RECOMMENDED INFORMATION FOR INCLUSION

IN SECTION 15.6.4 OF THE

FINAL SAFETY ANALYSIS REPORT FOR

R. E. GINNA NUCLEAR PLANT

WESTINGHOUSE ELECTRIC CORPORATION

NUCLEAR AND ADVANCED TECHNOLOGY DIVISION

ENGINEERING TECHNOLOGY DEPARTMENT

NUCLEAR SAFETY ANALYSIS

JULY 1992

RECOMMENDED INFORMATION FOR INCLUSION IN THE
FINAL SAFETY ANALYSIS REPORT FOR
R. E. GINNA NUCLEAR PLANT

15.6.4.2 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT ACCIDENT)

The analysis specified by 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors,"[1] is presented in this section for a major rupture of the reactor coolant system (RCS) pressure boundary for the R. E. Ginna Nuclear Plant.

15.6.4.2.1 Classification and Criteria

A major pipe rupture (large break), as considered in this section, is defined as a breech in the reactor coolant pressure boundary with a total cross-sectional area greater than 1.0 ft^2. This is considered a Condition IV event. Condition IV occurrences are faults which are not expected to occur during the lifetime of the R. E. Ginna Nuclear Plant, but are postulated because the consequences include the potential for the release of significant amounts of radioactive material.

The Condition IV major pipe rupture loss-of-coolant accident (LOCA) is the most drastic decrease in reactor coolant inventory event which must be designed against and thus represents the limiting design case for the Emergency Core Cooling System (ECCS). In Westinghouse designed nuclear steam supply system (NSSS) designs, Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100 and a single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault, including those of the ECCS and the containment. WASH-1400[2] presents the results of a study of the probability of occurrence of various accident sequences including pipe ruptures.

The R. E. Ginna Nuclear Plant reactor is designed to withstand the effects caused by a loss-of-coolant accident including the double ended severance of the largest pipe in the reactor coolant system. The

reactor core and internals together with the Safety Injection System are designed so that the reactor can be safely shut-down, the essential heat transfer geometry of the core preserved following the accident, and the long-term coolability maintained. The ECCS is designed to meet Acceptance Criteria which preclude fission product release to the environment in excess of the guideline values of 10 CFR 100.

The Acceptance Criteria for the loss-of-coolant accident, as prescribed in 10 CFR 50.46, are:

a. The calculated peak fuel element cladding temperature is below the limit of 2200°F.

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA. The ECCS is designed to meet Acceptance Criteria even when operating with the most severe single failure. During the injection mode, the loss of a safety injection pump is the limiting single failure.

15.6.4.2.2 Assumptions

For large break LOCAs, the limiting single failure is one which minimizes the amount of ECCS flow delivered to the core without reducing the capability of the emergency safeguards systems to reduce the containment pressure. A lower containment backpressure reduces the reflooding rate due to the increased difficulty in venting steam from increased steam binding. The lowest containment pressure is obtained only if all containment spray pumps and fan coolers operate subsequent to the postulated LOCA. Therefore, for the purposes of large break LOCA analyses, the most limiting single failure

would be the loss of one low head safety injection pump with full operation of the spray pumps and fan coolers.

The large break LOCA analyses for the R. E. Ginna Nuclear Plant conservatively assume both maximum containment spray and fan cooler operation (lowest containment pressure) and minimum ECCS safeguards (the loss of one SI pump), which results in the minimum delivered ECCS flow available to the RCS.

Additionally, the insertion of control rods to shut down the reactor is neglected in the large break LOCA analysis.

15.6.4.2.3 Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. Continued RCS depressurization results in accumulator injection to the intact loop cold leg. These countermeasures will limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

B. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

For the R. E. Ginna Nuclear Plant large break LOCA ECCS analysis using the WCOBRA/TRAC UPI methodology, one low head safety injection pump starts and delivers flow to the upper plenum. Additionally, three high head safety injection pumps start with one assumed to inject directly to the intact loop, one spilling on the broken loop, and one delivering to both loops such that some spills and some injects. The high head safety injection flows are modeled so that delivery to the RCS is minimized while maximizing spill to the containment. One low head safety injection pump is assumed to fail, representing the worst single failure assumption for the analysis. Assuming power to both emergency system trains is consistent with modeling full operation of the active containment heat removal systems.

Modeling the operation of all the containment heat removal systems is consistent with the US-NRC Branch Technical Position CSB 6-1[3] and is conservative for the large break LOCA.

To minimize delivery of safety injection flow to the reactor coolant system, the branch line chosen to spill is selected as the one with the minimum resistance. In addition, both low head and high head safety injection pump performance curves are assumed to be degraded.

The time sequence of events following a large break LOCA is presented in Figure 15.6.4.2-1 and Table 15.6.4.2-3. The safety injection performance during the transient, as predicted in the R. E. Ginna Nuclear Plant large break LOCA WCOBRA/TRAC UPI analysis, are presented in Figures 15.6.4.2-8A and 15.6.4.2-9A for the Appendix K calculation and in Figures 15.6.4.2-8B and 15.6.4.2-9B for the bounded calculation. Figures 15.6.4.2-14 and 15.6.4.2-15 illustrate the safety injection pump performance modelled in the WCOBRA/TRAC input.

15.6.4.2.4 Description of Large Break LOCA Transient

Before the break occurs, the R. E. Ginna Nuclear Plant is assumed to be in a full power equilibrium condition; i.e., the heat generated in the core is being removed through the steam generator secondary system. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. After the break develops, the time to departure from nucleate boiling is calculated. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes voided, both transition boiling and forced convection are considered as the dominant core heat transfer mechanisms. Heat transfer due to radiation is also considered.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of the large break LOCA, the primary pressure rapidly decreases below the secondary system pressure and the steam generators are an additional heat source. In the R. E. Ginna Nuclear Plant large break LOCA analysis using the WCOBRA/TRAC UPI methodology, the secondary system is conservatively assumed to be isolated at the initiation of the event to maximize the secondary side heat load.

When the RCS depressurizes to approximately 715 psia, the accumulators begin to inject borated water

into the reactor coolant loops. Borated water from the accumulator in the faulted loop is assumed to spill to containment and be unavailable for core cooling for breaks in the cold leg of the RCS. Flow from the accumulator in the intact loop may not reach the core during depressurization of the RCS due to the fluid dynamics present during the ECCS bypass period. ECCS bypass results from the momentum of the fluid flow up the downcomer due to a break in the cold leg which entrains ECCS flow out toward the break. Bypass of the Safety Injection diminishes as mechanisms responsible for the bypassing are calculated to be no longer effective.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere. After the end of the blowdown, refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core, BOC, recovery time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and on into the beginning of reflood, the intact loop safety injection accumulator tank rapidly discharges borated cooling water into the RCS. Although the portion injected prior to end of bypass is lost out the cold leg break, the accumulator eventually contributes to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The safety injection pumps aid in the filling of the downcomer and core and subsequently supply water to help maintain a full downcomer and complete the reflooding process. The UPI also aids the reflooding process by providing water to the core through the vessel upper plenum.

The end of the refill phase and the beginning of the reflood phase, i.e., BOC time, is not as significant an event or as easily defined for the Two-Loop UPI Large Break LOCA WCOBRA/TRAC Evaluation Model when compared to previous Westinghouse Large Break Evaluation Models. The typical practice for WCOBRA/TRAC analyses, is to report the time when low void fraction mixture is seen at the bottom of the core for BOC time. However, since a significant portion of the upper plenum safety injection water can flow down the low power/periphery channel of the core, significant cooling of the hot rod can occur prior to this time due to transverse flows within the core. In some cases, this cooling can be sufficient to cause the peak cladding temperature to occur prior to BOC time.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures

have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the sump recirculation mode of ECCS operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the safety injection pumps and returned to the upper plenum and RCS cold legs.

Long-term cooling includes long-term criticality control. Criticality control is achieved by maintaining subcriticality without credit for RCCA insertion due to the boron in the ECCS and sump. The necessary RWST and accumulator boron concentrations are a function of each core design and are verified each cycle.

15.6.4.2.5 Analysis of Effects and Consequences

15.6.4.2.5.1 Method of Analysis

The analysis was performed using the Westinghouse Large Break LOCA Best-Estimate Methodology for plants which incorporate Upper Plenum Injection (UPI) in the Safety Injection System design.[4,5] The Westinghouse Best-Estimate Methodology was developed consistent with guidelines set forth in the SECY-83-472 document.[6] These guidelines provide for the use of realistic models and assumptions in the calculational framework. The technical basis for the use of this model is discussed in detail in References 4 and 5.

The SECY-83-472 document states that there are three areas of conservatism in the current licensing models: the required Appendix K conservatism, the conservatism added by both the NRC staff and industry to cover uncertainties, and the conservatism imposed by the industry in some cases to reduce the complexity of the analysis. Based on a review of the available experimental data and the best estimate computer code calculations, the NRC staff concluded that there is more than sufficient safety margin to assure adequate performance of the ECCS, and that this excess margin can be reduced without an adverse effect on plant safety. Therefore, in the SECY-83-472 approach, the NRC staff suggested that the licensee could utilize a realistic model of the PWR to calculate the plant response to a LOCA at the most realistic (50 percent probability), i.e., the Nominal Analysis, and at a more conservative 95 percent probability level, i.e., the Bounded Analysis. The calculation at the 95 percent probability level would account for uncertainties in such things as power level, fuel initial temperature, nuclear parameters, and

computer code uncertainties. The parameters which imply uncertainty, and the methods by which the uncertainties would be combined (either statistically or as a one-sided bias) would have to be justified. The uncertainty analyses can be performed on a generic, realistic PWR model which is representative of a class of similar plants, that is, two-, three-, or four-loop PWRs so that generic uncertainties are applicable to the individual plants.

The WCOBRA/TRAC code uncertainty methodology calculation consists of two parts;

1) An assessment of the ability of WCOBRA/TRAC to model the PWR behavior[5], and

2) A quantified assessment of WCOBRA/TRAC capability to predict the measured quantities from various separate effects and systems effects experiments which cover the range of PWR accident conditions[4].

The sources of uncertainty within the code, and the specific application of the code to the PWR calculation have been addressed in accordance with requirements of SECY-83-472[5,7]. While performing this assessment it was determined that the uncertainty of several modeling effects could not be quantified by comparison to experimental data. Consequently, parametric sensitivity studies were performed which varied these modeling effects in the WCOBRA/TRAC computer code, and the uncertainty was determined based on the results of these sensitivity studies.

The numerical value for the code uncertainty was derived by comparing WCOBRA/TRAC to a wide range of experiments which covered the expected range of conditions for the PWR. The uncertainty analysis considered the following items:

1) Code bias - obtained by comparing the code calculated temperatures to the average of temperatures measured from various single effects and integral tests.

2) The uncertainty in the code bias - the standard deviation of the code bias is calculated as δ_1.

3) The uncertainty in the data for each of the experiments. The individual test data uncertainties are sample size weighted and pooled together to obtain a data uncertainty for all the experiments analyzed as δ_2.

4) The initial test condition uncertainty used in the WCOBRA/TRAC was assessed by

examination of repeat experiments and is calculated as δ_3.

5) The test modeling uncertainty was assessed by performing noding sensitivity analyses on different tests and averaging the differences between the different cases, and is calculated as δ_4.

The uncertainty analysis was undertaken for both a blowdown and reflood peak temperature. The code bias was a direct value added or subtracted from the calculated plant peak cladding temperature. The uncertainties from items 2 to 5 were statistically combined as the square-root-sum-of-squares and raised to the 95th percentile by multiplying by 1.645. The equation for the plant peak cladding temperature at the 95th percentile becomes:

$$PCT_{P \geq 95\%} = PCT_{PLANT} \pm Code\ Bias + 1.645 \sqrt{\delta_1^2 + \delta_2^2 + \delta_3^2 + \delta_4^2}$$

The nominal calculation is performed to provide assurance that the most probable PCT is well below the estimate of the 95 percent probability value. However, the nominal calculation is itself a conservative estimate since several conservative assumptions are retained.

To demonstrate compliance with the specific requirements of Appendix K to 10 CFR 50, a third calculation is performed in which the plant-specific realistic best estimate calculation includes the required Appendix K features, such as 1971 ANS decay heat plus 20 percent, Moody break flow model, no return to nucleate boiling during blowdown, etc. The realistic calculation with the Appendix K required features could be used to demonstrate compliance with the Acceptance Criteria of 10 CFR 50.46, provided that the peak cladding temperature exceeded the peak cladding temperature calculated at the 95 percent probability level but was below the Acceptance Criteria limit of 2200°F.

15.6.4.2.5.2 ECCS Evaluation Model

The Best Estimate UPI ECCS Evaluation Model is comprised of the WCOBRA/TRAC and COCO computer codes[5,8]. The WCOBRA/TRAC code is used to generate the complete transient (blowdown through reflood) system hydraulics as well as the cladding thermal analysis.

WCOBRA/TRAC is the Westinghouse version of the COBRA/TRAC[9] code originally developed by Battelle Northwest Laboratory in the late 1970's. It is an advanced computer code used to simulate

complex two-phase transient and steady-state phenomena in nuclear reactors or other large complex heat exchange equipment. WCOBRA/TRAC is a combination of two codes:

a) COBRA-TF, a 3-D, two-fluid, three-field model, capable of calculating complex flow fields in a wide variety of geometries.

b) TRAC-PD2, a 1-D, two-phase drift flux flow model used primarily to simulate piping systems.

The COBRA-TF computer code provides a transient or steady-state two-fluid, three-field representation of two-phase flow. Each field is treated in three dimensions and is compressible. Continuous vapor, continuous liquid and entrained liquid drops are the three fields. The conservation equations for each of the three fields and for heat transfer from the solid structures in contact with the fluid are solved using a semi-implicit, finite-difference numerical technique on an Eulerian mesh. The COBRA-TF vessel model features extremely flexible noding for both the hydrodynamic mesh and the heat transfer solution. The flexible noding allows representation of single rod bundle subchannel, or grouping of rod bundle subchannels into larger hydrodynamic channels.

Multiphase flows consisting of two or more fluids are separated by moving phase interfaces. In general, the phases can be present in any combination of liquid, solid, or gas. The flow pattern can assume any one of a wide variety of forms, such as bubbly flow, droplet flow, gas-particle flow, and stratified flow. Since the quantities of interest are the average behavior of each phase within the control volume, most work in multiphase flow is done using average equations across the control volume.

The average conservation equations used in COBRA-TF are derived following the methods of Ishii.[10] The average used is a simple Eulerian time average over a time interval, Δt, assumed to be long enough to smooth out the random fluctuations present in a multiphase flow but short enough to preserve any gross unsteadiness in the flow. The resulting average equations can be formulated in either the mixture form or the two-fluid form. Due to its greater physical appeal and broader range of application, and the possibility of reduced uncertainty, the two-fluid approach is used as the foundation for COBRA/TF.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction, heat and mass transfer interaction terms appearing in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The three-field formulation used in COBRA-TF is a straight-forward extension of the two-fluid model. The fields included are vapor, continuous liquid, and entrained liquid. Dividing the liquid phase into two fields is the most convenient and physically reasonable way of handling flows. For this representation of the liquid phase, the liquid can appear in both film and droplet form. This permits more accurate modeling of thermal-hydraulic phenomena such as entrainment, de-entrainment, fallback, liquid pooling and flooding.

One of the important features of the COBRA-TF vessel model is that the governing equations form a complete set. No terms are omitted particularly in the momentum equations where wall shear, momentum exchange due to turbulence and all the interfacial terms are represented. The COBRA-TF vessel model also has two energy equations to account for thermodynamic non-equilibrium between the two phases. This is particularly important for post CHF (dryout) conditions where the vapor phase can be superheated and the liquid phase remains at the saturation temperature.

A complete set of heat transfer and flow regime models is incorporated into COBRA-TF. These models are applicable over a wide range of fluid and heat transfer conditions, as required by the range of conditions found during light water reactor transients. The flow regime model covers the full range from low-void fraction, bubbly regimes to highly dispersed droplet regimes and corresponding heat transfer models exist for these flow regimes, for wall surface temperatures ranging from the fluid saturation temperature to approximately 3000°F.

WCOBRA/TRAC has been successfully utilized to analyze Westinghouse two-loop PWRs with Upper Plenum Injection[5,7]. The results of these calculations indicate that the WCOBRA/TRAC analysis method verified the safety performance of the upper plenum injection system for this class of plants. This successfully resolved a long standing US-NRC concern on the adequacy of this injection system design.

The system hydraulic transient is influenced by the containment pressure transient response to the mass and energy released from the reactor coolant system by the LOCA. In the Best Estimate UPI ECCS Evaluation Model, the containment pressure transient is provided as a boundary condition to the system hydraulic transient. The containment pressure transient applied is to be conservatively low and include the effect of the operation of all pressure reducing systems and processes. The COCO computer code[8] is used to generate the containment pressure response to the mass and energy release from the break. This containment pressure curve is then used as an input to the WCOBRA/TRAC code. It should be noted that safety injection actuation is based on the pressurizer low pressure SI signal, and not on containment pressure high pressure SI signal. Although the latter is computed to occur earlier, it is

conservative to model a later time for SI injection. Additionally, since the WCOBRA/TRAC and COCO computer codes do not run interactively, it would be difficult to model SI actuation on high containment pressure.

15.6.4.2.5.3 Plant Input Parameters and Initial Conditions

Important input parameters and initial conditions used in the analysis are listed in Tables 15.6.4.2-2A and 15.6.4.2-2B for the bounding and nominal cases respectively. The initial steady state fuel pellet temperature and fuel rod internal pressure used in the LOCA analysis was generated with the PAD 3.4 Fuel Rod Design Code[11] which has been approved by the US-NRC. The fuel parameters input to the code were at beginning-of-life (maximum densification) values.

In determining the conservative direction for bounding values and assumptions for UPI plants, many sensitivity studies were performed[5]. These sensitivities were performed using a representative two-loop plant with Upper plenum Injection (UPI) in the ECCS design. Since the representative two-loop plant has a higher peak linear heat rate and a higher core power to pumped ECCS flow ratio than the R. E. Ginna Nuclear Plant it will yield a greater change in peak cladding temperature for changes in plant parameters. These sensitivity studies were used to determine the direction of conservatism for choosing the bounding conditions for the 95th percentile calculation for the R. E. Ginna Nuclear Plant.

The parameters used in the COCO analysis to determine the containment pressure curve are presented in Table 15.6.4.2-5. The containment pressure transient used to calculate the system hydraulic transient is shown in Figure 15.6.4.2-2 for the bounded and Appendix K calculations.

Initial conditions for the R. E. Ginna Nuclear Plant large break LOCA analysis are delineated in Table 15.6.4.2-1. Most of these parameters were chosen at their limiting values in order to provide a conservative bound for evaluation of the calculated peak cladding temperature for the large break LOCA analysis. The hot assembly was located under a source plate location in the upper core plate. Past sensitivity studies showed the limiting location for peak cladding temperature to be an open hole[5] and the source plate location for the R. E. Ginna Nuclear Plant possesses the limiting flow area among the various upper core plate geometries.

15.6.4.2.5.4 Description of Appendix K Calculation

An Appendix K calculation was performed for the R. E. Ginna Nuclear Plant which conforms to the modeling requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50. The conservative assumptions used in the Appendix K calculation are listed in Table 15.6.4.2-2A.

Shortly after the break is assumed to open, the vessel rapidly depressurizes (Figure 15.6.4.2-4A) and the core flow quickly reverses. During the flow reversal, the hot assembly fuel rods dry out and begin to heat up momentarily (Figure 15.6.4.2-3A).

At approximately 9 seconds into the transient, maximum downflow is reached in the high and low power regions of the core. Figure 15.6.4.2-6A shows the liquid, vapor and entrained liquid flow rates at the bottom of the average power interior assemblies. Figure 15.6.4.2-11A shows the same three quantities for the guide tube assemblies. Similarly, Figure 15.6.4.2-12A and Figure 15.6.4.2-13A show the same three quantities for the low power/periphery assemblies and for the hot assembly respectively. This flow is sufficient to cool the hot assembly and maintain the rest of the core near the fluid saturation temperature (Figure 15.6.4.2-3A).

As the vessel continues to depressurize, liquid inventory continues to be depleted, and core void fractions increase (Figure 15.6.4.2-7A). This results in reduced core flow and resulting cladding heatup, first for the hot assembly, and later for the other regions of the core.

At approximately 8 seconds into the transient, the accumulator begins to inject water into the intact cold leg (Figure 15.6.4.2-10A). This water fills the cold leg and upper downcomer region, and is bypassed to the break initially. At approximately 19 seconds, accumulator water begins to flow into the lower plenum.

At approximately 20 seconds, pumped injection into the cold leg and into the upper plenum begins (Figure 15.6.4.2-8A and -9A). This water begins to flow through the low power peripheral region of the core, and contributes to some core cooling, but primarily flows through the core into the lower plenum.

At approximately 47 seconds, the lower plenum has filled to the point that water begins to reflood the core from below. The void fraction in the upper plenum begins to decrease (Figure 15.6.4.2-5A), as well as the core void fraction (Figure 15.6.4.2-7A). At this time core cooling increases substantially and

the peak cladding temperature begins to decrease.

Figure 15.6.4.2-5A shows the void fraction in the upper plenum interior global channel, below the hot leg elevation. The upper plenum interior global channel is that region above the upper core plate which has no flow path from the core (i.e., above the interior solid metal portions of the upper core plate).

The peak cladding temperature calculated for the R. E. Ginna Nuclear Plant Appendix K large break LOCA analysis is 1928°F, assuming a peak hot rod power of 15.218 kw/ft. This result is below the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is well below the embrittlement Acceptance Criteria limit of 17 percent. The limiting total core metal-water reaction is also less than 1.0 percent, in accordance with the Acceptance Criteria. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

15.6.4.2.5.5 Description of Bounded Calculation

A calculation was performed for the R. E. Ginna Nuclear Plant which combined all the parameters at their conservative values. This calculation has been shown in previous studies to conservatively estimate the 95 percent probability PCT, and is called the Bounded calculation. The conservative assumptions used in the Bounded calculation are listed in Table 15.6.4.2-2B.

The very same figures described for the Appendix K calculation are provided for the Bounded calculation.

The peak cladding temperature calculated for the R. E. Ginna Nuclear Plant bounded large break LOCA analysis, with uncertainty, is 1842°F, assuming a peak hot rod power of 13.40 kw/ft. This result is below the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is well below the embrittlement Acceptance Criteria limit of 17 percent. The limiting total core metal-water reaction is also less than 1.0 percent, in accordance with the Acceptance Criteria. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

15.6.4.2.6 Conclusions

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria. These criteria are as follows:

1. The calculated peak fuel element cladding temperature is below the requirements of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

In keeping with the SECY-83-472 approach, large break LOCA analyses were performed for most probable (50 percent probability - also called nominal) level, the 95 percent probability level (known as a "bounded" calculation), and an Appendix K calculation. - Table-15.6.4.2-4 has a summary of the results for both the Appendix K case and the Bounded case. The Appendix K calculation had a peak cladding temperature of 1928°F. The bounded calculation resulted in a peak cladding temperature of 1667°F. With a total bias and uncertainty value of 175°F[7], the 95th probability value is 1667G°F + 175°F = 1842°F. These results clearly meet the Acceptance Criteria specified in 10 CFR 50.46.

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TABLE 15.6.4.2-1

INITIAL CONDITIONS FOR THE R. E. GINNA NUCLEAR PLANT LARGE BREAK LOCA ANALYSIS

PARAMETER	APPENDIX K AND BOUNDED ANALYSIS DESIRED VALUE
Plant Internals	Flat Upper Support Plate
Barrel Baffle Design	Downflow
Core Bypass Flow	4.5%
NSSS Power, 102% of (MWT)	1520
System Pressure (psia)	2250.
Primary System Fluid Temperatures	
T_{HOT} (°F)	605.3
T_{COLD} (°F)	541.7
$T_{UPPER\ HEAD}$ (°F)	605.3
Fuel Type	14 x 14 OFA (Optimized Fuel Assembly)
Fuel Stored Energy	Beginning of Life
Fuel Data Source	Pad 3.4[11]
Fuel Rod Backfill Pressure (psig)	275.
$F_Q T$	2.50
$F_{\Delta H}$	1.70
Peak Linear Power, kw/ft (Appendix K)	15.22
Peak Linear Power, kw/ft (Bounded)	13.40
Relative Power in the Outer Core Channel	0.6
Loop Flow Rate (GPM)	83,000
Reactor Coolant Pumps	Running
Steam Generator Tube Plugging (Symmetric)	20%

TABLE 15.6.4.2-1 (Continued)

INITIAL CONDITIONS FOR THE R. E. GINNA NUCLEAR PLANT LARGE BREAK LOCA ANALYSIS

PARAMETER	APPENDIX K AND BOUNDED ANALYSIS DESIRED VALUE
Steam Generator Secondary Pressure (psia)	697.3
Accumulators In Operation	2 (one injects into the Intact loop, one spills to containment)
Accumulator Conditions per Accumulator:	
Water Volume (ft^3)	1,100.
Nitrogen Pressure (psia)	715.
Water Temperature (°F)	90.
Safety Injection Conditions -	
Pumps in Operation:	1 LHSI Into Upper Plenum and 3 HHSI Into Two Cold Legs
Pump Flow	Degraded
Water Temperature (°F)	90.0
Delay Time (seconds)	12.0
Containment Pressure (psia)	14.7

TABLE 15.6.4.2-2A

ASSUMPTIONS USED IN THE APPENDIX K CALCULATION

1. PLANT CONFIGURATION
   - a. Pressurizer in Intact Loop
   - b. Total Peaking Factor ($F_Q T$) at 2.50
   - c. Nuclear Enthalpy Rise Peaking Factor ($F_{\Delta H}$) at 1.70
   - d. 102% of 1520 MWt
   - e. 20% steam generator tube plugging level
   - f. Thermal design minimum loop flow rate
   - g. Beginning of cycle fuel temperature
   - h. Beginning of cycle fuel pressure
   - i. Conservative power distribution

2. SAFETY INJECTION CONFIGURATION
   - a. Worst single failure
   - b. One line HHSI spilling to containment pressure
   - c. Maximum safety injection delay time

3. MODEL ASSUMPTIONS
   - a. Accumulator nitrogen modeled
   - b. Conservative reactor coolant pump two-phase model
   - c. Cross flow de-entrainment
   - d. No locked pump rotor during reflood
   - e. Limiting break discharge coefficient (0.4)
   - f. Lower bound containment pressure
   - g. ANS 1971 Decay Heat Plus 20%
   - h. Baker-Just Metal Water reaction
   - i. Swelling and blockage model
   - j. Can not return to nucleate boiling during blowdown
   - k. Clad burst
   - l. ECCS bypass

TABLE 15.6.4.2-2B

ASSUMPTIONS USED IN THE BOUNDED CALCULATION

1. PLANT CONFIGURATION

   a. Pressurizer in Intact Loop

   b. Total Peaking Factor ($F_Q T$) at 2.242

   c. Nuclear Enthalpy Rise Peaking Factor ($F_{\Delta H}$) at 1.70

   d. 102% of 1520 MWt

   e. 20% steam generator tube plugging level

   f. Thermal design minimum loop flow rate

   g. Beginning of cycle fuel temperature

   h. Beginning of cycle fuel pressure

   i. Conservative power distribution

2. SAFETY INJECTION CONFIGURATION

   a. Worst single failure

   b. One line HHSI spilling to containment pressure

   c. Maximum safety injection delay time

3. MODEL ASSUMPTIONS

   a. No accumulator nitrogen modeled

   b. Conservative reactor coolant pump two-phase model

   c. No cross flow de-entrainment

   d. Locked pump rotor during reflood

   e. Limiting break discharge coefficient (0.6)

   f. Lower bound containment pressure

   g. Decay heat at 95/95 upper bound for hot rod

   h. Metal water reaction at 95/95 upper bound for hot rod

TABLE 15.6.4.2-3

LARGE BREAK

TIME SEQUENCE OF EVENTS FOR DECLG BREAK

(All Results Are From The WCOBRA/TRAC Computer Code)

	APPENDIX K Time (seconds)	BOUNDED Time (seconds)
Start	0.0	0.0
Reactor Trip Signal	~0.1	~0.1
Accumulator Injection Begins	8.0	6.0
S.I. Signal	2.0	2.0
Blowdown Peak Cladding Temperature Occurs	10.0	9.0
End of Blowdown	22.0	19.0
Pumped Safety Injection Begins (HHSI)	12.0	12.0
Bottom of Core Recovery	29.7	29.7
Reflood Peak Cladding Temperature Occurs	47.2	36.5
Accumulator Water Empty	52.3	49.4

TABLE 15.6.4.2-4

LARGE BREAK

DECLG

Results	APPENDIX K	BOUNDED
Calculated Peak Cladding Temp., °F	1928.	1667.
Peak Cladding Temp. (P > 95%), °F	1928.	1842.*
Peak Cladding Temp. Location, Ft.	7.625	5.875
Local Zr/H_2O Reaction (max), %	<17.0	<17.0
Local Zr/H_2O Location, Ft.	7.875	-
Total Zr/H_2O Reaction, %	<1.0	<1.0

* Peak cladding temperature at the 95th percentile probability level with 95 percent confidence is obtained by adding the calculated peak cladding temperature to the code bias plus uncertainties, as discussed in Section 15.6.4.2.5.1. The sum of the code bias plus uncertainties has been determined to be 175°F as specified in Reference 7.

TABLE 15.6.4.2-5

LARGE BREAK CONTAINMENT DATA (DRY CONTAINMENT)

NET FREE VOLUME	1.066×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90°F
RWST Temperature	60°F
Service Water Temperature	35°F
Outside Temperature	-10°F
SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flow Rate	1,800 gpm each
Actuation Time	10 seconds[1]
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	4
Fastest Post Accident Initiation of Fan Coolers	30 seconds

[1] Although a later spray initiation time can be justified, modelling an earlier spray start time is conservative for large break LOCA analyses.

TABLE 15.6.4.2-5 (Continued)

STRUCTURAL HEAT SINK DATA

Descriptive Surface	Thickness	Material	Area,(ft^2)
Insulated Portion of dome and containment wall	1.25 in.	Insulation	36285
	0.375 in.	Steel	
	3.5 ft.	Concrete	
Uninsulated Portion of dome	0.375 in.	Steel	12370
	2.5 ft.	Concrete	
Basement floor	2.0 ft.	Concrete	6576
	0.25 in.	Steel	
	2.0 ft.	Concrete	
Walls of sump in basement floor	2.0 ft.	Concrete	2480
	0.25 in.	Steel	
	3.0 ft.	Concrete	
Floor of sump	2.0 ft.	Concrete	400
	0.25 in.	Steel	
	1.0 ft.	Concrete	
Inside of refueling cavity	0.25 in.	Steel	5609
	2.5 ft.	Concrete	
Bottom of refueling cavity	0.25 in.	Steel	1143
	2.5 ft.	Concrete	
Area on outside of refueling cavity walls	2.5 ft.	Concrete	6750
Area inside of loop and steam generator compartment	4.0 ft.	Concrete	10370
Intermediate level floor area	6.0 ft.	Concrete	5320
Operating floor	2.0 ft.	Concrete	6500
1.48 in. thick I-beam	1.48 in.	Steel	1822
0.9 in. thick I-beam	0.9 in.	Steel	680

TABLE 15.6.4.2-5 (Continued)

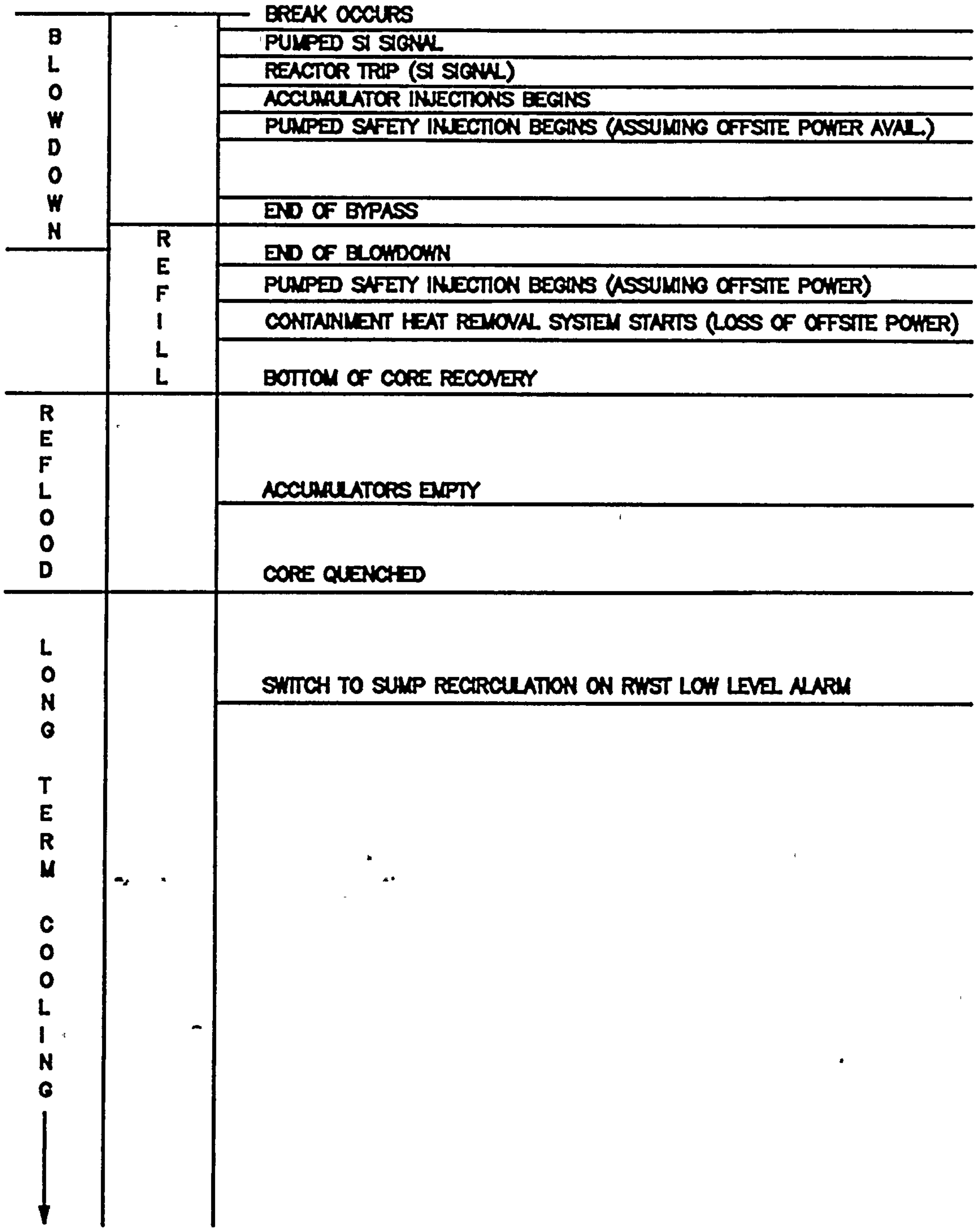
STRUCTURAL HEAT SINK DATA (continued)

Descriptive Surface	Thickness	Material	Area,(ft^2)
0.52 in. thick I-beam	0.52 in.	Steel	4180
0.61 in. thick I-beam	0.61 in.	Steel	1190
Cylindrical Supports for steam generator and reactor coolant pumps	0.5 in.	Steel	470
Plant crane support columns	0.75 in.	Steel	4810
Beams used for crane structure	1.5 in.	Steel	3390
Structure on operating floor	2.0 ft.	Concrete	2060
Grating, stairs, miscellaneous steels	0.125 in.	Steel	7000

FIGURE 15.6.4.2-1

LARGE BREAK LOCA

SEQUENCE OF EVENTS



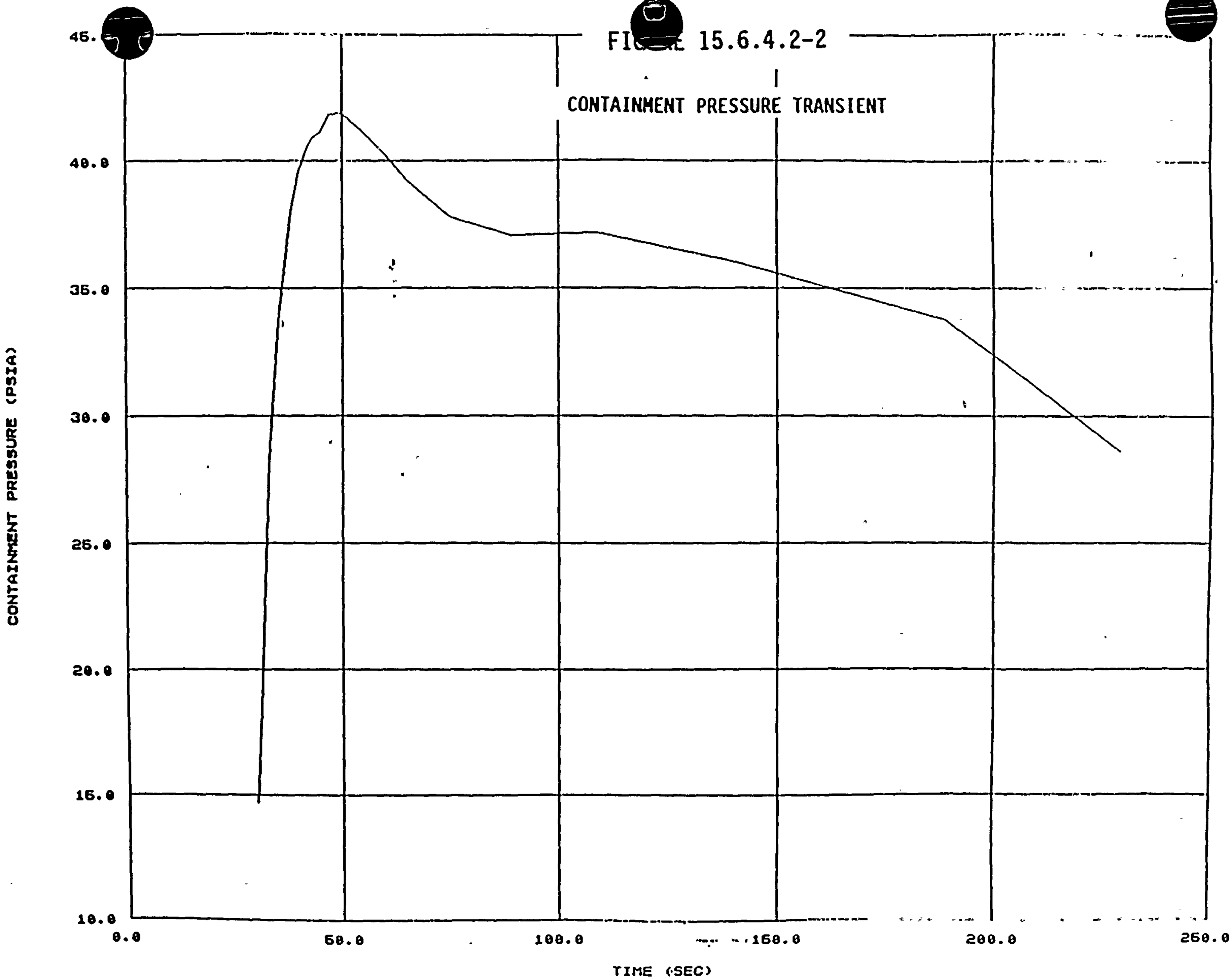


FIGURE 15.6.4.2-2

CONTAINMENT PRESSURE TRANSIENT

FIGURE 15.6.4.2-3A

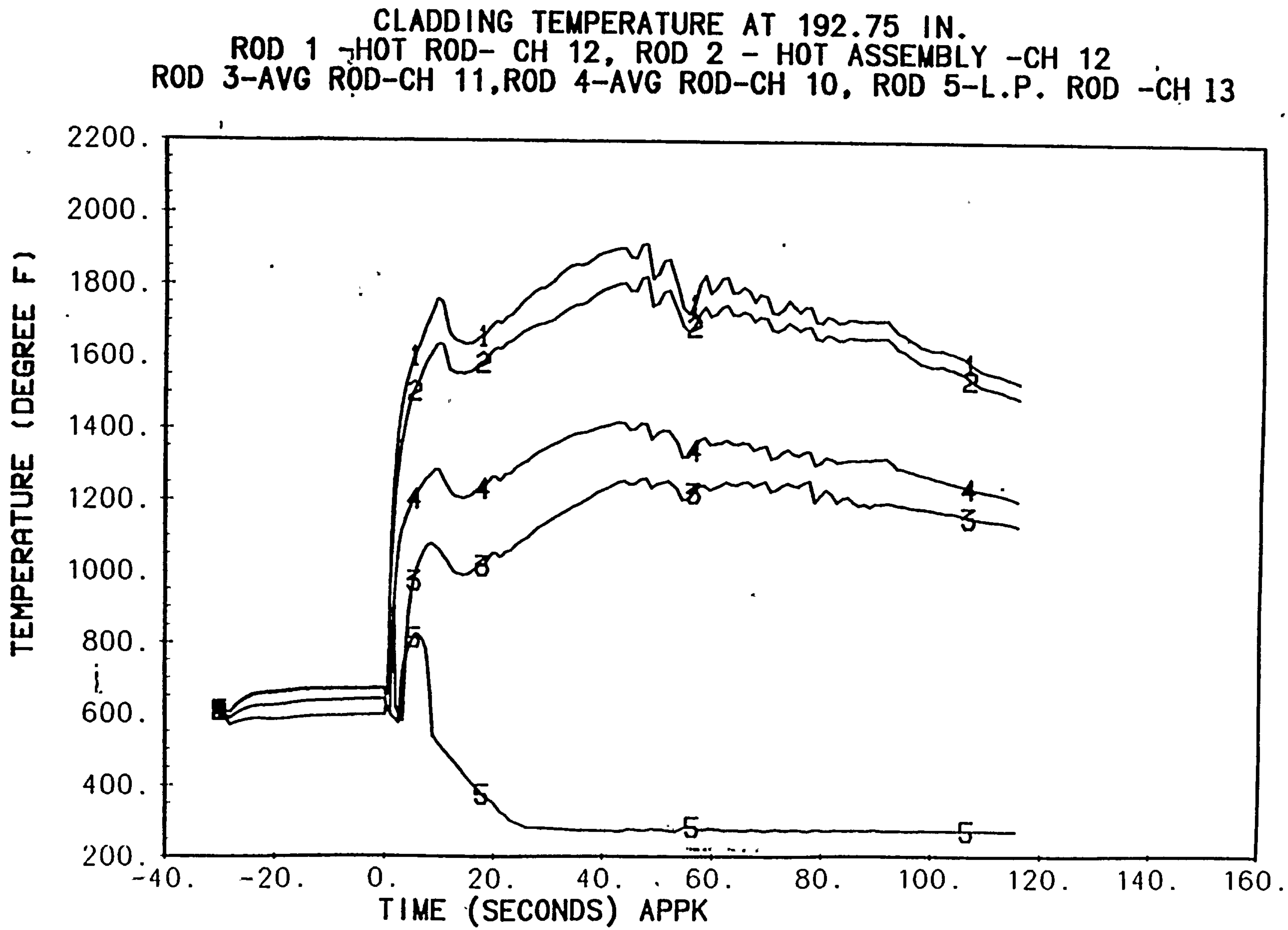


FIGURE 15.6.4.2-3B

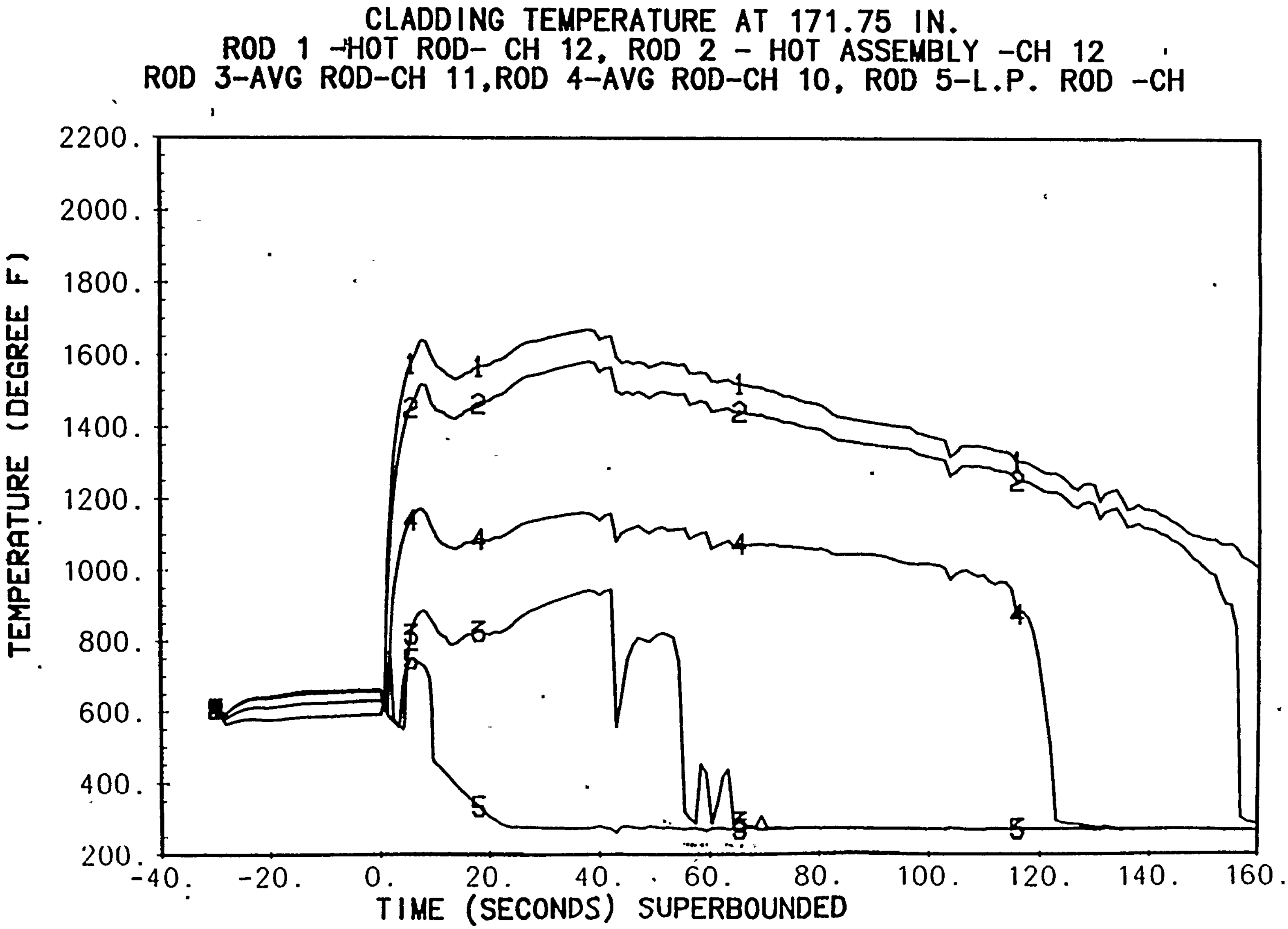


FIGURE 15.6.4.2-4A

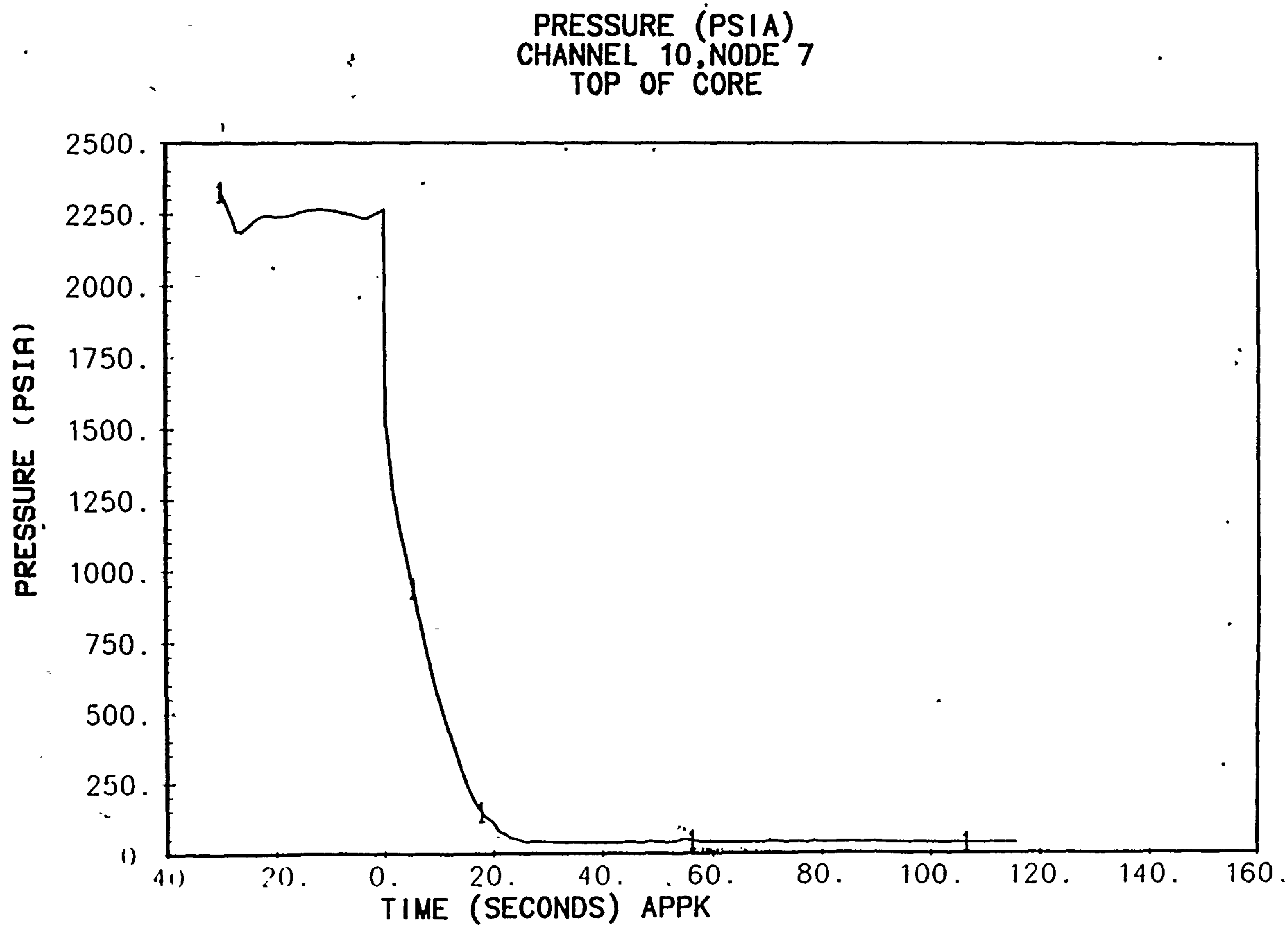


FIGURE 15.6.4.2-4B

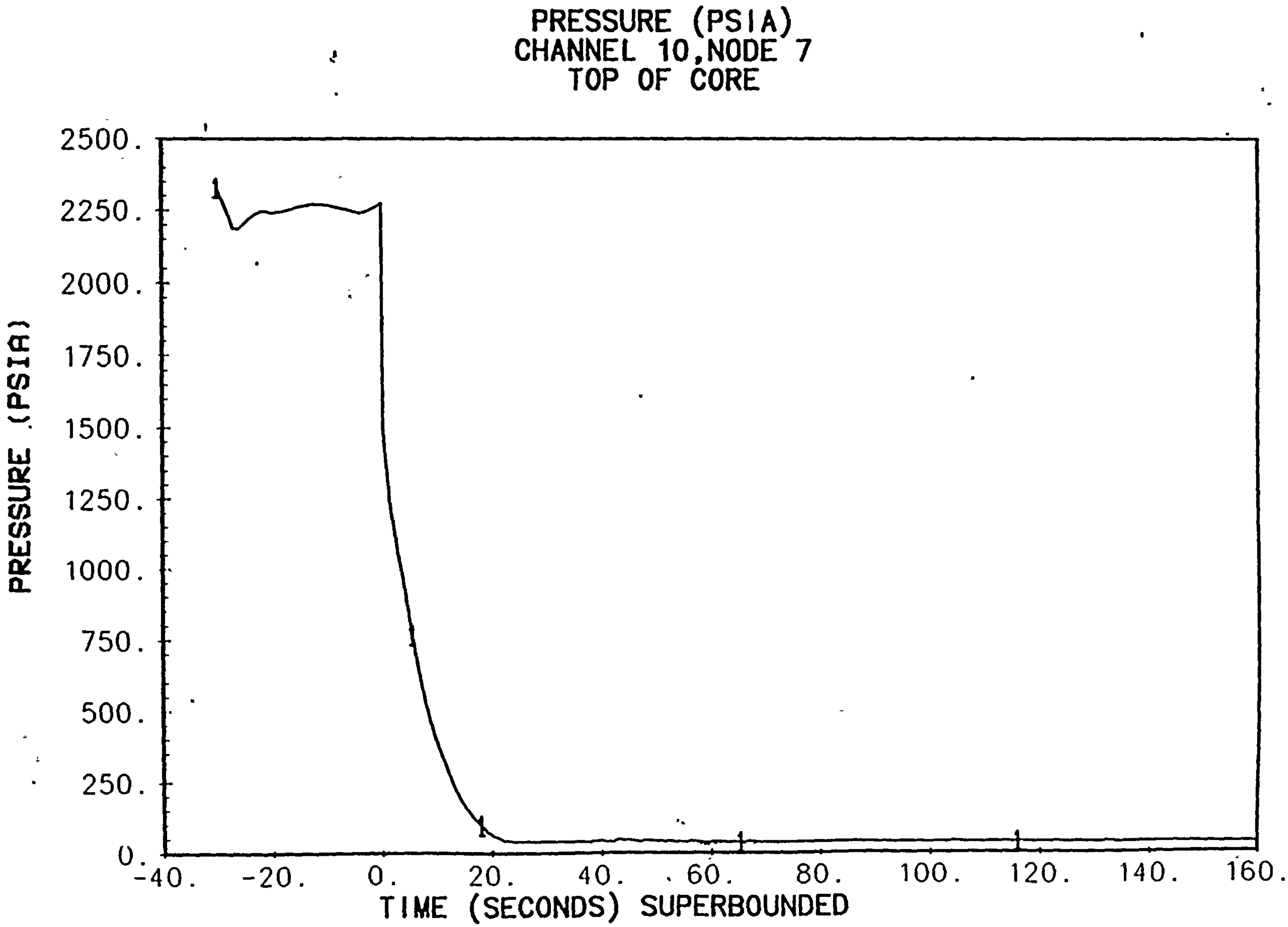


FIGURE 15.6.4.2-5A

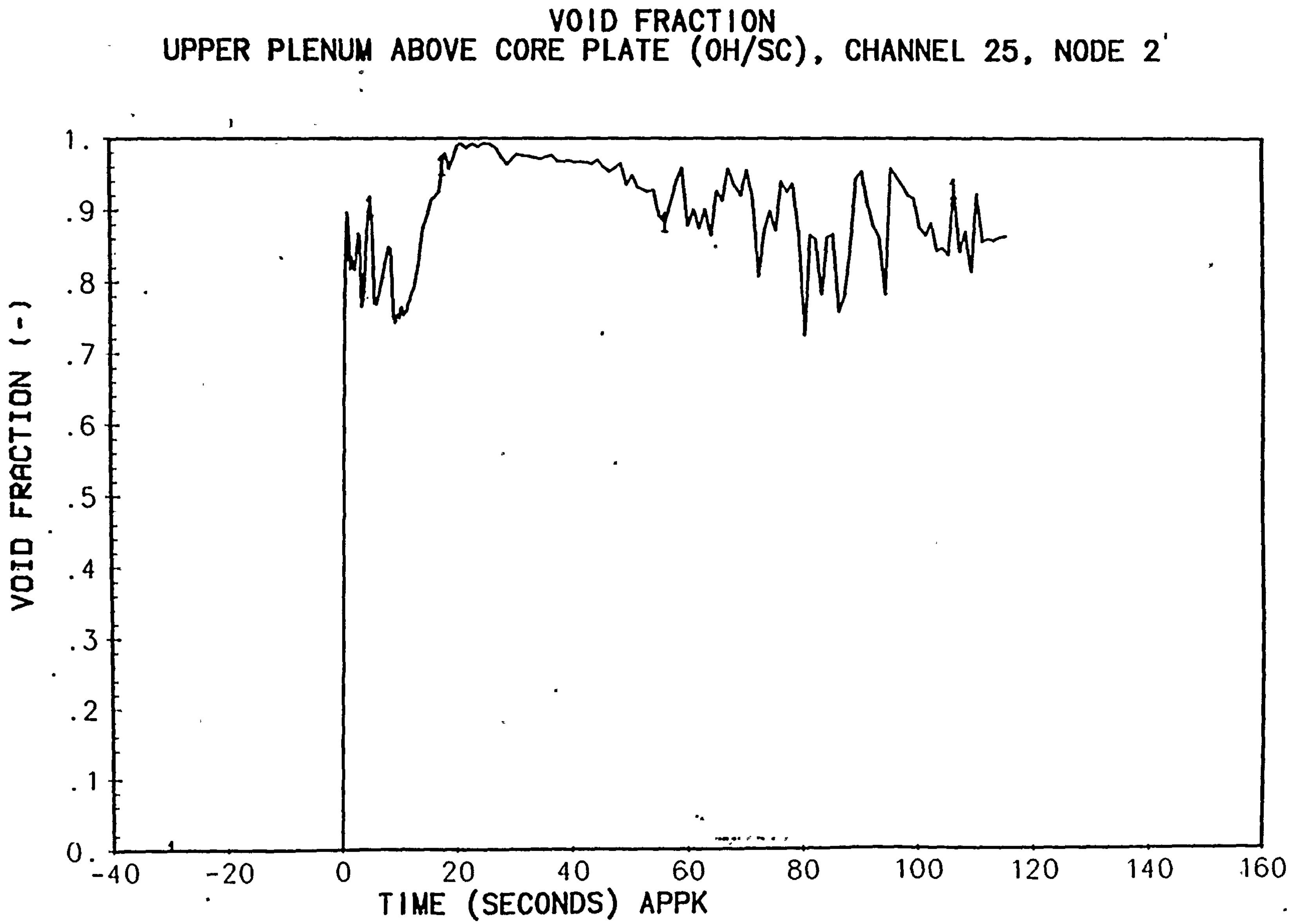


FIGURE 15.6.4.2-5B

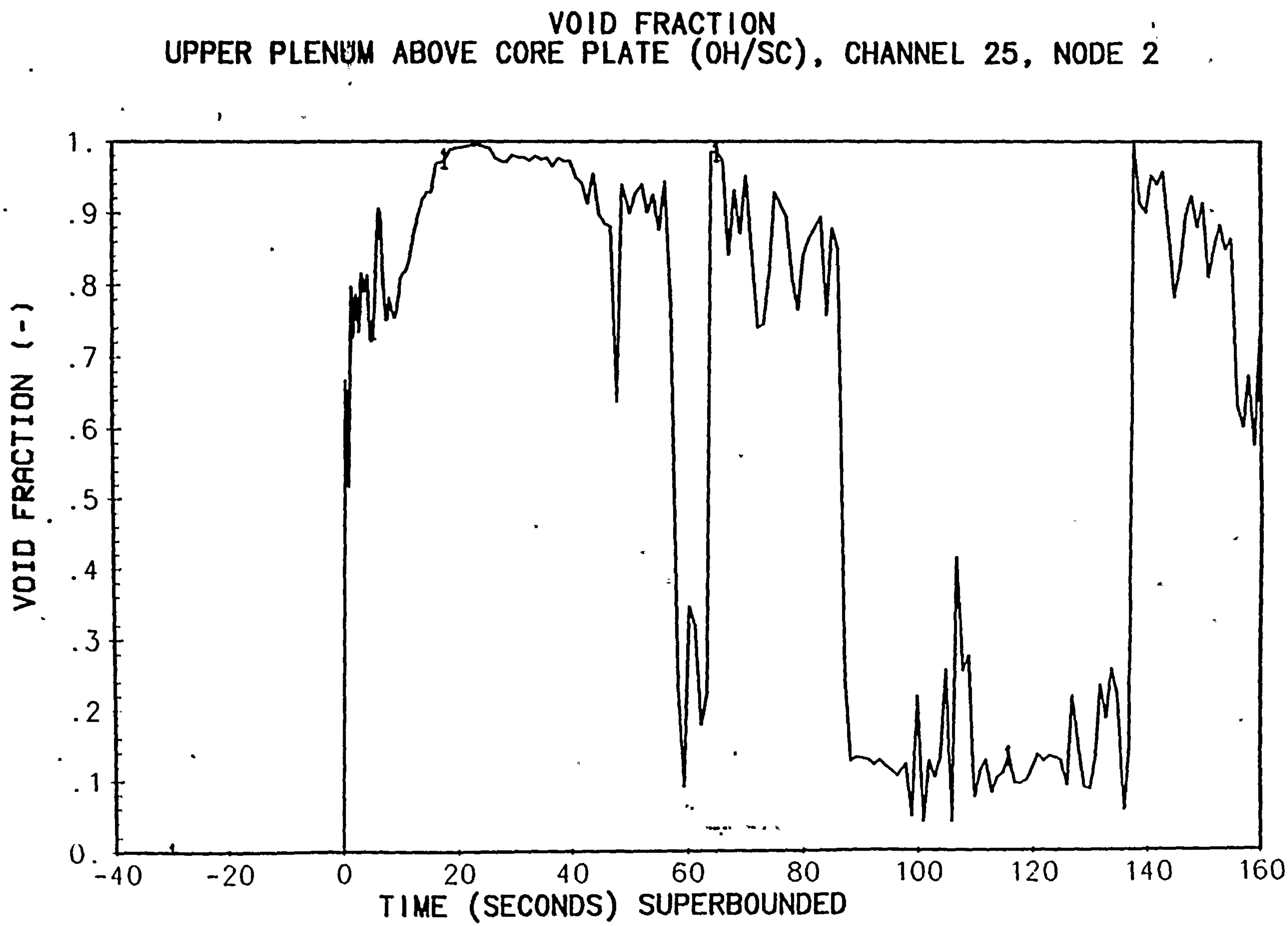


FIGURE 15.6.4.2-6A

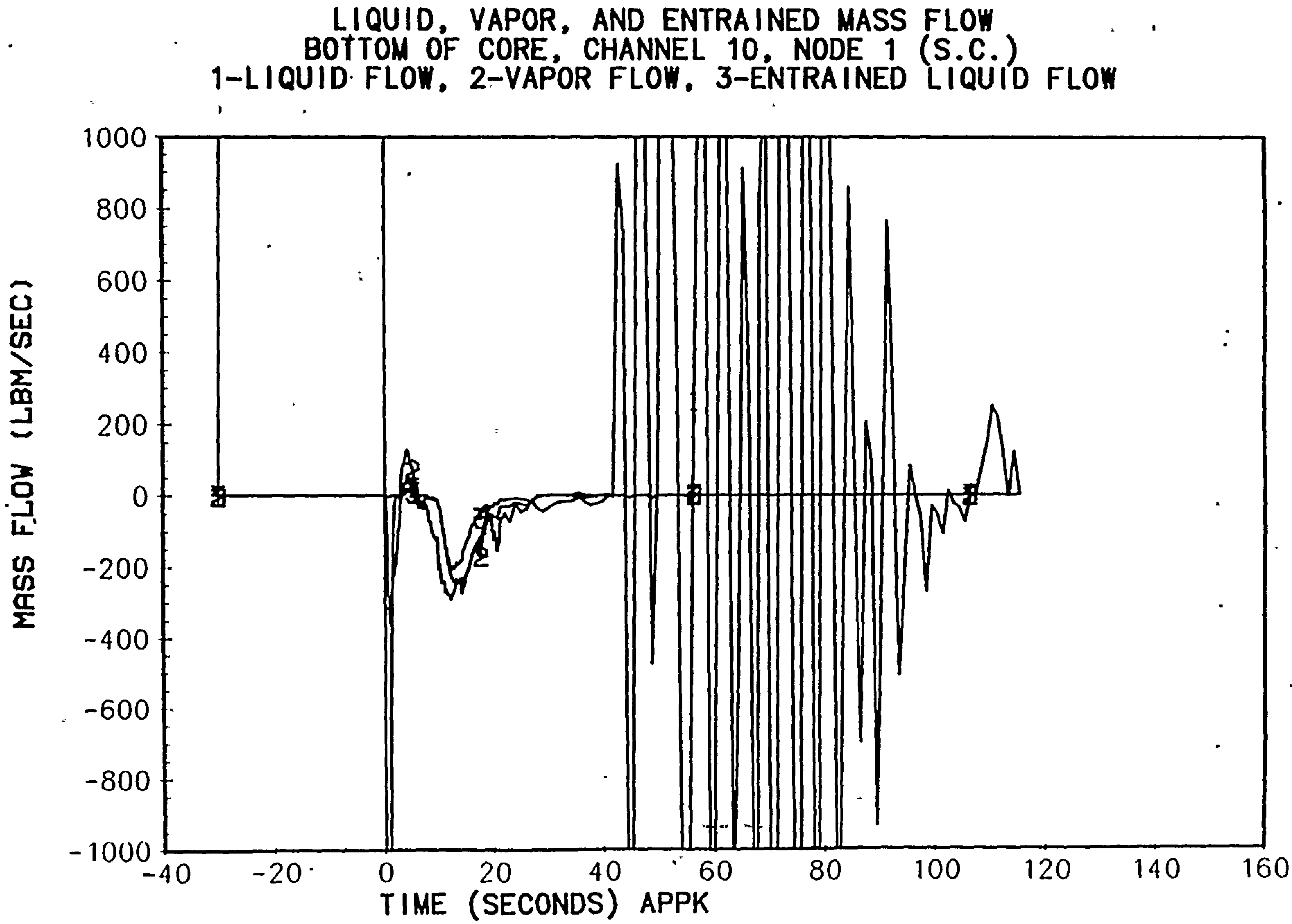


FIGURE 15.6.4.2-6B

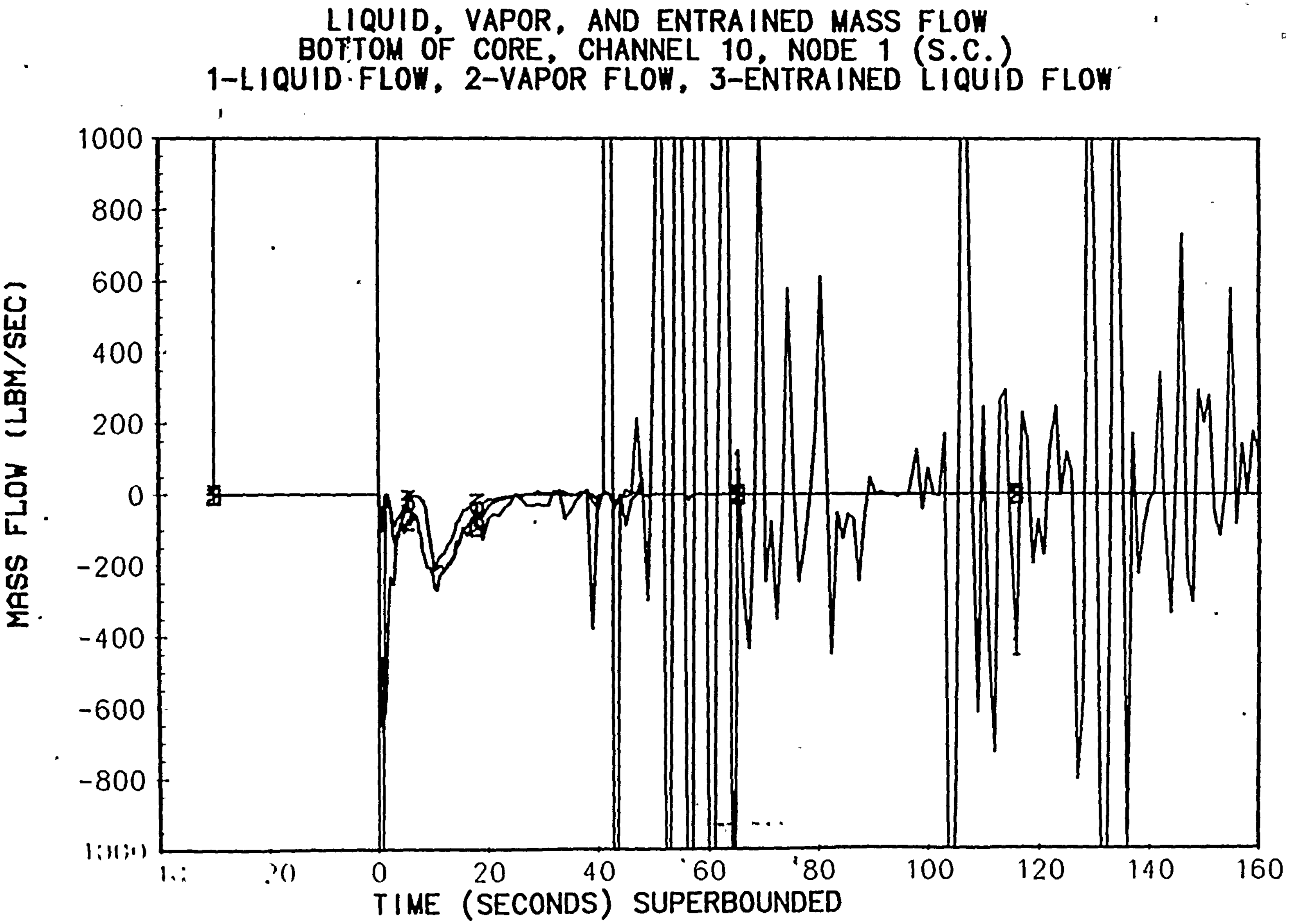


FIGURE 15.6.4.2-7A

VOID FRACTION
CORE TOP AND BOTTOM - CHANNEL 10 NODES 2,7

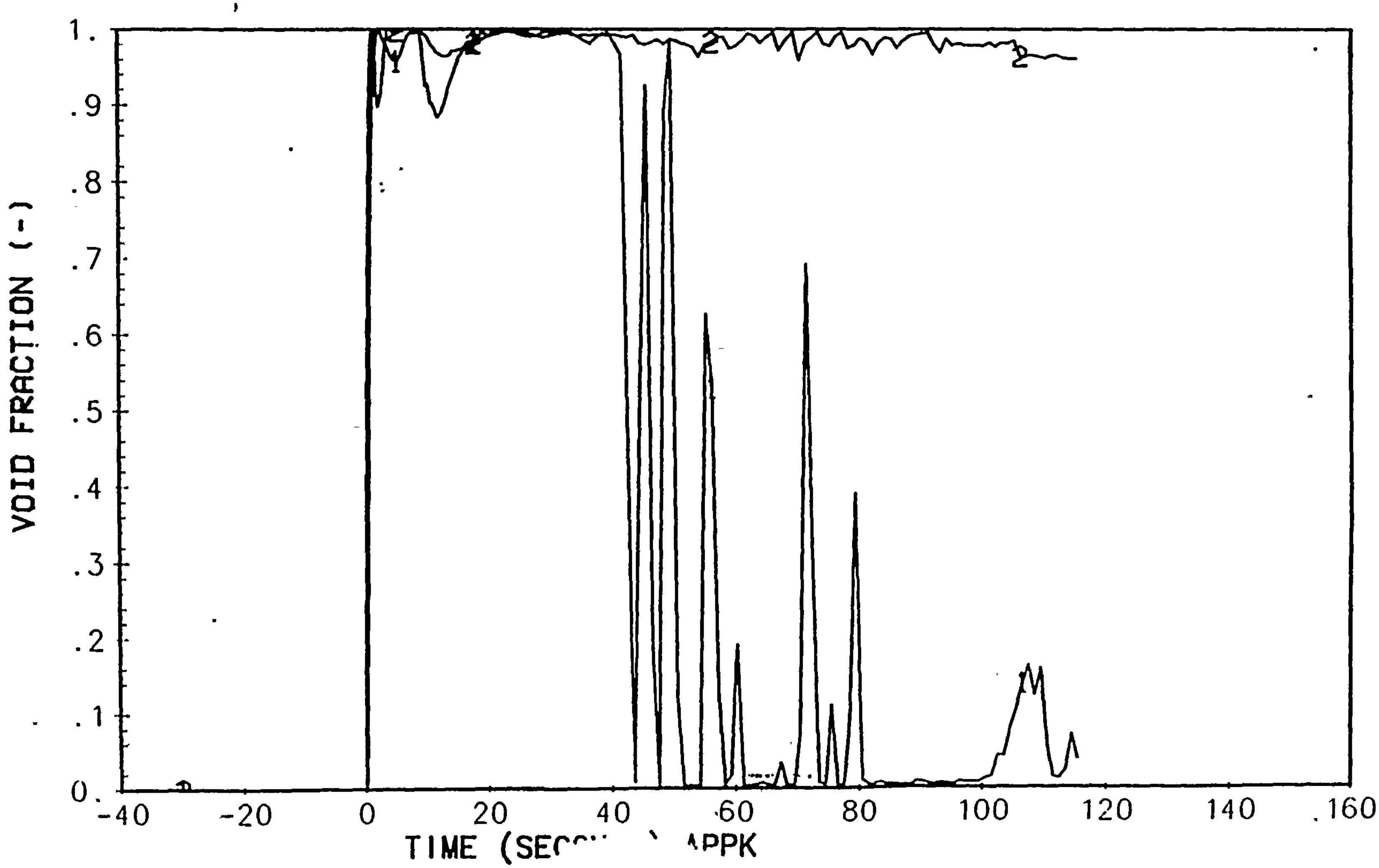


FIGURE 15.6.4.2-7B

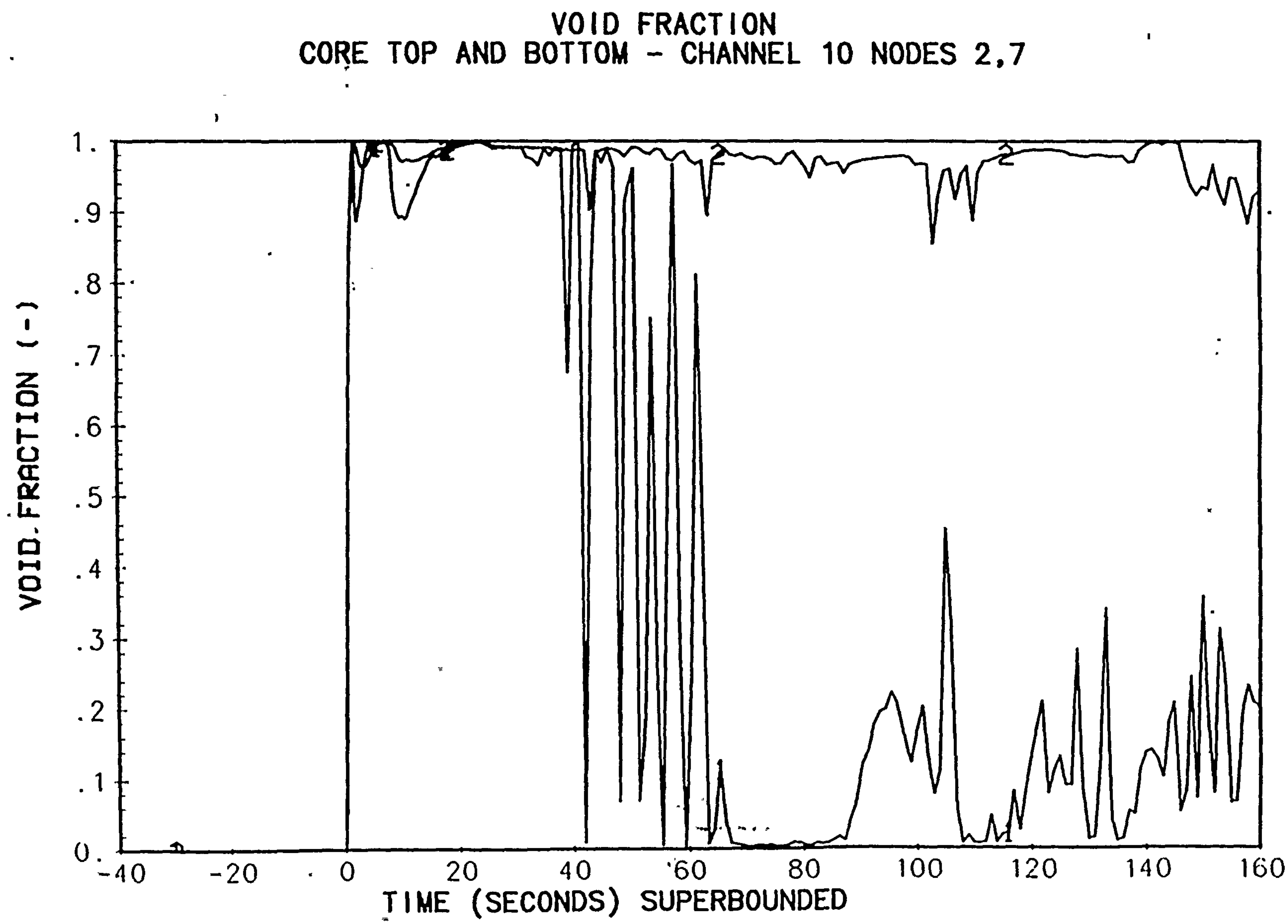


FIGURE 15.6.4.2-8A

TOTAL FLOW
RHR FLOW UPPER PLENUM
COMPONENT 22,CELL 1

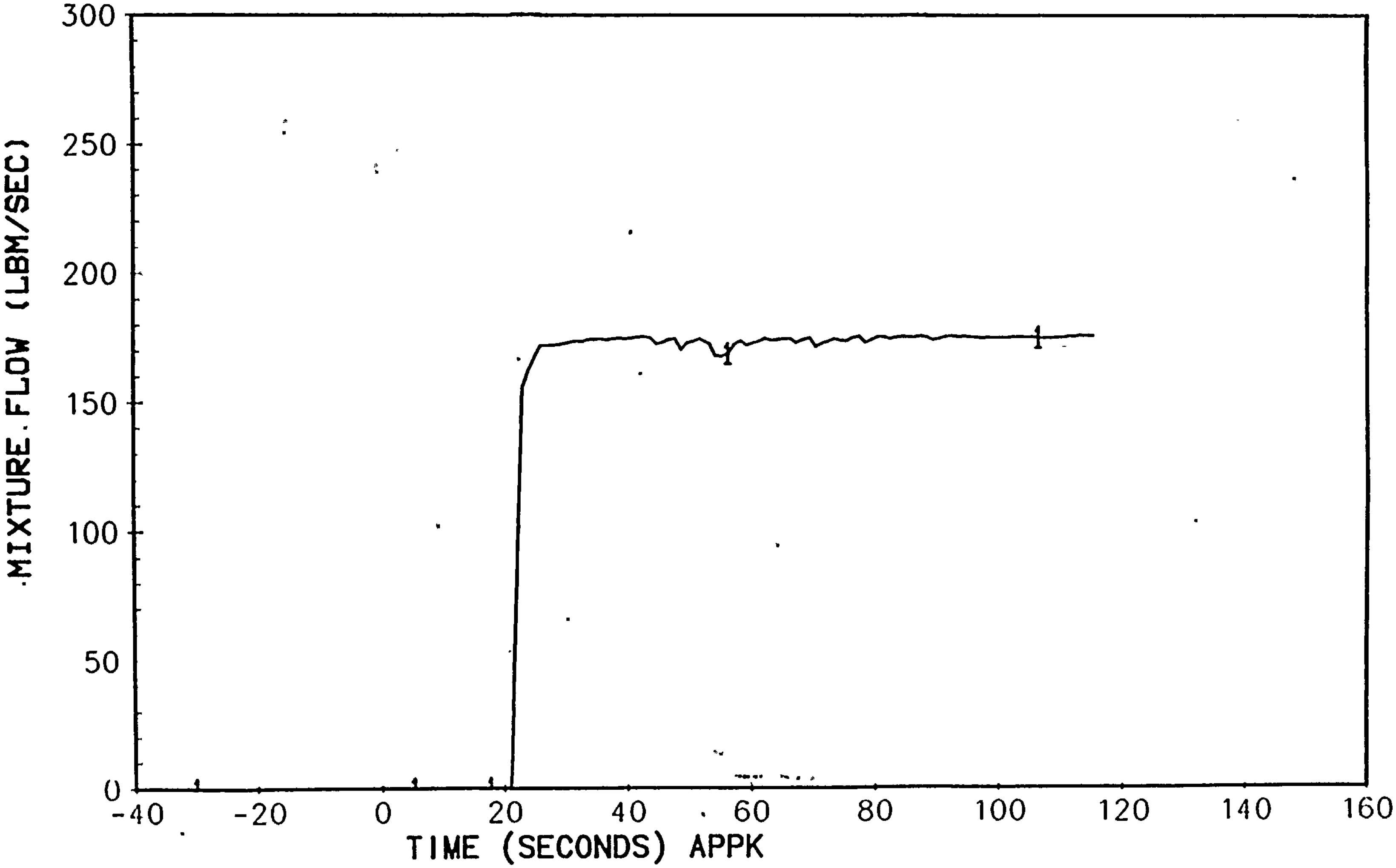


FIGURE 15.6.4.2-8B

TOTAL FLOW
RHR FLOW UPPER PLENUM
COMPONENT 22,CELL 1

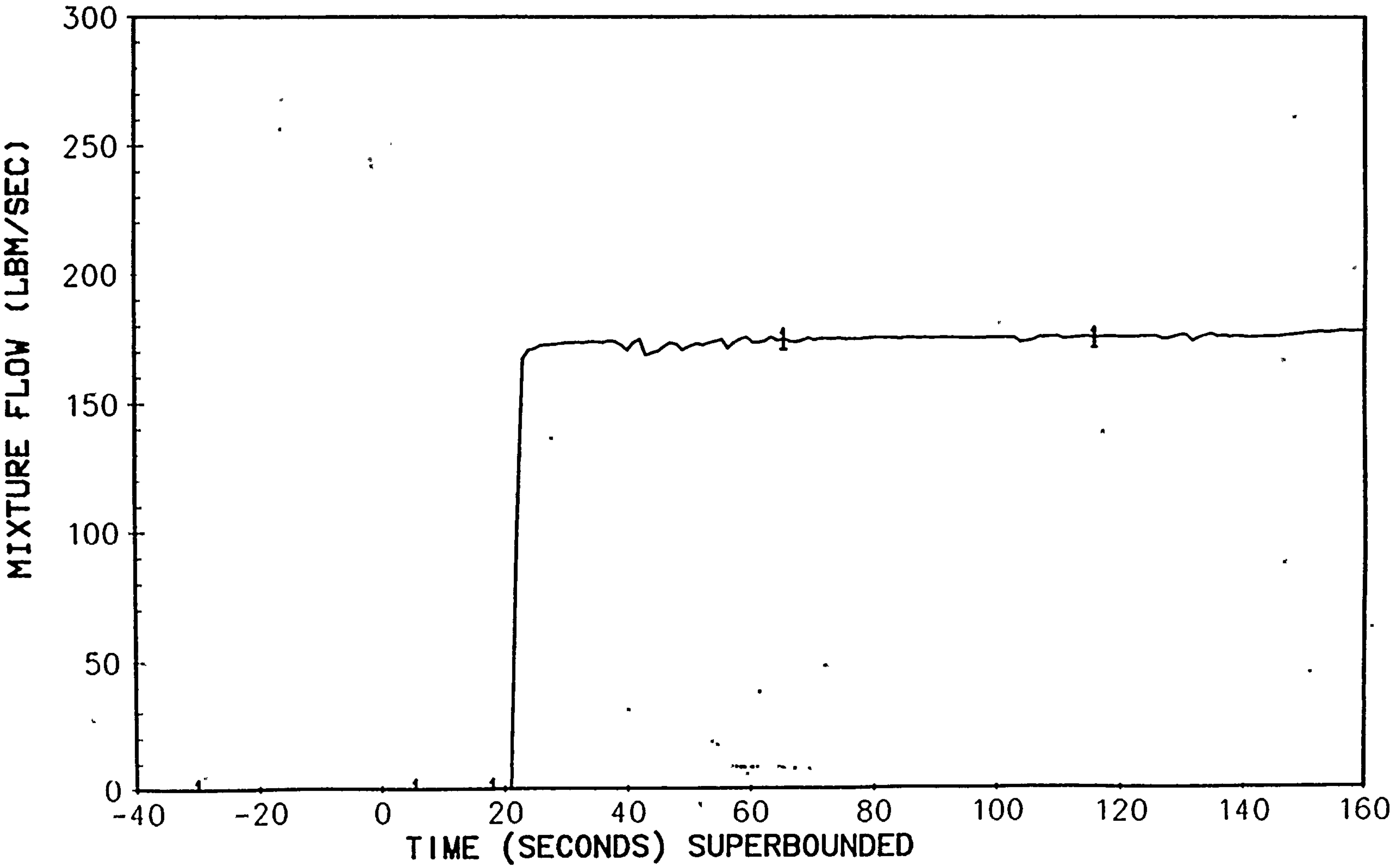


FIGURE 15.6.4.2-9A

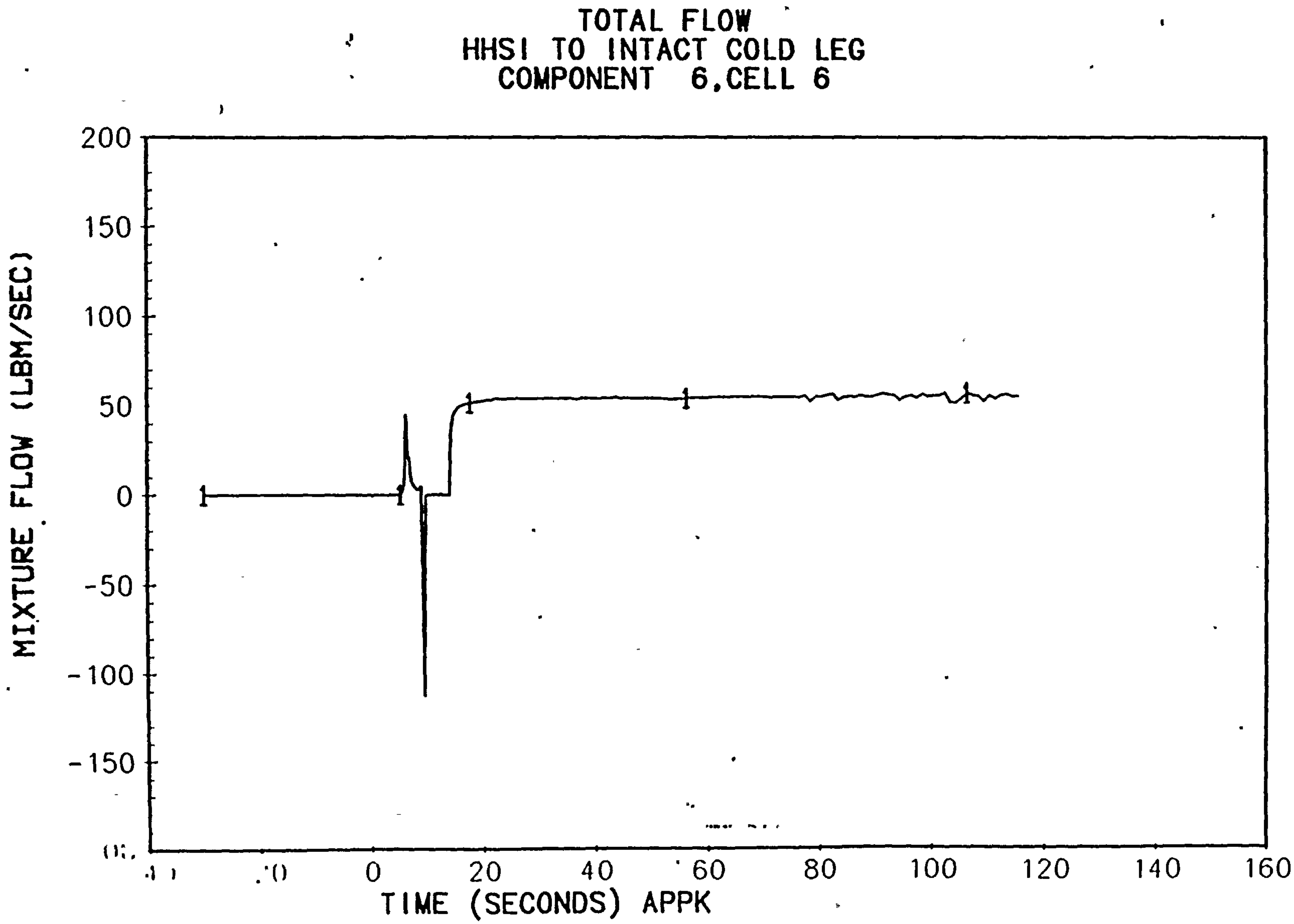


FIGURE 15.6.4.2-9B

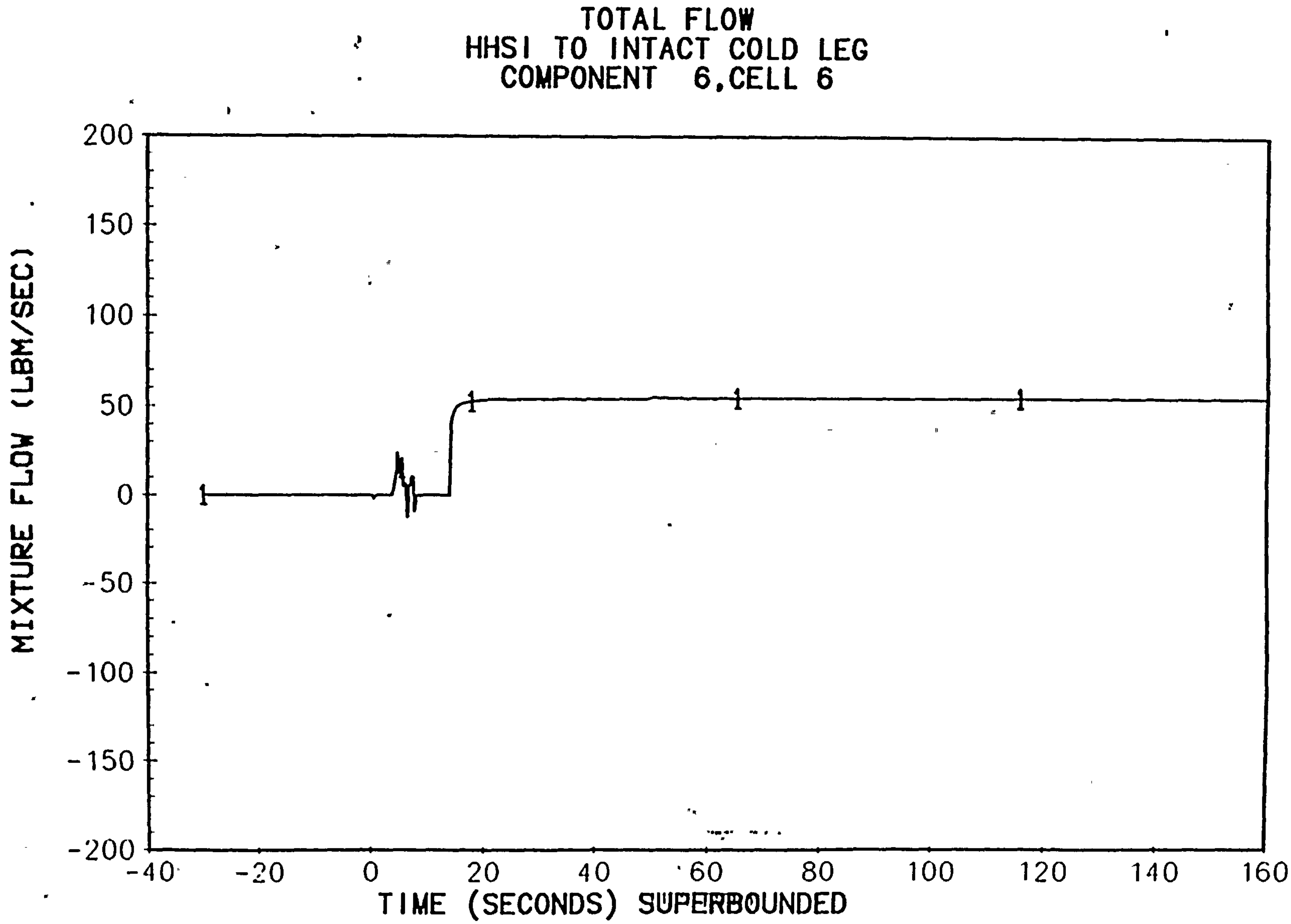


FIGURE 15.6.4.2-10A

TOTAL FLOW
ACCUMULATOR TO INTACT COLD LEG
COMPONENT 10,CELL 3

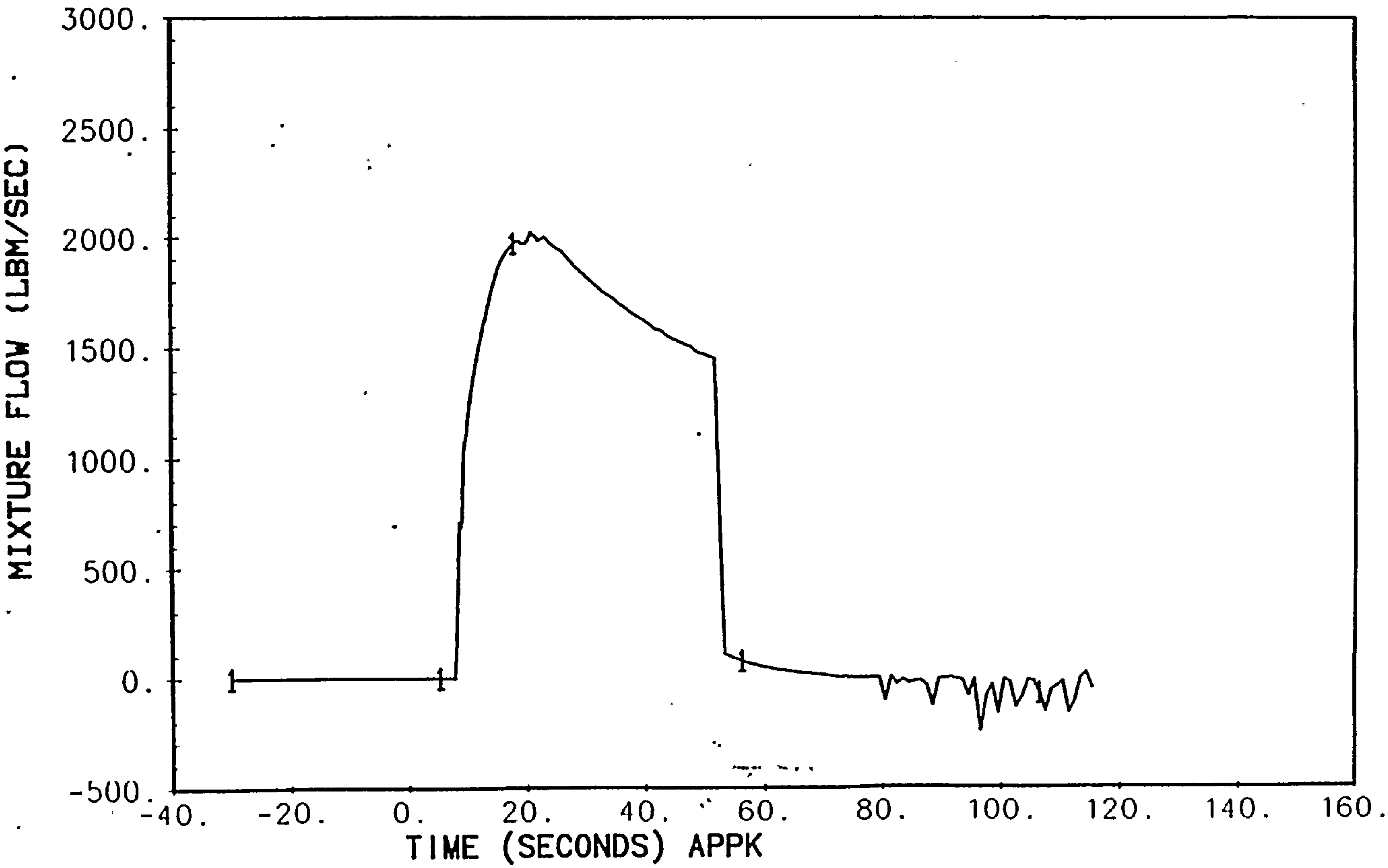


FIGURE 15.6.4.2-10B

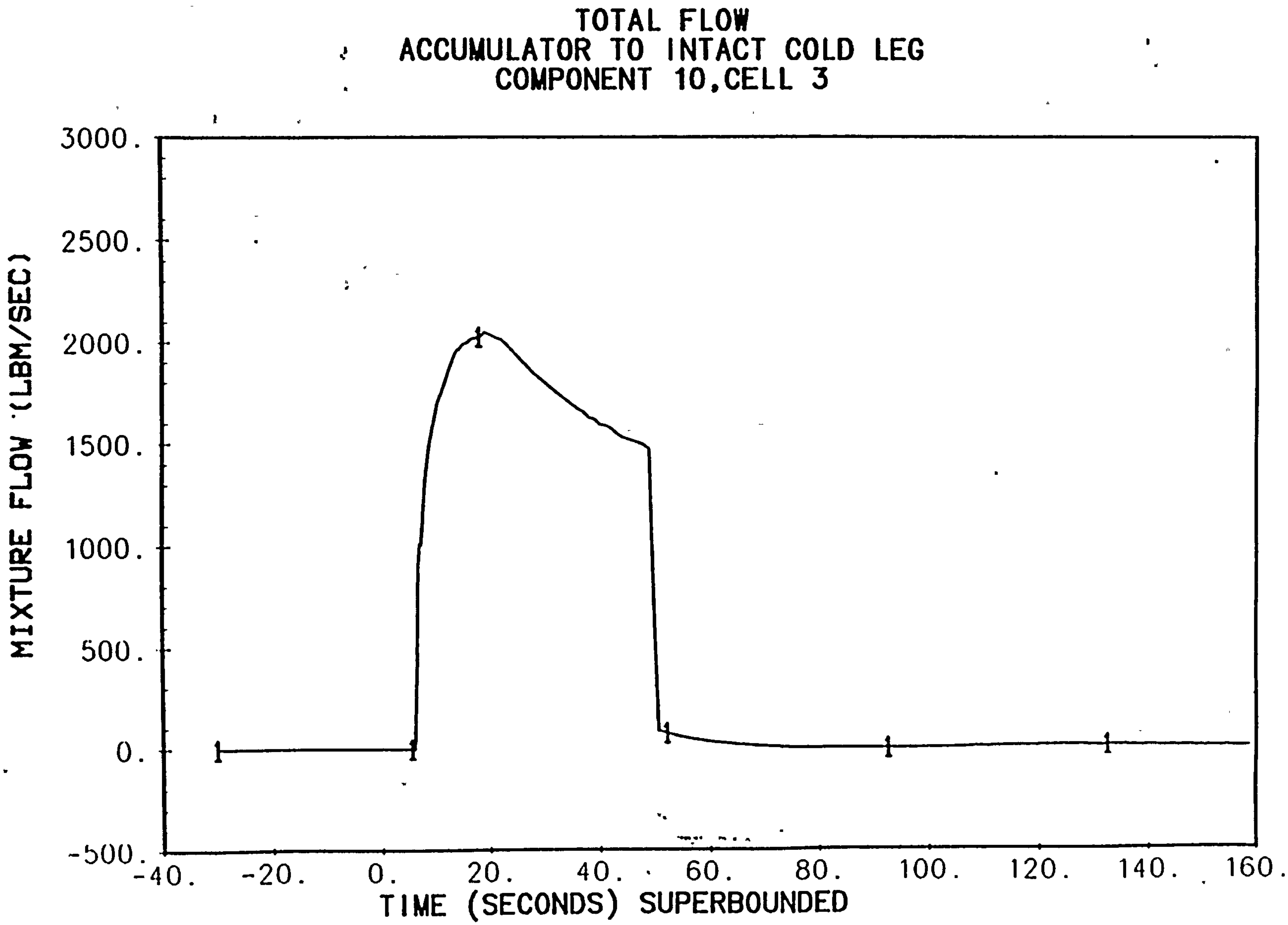


FIGURE 15.6.4.2-11A

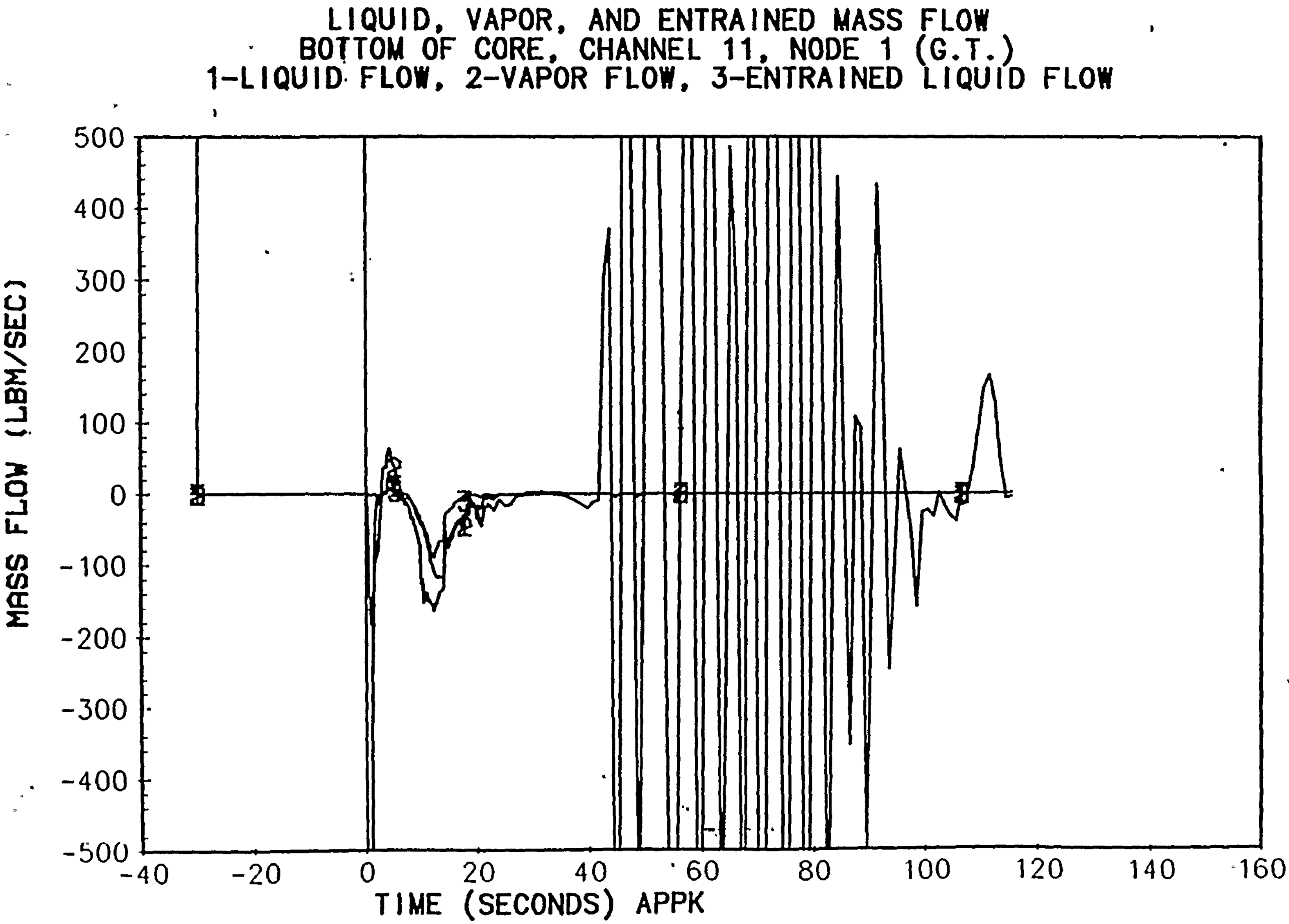


FIGURE 15.6.4.2-11B

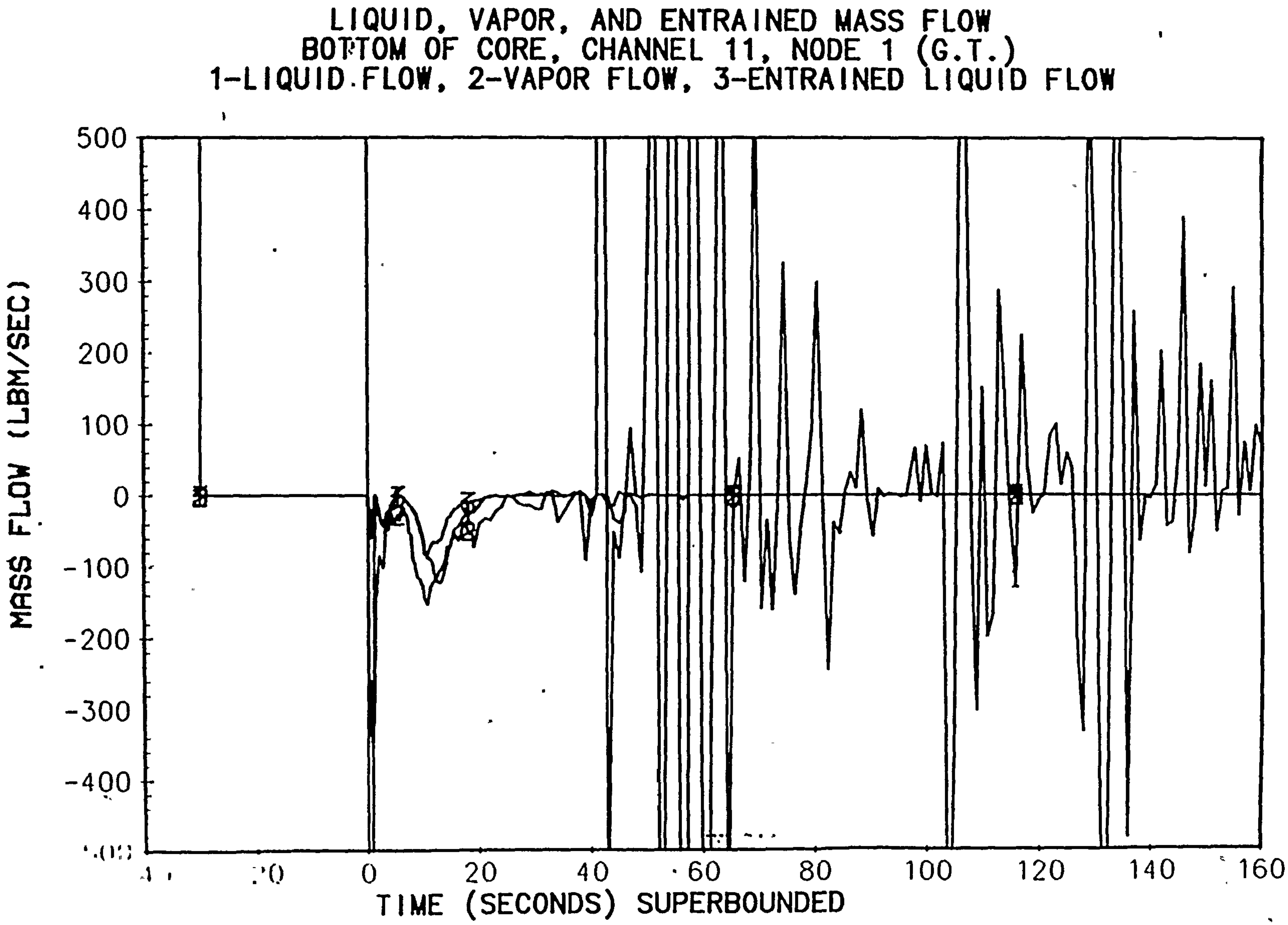


FIGURE 15.6.4.2-12A

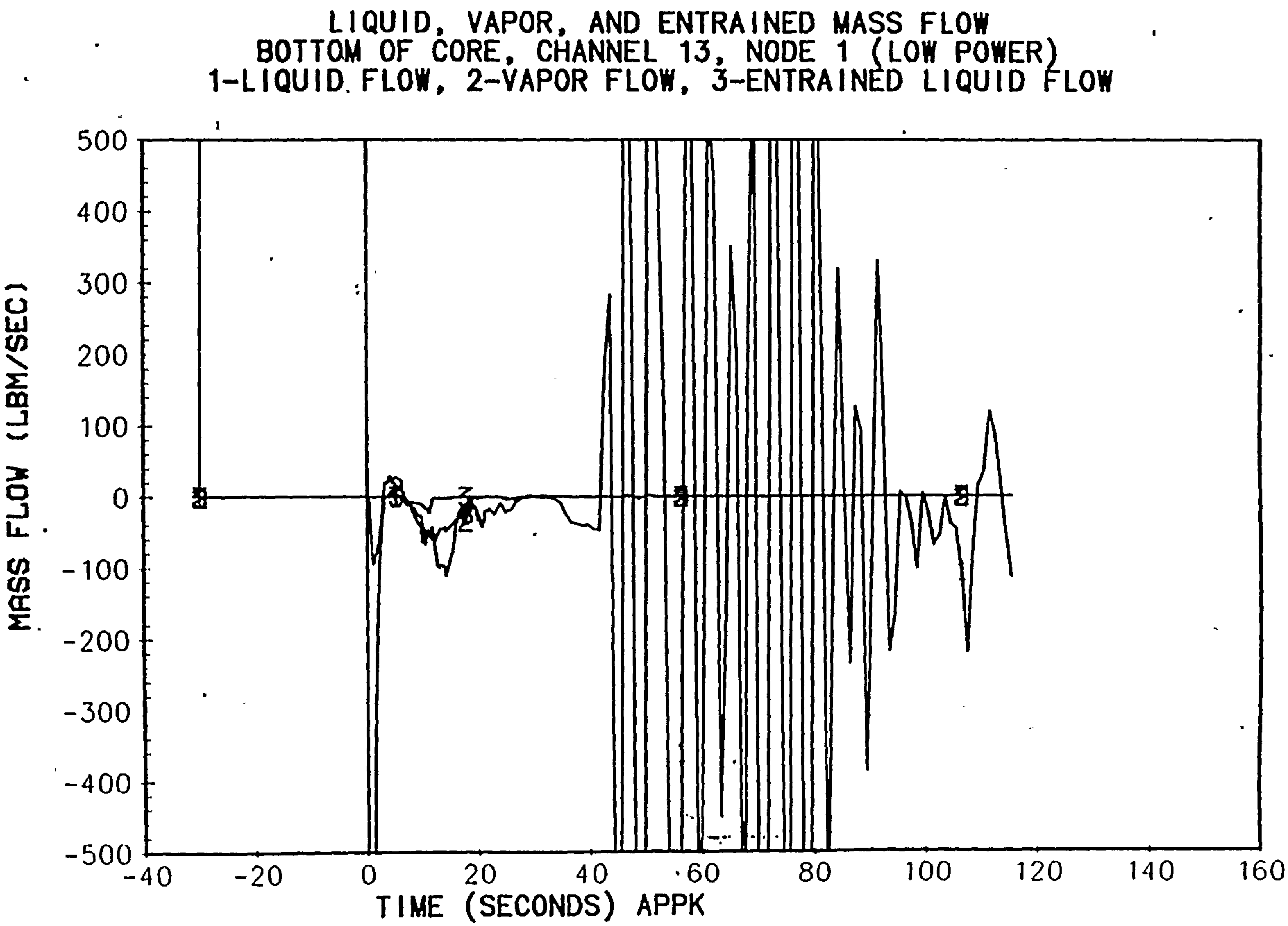


FIGURE 15.6.4.2-12B

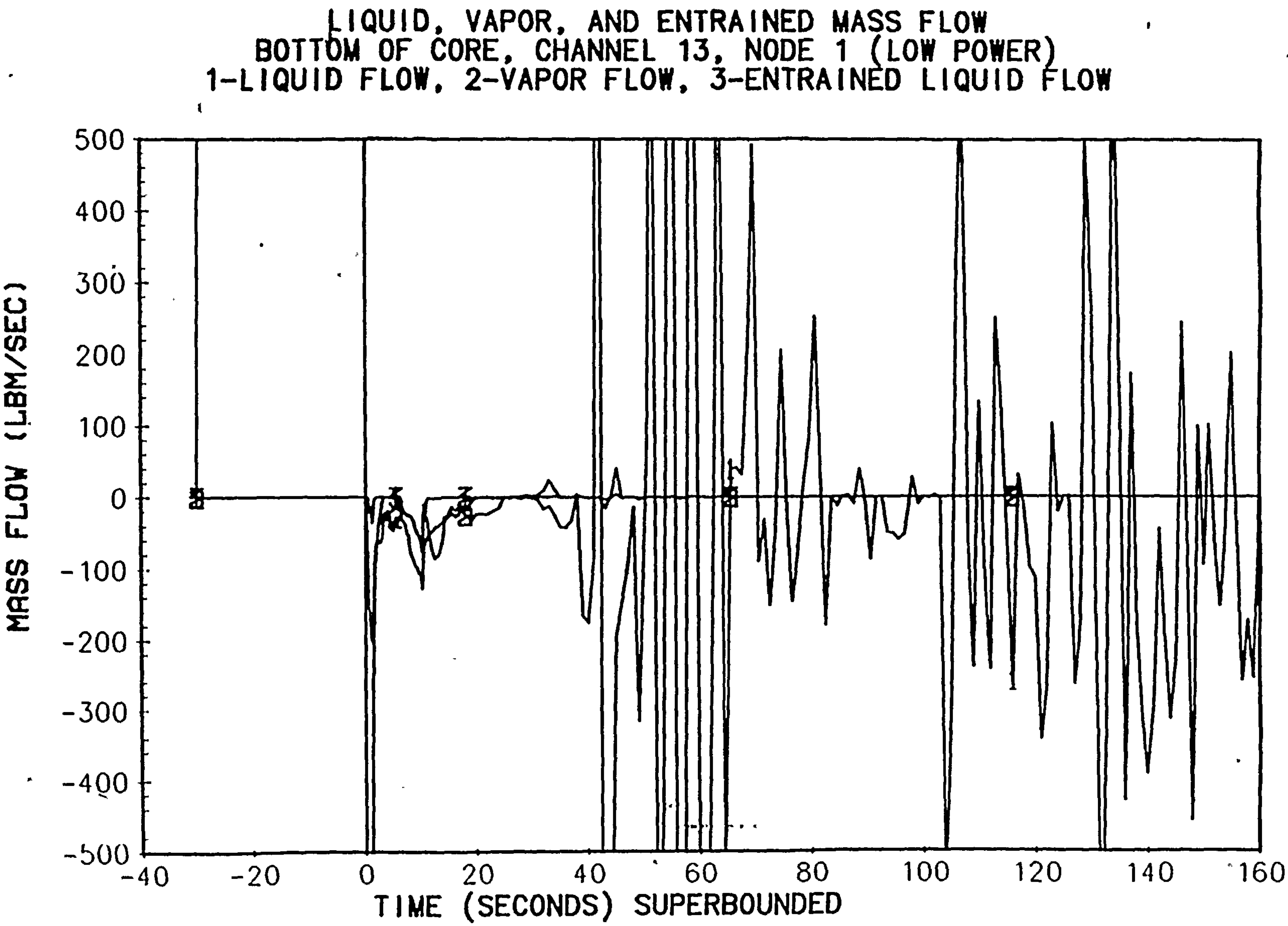


FIGURE 15.6.4.2-13A

LIQUID, VAPOR, AND ENTRAINED MASS FLOW
BOTTOM OF CORE, CHANNEL 12, NODE 1 (HOT ASSEMBLY)
1-LIQUID FLOW, 2-VAPOR FLOW, 3-ENTRAINED LIQUID FLOW

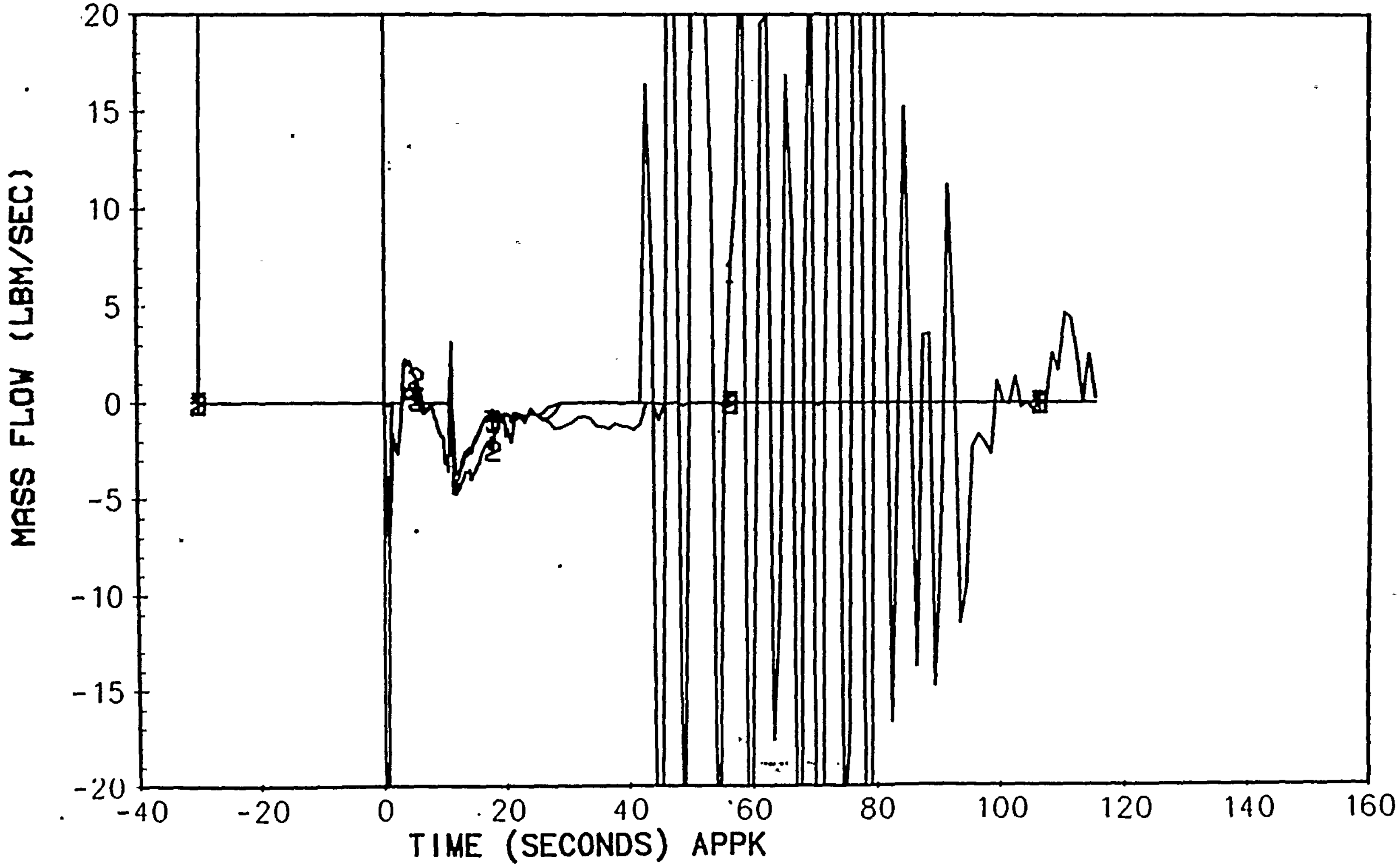


FIGURE 15.6.4.2-13B

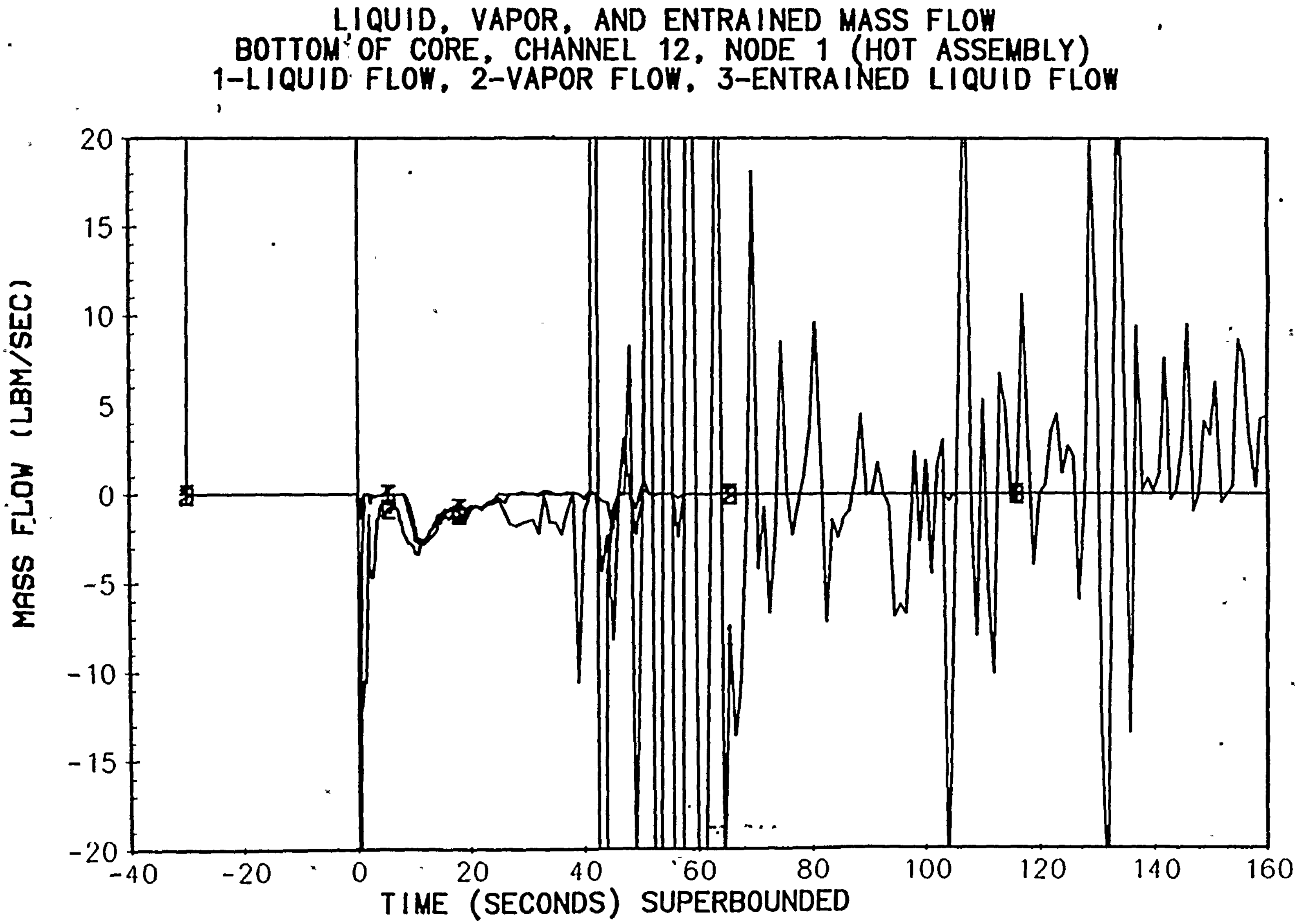


FIGURE 15.6.4.2-14

R. E. Ginna WCOBRA/TRAC Intact Loop Safety Injection Flow

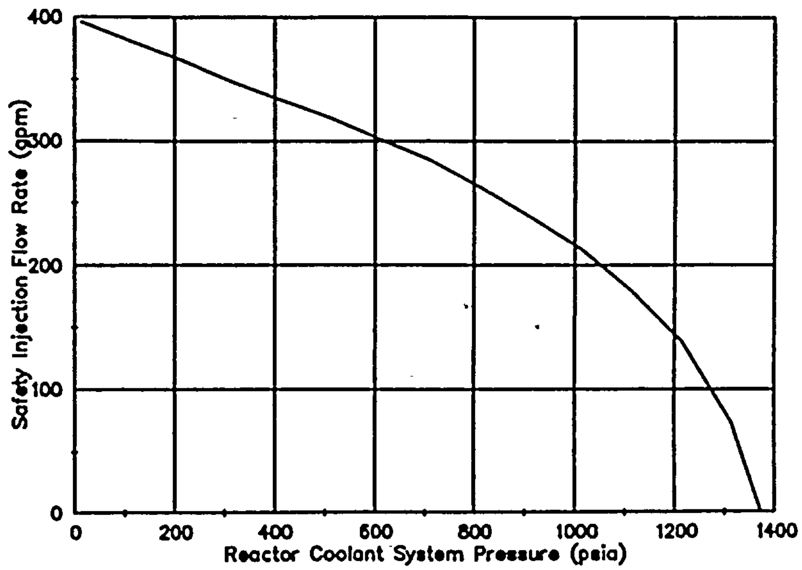


FIGURE A5.6.4.2-15

R. E. Ginna WCOBRA/TRAC Upper Plenum Safety Injection Flow