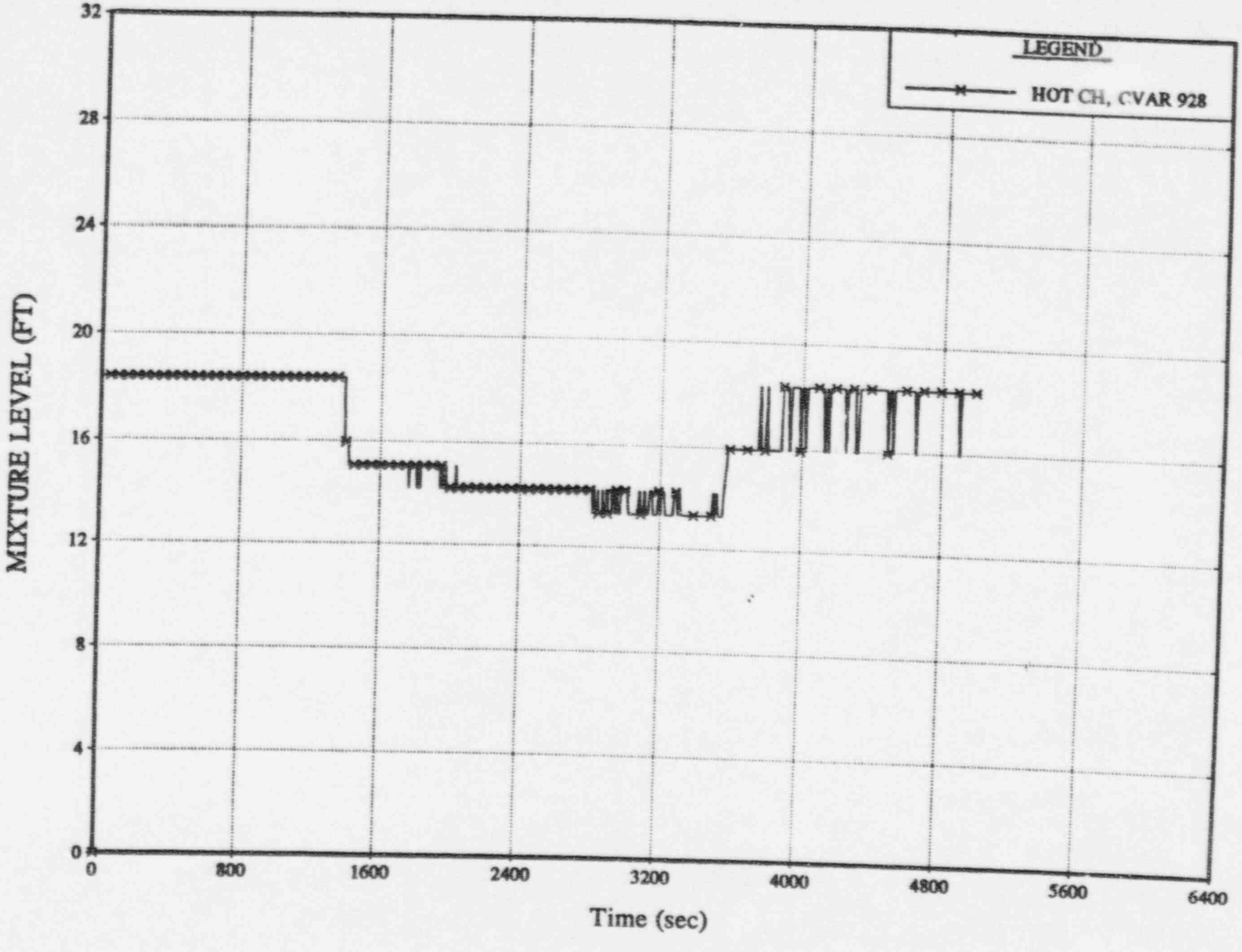
R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP




R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP
LEGEND
IL ECC CVAR 17
BL ECC CVAR 18
MASS FLOW, LBM/S
Time (sec)
0
50
100
150
200
250
300
350
400
0
800
1600
2400
3200
4000
4800
5600
6400

15

R5/2 2.1 INCH PD BREAK Split BL Pump Vol 260
RELAP5/MOD2 Ver 20.0HP



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TATB VWRO CNTRL049

SBLOCA Long-Term Cooling: In our 3/28/96 telecon, Bob Jones raised an issue as to the sufficiency of FTI's SBLOCA long-term cooling write-up on page 8-1 of BAW-10168, Revision 2, Volume II. He indicated that the appropriateness of the methodology was difficult to judge relative to the criterion of 10CFR50.46. As stated on page 8-1, FTI's SBLOCA long-term cooling methodology is basically the same as that used for LBLOCA and discussed in detail on page 8-1 of Volume I. It is repeated below.

FTI continues its transient small break LOCA computer analysis until the core is covered by mixture and the clad temperatures have decreased to the coolant saturation temperature. For the long-term, the clad will be maintained within several degrees of the coolant saturation temperature by a continuous flow of ECC water. Each plant has established NRC-approved procedures for an orderly transition to long-term cooling, assuring a continuous flow of ECC water to the reactor vessel and preventing the crystallization of boric acid in the core. The plant procedures specify the operator actions necessary to switch to sump recirculation--providing for a continuous ECC flow--and to assure a throughput of water to the core--maintaining boric acid concentrations at or below previously-established acceptable levels.

FTI plant applications performed under BAW-10168 will validate the appropriateness of previously-established operator action times, assuring the effective establishment of long-term cooling. If the need for new operator action times is demonstrated, analyses necessary to do so will be performed for and reported in the plant-specific LOCA application. For SBLOCA, such calculations are usually unnecessary, since, in general, it is bounded by LBLOCA predictions and that analysis is used to satisfy the long-term cooling criterion. In FTI's approach, the LOCA/plant procedure interface is properly addressed and in combination with as-designed plant emergency systems requirements the long-term cooling criterion of 10CFR50.46 is satisfied.

Equilibrium Core Heat Transfer Calculations: FTI's original NRC-approved evaluation model (for both large and small breaks)--BAW-10168, Revision 1--used equilibrium conditions for the RELAP5 computation of core heat transfer; this issue was thoroughly explored by the INEL reviewers and it was approved by the NRC. In Revision 2 of the EM, FRAP-T6 was deleted from the large break LOCA calculational technique. No changes were made to the core heat transfer package other than the calculations for the hot channel were now performed in RELAP5. The modeling was still based on equilibrium and it was found to be acceptable for licensing use by the NRC. In Revision 3, FRAP-T6 was deleted from SBLOCA. Again, no changes, other than code location, were made to the equilibrium core heat transfer package.

When the RSG evaluation model was originally assembled, FTI installed in RELAP5 core heat transfer correlations, covering most of the boiling curve, that were formulated based on equilibrium conditions. The RELAP5 core heat transfer package, designed after that in FRAP-T6, was used and approved for both large and small break applications. The EM was benchmarked, most recently against ROSA IV, and shown to produce conservative PCTs. FTI understands that it could upgrade RELAP5 to a nonequilibrium core heat transfer calculation, but it would require a substantial investment (code revisions, benchmarks, topical report revisions, and licensing) and there is no identified calculational or safety benefit to such a modification. Therefore, FTI has

decided to continue to use the equilibrium option. The T-H role of RELAP5 is unchanged, and an equilibrium core heat transfer calculation, previously found acceptable in FRAP-T6, is still being used and has already been approved for LBLOCA calculations. The RELAP5 equilibrium approach is NRC-approved and the removal of FRAP-T6 from the SBLOCA EM has no bearing on its continued validity.