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SUBJECT: Forwards summary of ECCS analysis model reflecting upper plenum injection which satisfy acceptance criteria of 10CFR50.46.Analysis will be used to revise Kewaunee updated SAR.

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June 4, 1997

10CFR50.46

U.S. Nuclear Regulatory Commission
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A001

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Implementation of ECCS Analysis Model Reflecting Upper Plenum Injection

References:

1) Letter from M. L. Marchi (WPSC) to Document Control Desk dated July 23, 1996 (ECCS Analysis Update and Exemption Request)

2) "Westinghouse Large-Break LOCA Best Estimate Methodology, Volume 1: Model Description and Validation," WCAP-10924-P-A, Vol. 1, Revision 1, and Addenda 1, 2, & 3 (Proprietary), December 1988 (including Addendum 4, approved February 8, 1991)

3) "Westinghouse Large-Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWR's Equipped with Upper Plenum Injection, Addendum 1: Responses to NRC Questions," WCAP-10924-P-A, Revision 2 (Proprietary), WCAP-12071 (Non-Proprietary), December 1988

4) Letter from R. J. Laufer (NRC) to M. L. Marchi (WPSC) dated November 19, 1996 (Exemption)

5) Letter from M. L. Marchi (WPSC) to Document Control Desk dated October 25, 1996

6) Letter from R. J. Laufer (NRC) to M. L. Marchi (WPSC) dated December 3, 1996 (LBLOCA EM-Request for Additional Information)

7) Letter from M. L. Marchi (WPSC) to Document Control Desk dated December 20, 1996 (Licensee Event Report 96-010)

In Reference 1, Wisconsin Public Service Corporation (WPSC) provided the Nuclear Regulatory Commission (NRC) an update on the status of activities to implement a revised large break loss-of-coolant accident analysis for the Kewaunee Nuclear Plant. This analysis uses an Emergency Core Cooling System (ECCS) Evaluation Model (EM) that explicitly incorporates the effects of low pressure coolant injection to the upper plenum of the reactor vessel. Attachment 1 summarizes the completed analysis and will be used to revise the Kewaunee Updated Safety Analysis Report.

The analysis was completed using NRC previously approved methods as described in References 2 and 3. As detailed in Reference 1, to implement these methods WPSC was subject to three conditions: NRC approval of an exemption to 10CFR50 Appendix K, the performance of sensitivity studies to determine the highest peak clad temperature, and NRC approval of any decay heat model changes. An exemption to Appendix K requirements was issued by the NRC on November 19, 1996 (Reference 4); sensitivity studies to identify the highest peak clad temperature are discussed in Attachment 1; and the analysis used the previously approved decay heat model with no additional changes.

The results of the analysis as detailed in Reference 1 satisfy the acceptance criteria of 10CFR 50.46, that is:

1) peak fuel element cladding temperature does not exceed 2200 degrees Fahrenheit,

2) total oxidation of the cladding nowhere exceeds 17 percent of the total cladding thickness before oxidation,

3) total amount of hydrogen generated from cladding reaction does not exceed one percent of the amount that would be generated if all of the cladding reacted,

4) core geometry remains amenable to cooling, and

5) core temperature is maintained at an acceptably low value and decay heat is removed for an extended period of time required for the long-lived radioactivity remaining in the core.

Having satisfied the applicable conditions for use of the Westinghouse methodology and the regulatory requirements, WPSC will implement the new analysis as the analysis of record (AOR) with the startup of Cycle 22. This transmittal is provided to fulfill the requirements of 10CFR 50.46(a)(3)(ii). The peak clad temperature from the new analysis of record is 2009 degrees Fahrenheit.

With implementation of the Westinghouse LBLOCA evaluation model as the AOR, WPSC will no longer rely on the Siemens Power Corporation LBLOCA analysis, the previous AOR. This fulfills our commitments as described in References 5, 6, and 7 to have an acceptable LBLOCA ECCS evaluation completed prior to plant startup.

If you require any additional information or have questions concerning this submittal, please contact a member of my staff.

Sincerely,

Warren Steinhardt

for M. L. Marchi
Manager - Nuclear Business Group

RPP

Attach.

cc - US NRC, Region III
US NRC Senior Resident Inspector

50-305 KEWAUNEE WPSC

IMPLEMENTATION OF ECCS ANALYSIS MODEL THAT REFLECTS UPPER PLENUM INJECTION

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ATTACHMENT 1

Letter from M.L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 4, 1997

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

The analysis specified by 10 CFR 50.46 (Reference 1) "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors," is presented in this section. The results of this analysis show compliance with the requirements of Appendix K of 10 CFR 50, which are described in the topical report, "Westinghouse ECCS Evaluation Model - Summary" (Reference 2).

For the purpose of ECCS analyses, Westinghouse (W) defines a large break loss-of-coolant accident (LOCA) as a rupture 1.0 ft^2 or larger of the reactor coolant system piping including the double ended rupture of the largest pipe in the reactor coolant system or of any line connected to that system. The boundary considered for loss of coolant accidents as related to connecting piping is defined in Section 4.1.3.

Should a major break occur, rapid depressurization of the Reactor Coolant System (RCS) occurs in approximately 20 seconds to a pressure nearly equal to the containment pressure, with a nearly complete loss of system inventory. 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. However, since no credit is taken for rod insertion in the large break LOCA, the trip is not modeled in the analysis. A safety injection actuation signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. Although containment high pressure will actuate safety injection several tenths of a second before low pressurizer pressure, there is no way of modeling the containment high pressure setpoint in the WCOBRA/TRAC large break LOCA Evaluation Model. These actions will limit the consequences of the accident in two ways:

1) Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. The insertion of control rods to shut down the reactor is neglected in the large break LOCA analysis.

2) Injection of borated water provides for heat transfer from the core and prevents excessive cladding temperature.

Before the break occurs, the reactor 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 Kewaunee Nuclear Plant Large Break LOCA analysis using the WCOBRA/TRAC UPI methodology, the steam generator secondary is conservatively assumed to be isolated (main feedwater and steam line) at the initiation of the event to maximize the secondary side heat load.

When the RCS depressurizes to approximately 700 psig, 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. Appendix K to 10 CFR 50 (Reference 1) requires conservative calculation of the end of the bypass period, as well as expulsion of all ECCS computed to inject prior to the end of bypass. Bypass of the ECCS 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 2280 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 active fuel region 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 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 high head safety injection (HHSI) pump aids in the filling of the downcomer and core and subsequently supply water to help maintain a full downcomer and complete the reflooding process. The low head safety injection (LHSI), which injects into the upper plenum (hence, upper plenum injection - UPI) also aids the reflooding process by providing water to the core through the vessel upper plenum.

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 LHSI pumps (also called the Residual Heat Removal pumps, or RHR pumps) and returned to the upper plenum and RCS cold legs. Figure 14.3.2-1 contains a schematic of the bounding sequence of events for the Kewaunee large break LOCA transient.

Large Break LOCA Thermal Analysis

The reactor is designed to withstand thermal effects caused by a loss-of-coolant accident including the double-ended severance of the largest reactor coolant system pipe. The reactor core and internals together with the Emergency Core Cooling System are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident. The Emergency Core Cooling System, even when operating during the injection mode with the most severe single active failure, is designed to meet the Acceptance Criteria (Reference 1).

The large break LOCA ECCS analysis applicable to Kewaunee with a full core of Siemens-Designed 14x14 Standard fuel was performed with the Westinghouse WCOBRA/TRAC Two-Loop Upper Plenum Injection (UPI) Large Break LOCA Evaluation Model (References 3 and 4). The evaluation model and the major input parameters and initial conditions are described more fully in the following sections.

Large Break LOCA Evaluation Model

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 (References 3 and 4). The Westinghouse Best-Estimate Methodology was developed consistent with guidelines set forth in the SECY-83-472 document (Reference 5). 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 3 and 4.

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 Superbounded 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 (Reference 4), 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 (Reference 3).

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. 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:

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

The nominal calculation is performed to provide assurance that the most probable PCT is 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 is used to demonstrate compliance with the Acceptance Criteria of 10 CFR 50.46, provided that the peak cladding temperature exceeds the peak cladding temperature calculated at the 95 percent probability level but is below the Acceptance Criteria limit of 2200°F.

The Best Estimate UPI ECCS Evaluation Model is comprised of the WCOBRA/TRAC and COCO computer codes (References 3, 4 and 6). 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 (Reference 7) 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 (Reference 8). 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 (Reference 4). 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.

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 (Reference 6) 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 high head and low head 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 HHSI and LHSI injection. Additionally, since the WCOBRA/TRAC and COCO computer codes do not run interactively, it would be difficult to model HHSI and LHSI actuation on high containment pressure.

Large Break LOCA Input Parameters and Initial Conditions

Important input parameters and initial conditions used in the analysis are listed in Table 14.3.2-1. The assumptions made in the Appendix K, Superbounded and Nominal cases respectively are listed in Tables 14.3.2-2 through 14.3.2-4. The initial steady state fuel pellet temperature and fuel rod internal pressure used in the LOCA analysis were generated with the PAD 3.4 Fuel Rod Design Code (Reference 9) which has been approved by the US-NRC. The fuel parameters input to the code were at beginning-of-life (maximum densification) values for the hot assembly and hot rod, and the remainder of the assemblies were modeled at a conservative value representative of average core burnup.

In determining the conservative direction for Superbounded values and assumptions for UPI plants, many sensitivity studies were performed (Reference 4). 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 higher core power to pumped ECCS flow ratio than the Kewaunee Nuclear Power 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 Kewaunee analysis.

The parameters used to determine the containment pressure curve are presented in Tables 14.3.2-5 through 14.3.2-8. Table 14.3.2-5 contains the containment data, Table 14.3.2-6 tabulates the fan cooler performance curve, and Table 14.3.2-7 lists the structural heat sinks in the containment. Table 14.3.2-8 contains the mass and energy releases to the containment for the Appendix K calculation. These releases and other containment data are then used to generate a containment pressure transient. This pressure transient is compared with the pressure transient originally used in the system hydraulic transient, and the more conservative pressure transient is used. The containment pressure transient used to calculate the system hydraulic transient is shown in Figure 14.3.2-2 for the Superbounded and

Appendix K calculations. The Nominal calculation uses a Best-Estimate containment pressure transient, which is 10 psi higher than the Appendix K and Superbounded containment pressure transient, consistent with the methodology in WCAP-10924 (Refences 3 and 4).

Initial conditions for the Kewaunee large break LOCA analysis are delineated in Table 14.3.2-1. 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 selected as an interior assembly under a support column surrounded by non-guide tube assemblies. Past sensitivity studies have demonstrated the limiting location for peak cladding temperature to be the limiting flow area at the upper core plate (Reference 4), and a support column has the limiting flow area among the various upper core plate geometries in Kewaunee.

Large Break LOCA Analysis Results

Three transient calculations are performed with WCOBRA/TRAC for Kewaunee with a full core of Siemens 14x14 Standard fuel. The first is an Appendix K calculation that demonstrates the compliance with the specific requirements of Appendix K to 10 CFR 50. The second is the Superbounded calculation as per the approach defined in SECY-83-472. This calculation is performed at the 95-percent probability level and accounts for uncertainties in such things as power level, fuel initial temperature and computer code uncertainties. The third calculation is a Nominal calculation, which is performed at the 50 percent probability level, and is the most realistic.

Appendix K Calculation Results

An Appendix K calculation was performed for the Kewaunee 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 14.3.2-2. Table 14.3.2-9 shows the time sequence of events for the Appendix K calculation, and Table 14.3.2-10 provides a brief summary of the important results. Figure 14.3.2-3 contains the power shape used in the Appendix K calculation.

Shortly after the break is assumed to open, the vessel rapidly depressurizes (Figure 14.3.2-5) and the core flow quickly reverses. During the flow reversal, the hot assembly fuel rods dry out and begin to heat up momentarily (Figure 14.3.2-4).

At approximately 10 seconds into the transient, maximum blowdown downflow is reached in the high and low power regions of the core. Figures 14.3.2-12 through 14.3.2-19 show the liquid flow (which is the sum of the liquid and entrained liquid flows) and the vapor flow at the top of the core assemblies[1]. This flow is sufficient to cool the hot assembly and average rods, and maintain the low power / periphery region of the core near the fluid saturation temperature (Figure 14.3.2-4).

As the vessel continues to depressurize, liquid inventory continues to be depleted (Figure 14.3.2-6). This results in reduced core flow, first in the hot assembly and soon after in the other assemblies (Figures 14.3.2-12 through 14.3.2-19) and resulting cladding heatup, also first in the hot assembly, and later in the other regions of the core (Figure 14.3.2-4).

At approximately 7.9 seconds into the transient, the accumulator begins to inject water into the intact cold leg (Figure 14.3.2-9). This water fills the cold leg and upper downcomer region (Figure 14.3.2-20), and is bypassed to the break initially. The end of bypass period occurs when flow in the core reverses for the last time during blowdown. The end of blowdown occurs at 20.19 seconds, and the downcomer and lower plenum begin to refill shortly thereafter (Figures 14.3.2-20 and 14.3.2-21).

At about 22 seconds, pumped high head safety injection (HHSI) into the cold leg also begins (Figure 14.3.2-10). The pumped low head safety injection (LHSI) into the upper plenum begins at approximately 27 seconds (Figure 14.3.2-11). Water

1 The core channels are defined as follows: Channel 10 is the "Open Hole / Support Column" Channel (OH/SC), which contains the 63 fuel assemblies located below open holes or support columns in the average power region of the core. Channel 11 is the "Guide Tube" Channel (GT), which contains the 33 fuel assemblies located below guide tubes in the average power region of the core. Channel 12 is the "Hot Assembly" Channel (HA), which contains the hottest assembly in the core (one rod of which is the "Hot Rod"), and is modeled as the limiting fuel assembly, a support column. Channel 13 is the "Low Power Region" Channel (LP), which contains the 24 fuel assemblies located in the periphery of the core, that are at a lower power than the rest of the core.

from the LHSI flows down the low power / periphery region of the core (Figure 14.3.2-18) and cross-flows into the other core regions, providing some limited core cooling.

At approximately 28 seconds, the pressure in the vessel has dropped to near containment pressure (Figure 14.3.2-5), so accumulator and HHSI water begins to flow into the lower plenum from the downcomer (Figures 14.3.2-20 and 14.3.2-21), and the downcomer continues to refill. The LHSI flows down the low power / periphery region of the core and also fills the lower plenum. The inventory of the vessel begins to increase quickly as the lower plenum refills and break flow decreases to less than the HHSI and LHSI flows (Figures 14.3.2-6 through 14.3.2-11).

The bottom-of-core recovery time occurs at 27.91 seconds. The lack of cooling in the core causes the core temperatures to continue to rise, and the hot rod bursts due to high temperature creep at 32.88 seconds. The hot assembly also bursts at 48.66 seconds, which is just before the core begins to refill.

The accumulators empty at 34.35 seconds. At this time, the accumulator nitrogen injects into the cold leg and forces water down the downcomer and up through the core (Figures 14.3.2-12 through 14.3.2-19). The added flow causes the temperature in the assemblies to substantially decrease for a brief period (Figure 14.3.2-4). However, the increased flow through the core caused by the nitrogen injection forces water out the break (Figure 14.3.2-8), and also stops the downflow in the low power / periphery region of the core (Figure 14.3.2-18). The temperature in the core assemblies therefore increases due to the lack of cross-flow from the low power / periphery region, and also the reduced vessel water inventory. As the nitrogen injection slows, the downflow in the low power / periphery region increases, and the ECCS water begins to refill the vessel once again (Figure 14.3.2-6).

At approximately 45 seconds, the lower plenum has filled to the point that water begins to reflood the core from below (Figures 14.3.2-21 and 14.3.2-22). The void fractions in the lower plenum and the bottom of the core (Figures 14.3.2-23 and 14.3.2-24) are reduced to nearly zero as the core refills.

At approximately 75 seconds, sufficient cooling is provided by the safety injection system to begin cooling the core (Figure 14.3.2-4). However, at approximately 100 seconds, downcomer boiling begins to occur due to the stored energy in the vessel wall metal, and the water levels in the downcomer decrease (Figure 14.3.2-20) and the void fractions in the core channels increase (Figure 14.3.2-24).

Also, boiling occurs in the lower plenum, beginning at approximately 180 seconds (Figures 14.3.2-21 and 14.3.2-23). The cooling in the hot assembly decreases due to the reduced flow, and the peak clad temperature increases (Figure 14.3.2-4).

At approximately 180 seconds, the downcomer begins to refill (Figure 14.3.2-20). At 185 seconds, the lower plenum begins to refill (Figure 14.3.2-21). and void fractions in the lower plenum and core decrease (Figures 14.3.2-23 and 14.3.2-24). The vessel inventory also increases to an equilibrium state (Figure 14.3.2-6). The amount of liquid in the core slowly increases (Figure 14.3.2-22), providing cooling to the hot assembly and average assemblies, and the clad temperatures of all of the assemblies begin to decrease (Figure 14.3.2-4). Sufficient cooling is provided by the ECCS to continue cooling the core.

The peak cladding temperature calculated for the Kewaunee Nuclear Plant Appendix K large break LOCA analysis is 2009°F, assuming a peak hot rod power of 14.409 kw/ft (F_Q = 2.280). This temperature occurs during reflood at approximately 200 seconds. This result is below the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 4.11 percent, which is below the embrittlement Acceptance Criteria limit of 17 percent. The limiting total core metal-water reaction is 0.005 percent, which is much less than the 1.0 percent limit, 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.

Superbounded Calculation Results

A Superbounded calculation was performed at the 95 percent probability level for the Kewaunee Nuclear Plant as per the SECY-83-472 approach. The assumptions used in the Superbounded calculation are listed in Table 14.3.2-3. Table 14.3.2-9 shows the time sequence of events for the Superbounded calculation, and Table 14.3.2-10 provides a brief summary of the important results.

The Superbounded transient is similar to the Appendix K transient described above. The same figures have been provided as in the Appendix K results.

Most of the differences are timing issues. In general, the events occur earlier in the Superbounded calculation, as can be seen in Table 14.3.2-9, since the break discharge coefficient, C_D, is larger in the Superbounded calculation than in the Appendix K calculation, i.e., 0.6 versus 0.4. Therefore, the blowdown peak clad temperature occurs earlier in the Superbounded calculation, as does the end of the blowdown portion of the transient.

The vessel depressurizes more rapidly, allowing the accumulator and HHSI and LHSI flows to begin earlier, and the core begins to refill earlier. The accumulators inject faster since the system pressure is lower, and therefore are empty sooner than in the Appendix K calculation.

Nitrogen injection is not modeled in the Superbounded calculation, so the dip in peak clad temperature due to the nitrogen in the Appendix K calculation is not seen in the Superbounded calculation, and the cladding continues to heat up (Figure 14.3.2-25).

Downcomer and lower plenum boiling occurs at approximately 140 seconds in the Superbounded transient (Figures 14.3.2-41, 14.3.2-42 and 14.3.2-44). However, the low power and average power assemblies are all already quenched, and sufficient cooling is provided to the hot assembly, such that the hot assembly clad temperature continues to decrease (Figure 14.3.2-25).

At approximately 160 seconds, the downcomer begins to refill (Figure 14.3.2-41), water levels in the core channels increase (Figure 14.3.2-43), and void fractions in the lower plenum and core channels decrease (Figures 14.3.2-44 and 14.3.2-45). The downflow in the low power region increases and becomes more stable and continuous (Figure 14.3.2-39), which provides sufficient cooling to quench the hot assembly (Figure 14.3.2-25).

The peak cladding temperature calculated for the Kewaunee Nuclear Plant Superbounded large break LOCA analysis is 1827°F, assuming a peak hot rod power of 14.409 kw/ft (which is equivalent to an Appendix K peak hot rod power of 15.85 kw/ft). This temperature occurs during blowdown at approximately 6.5 seconds, and includes 260.5°F added for uncertainties (Reference 3, Addendum 4). This result is below the Appendix K result of 2009°F, as required by SECY-83-472 (Reference 5). The maximum local metal-water reaction is less than 17 percent, and the limiting total core metal-water reaction is less than the 1.0 percent limit, in accordance with the Acceptance Criteria. By the end of the transient, the core temperature has dropped to the saturation temperature, and all of the rods modeled have quenched. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. Therefore, the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Nominal Calculation Results

A Nominal calculation was performed at the 50 percent probability level for the Kewaunee Nuclear Plant as per the SECY-83-472 approach. The assumptions used in the Nominal calculation are listed in Table 14.3.2-4. Table 14.3.2-9 shows the time sequence of events for the Nominal calculation, and Table 14.3.2-10 provides a brief summary of the important results.

The Nominal transient is similar to the Appendix K transient described above. The same figures have been provided as in the Appendix K results.

Most of the differences are timing issues. The events occur earlier in the Nominal calculation, as can be seen in Table 14.3.2-9, since the Nominal calculation has a more realistic break model than the Moody pipe model used in the Appendix K transient. Therefore, the vessel depressurizes more rapidly, allowing the ECCS flows to begin earlier, the blowdown portion of the transient to end earlier, and the core to begin refilling earlier. The accumulators inject faster since the system pressure is lower due to the more realistic break model, and therefore are empty sooner than in the Appendix K calculation.

The Nominal calculation models nitrogen injection at 32.80 seconds. However, the effect of the nitrogen flow on the peak clad temperature (Figure 14.3.2-46) is less than the effect seen in the Appendix K calculation.

The downcomer and lower plenum boiling begins at approximately 120 seconds. However, the effect of the boiling on the clad temperatures is minimal in the Nominal calculation, since the hot assembly clad temperature is very low, and the other rods remain nearly quenched for the entire transient.

The peak cladding temperature calculated for the Kewaunee Nuclear Plant Nominal large break LOCA analysis is 1525°F, which occurs during reflood at approximately 178 seconds. This result is considerably lower than the Appendix K and Superbounded peak clad temperatures, and is indicative of the margin available in a Best Estimate calculation. The maximum local metal-water reaction is less than 17 percent, and the limiting total core metal-water reaction is less than the 1.0 percent limit, in accordance with the Acceptance Criteria. By the end of the transient, the entire core except the hot assembly has quenched. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the hot assembly 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.

Large Break LOCA Analysis Conclusions

This large break LOCA analysis has determined that Kewaunee operation with Siemens-Designed 14x14 Standard fuel is acceptable. The large break LOCA analysis presented in this section shows that the high head and low head safety injection, together with the accumulators, provide sufficient core flooding to meet the 10 CFR 50.46 Acceptance Criteria. That is:

1) The calculated peak fuel element cladding temperature does not exceed 2200°F.

2) The calculated total oxidation of the cladding nowhere exceeds 17 percent of the thickness before oxidation.

3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1.0 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react.

4) The core remains amenable to cooling.

5) The core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required for the long-lived radioactivity remaining in the core.

References

1. 10 CFR Part 50.46 and Appendix K of 10 CFR, Part 50, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," Federal Register, Volume 39, Number 3 , January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339 (Non-Proprietary), July 1974.

3. Hochreiter, L. E., Schwarz, W. R., Takeuchi, K., Tsai, C. K., and Young, M. Y., Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation, WCAP-10924-P-A, Vol. 1, Rev. 1, and Addenda, (Proprietary Version), December 1988 (including Addendum 4, approved February 8, 1991).

4. Dederer, S. I., Hochreiter, L. E., Schwarz, W. R., Stucker, D. L., Tsai, C-K., and Young, M. Y., Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection, WCAP-10924-P-A, Vol. 2, Rev. 2, and Addenda, December 1988.

5. NRC Staff Report, Emergency Core Cooling System Analysis Methods, USNRC-SECY-83-472, November 1983.

6. Bordelon, F. M., and Murphy, E. T., Containment Pressure Analysis Code (COCO), WCAP-8327 (Proprietary Version), WCAP-8326 (Non-Proprietary Version), June 1974.

7. Thurgood, M.J., Kelly, J.M., Guidotti, T.E., Kohrt, R.J., Crowell, K.R., COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems: Equations and Constitutive Models, NUREG/CR-3046, PNL-4385 Vol. 1 R4, March 1983.

8. Ishii, M., Thermo-Field Dynamic Theory of Two-Phase Flow, Eyrolles, 1975.

9. Weiner, R. A., et al., Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, WCAP-10851-P-A (Proprietary Version), August 1988.

Table 14.3.2-1 (Page 1 of 2)

Initial Conditions for the Kewaunee Nuclear Plant WCOBRA/TRAC Large Break LOCA Analysis

Parameter	Analysis Value
Plant Internals	Flat Upper Support Plate
Barrel Baffle Design	Downflow
Core Bypass Flow (%)	6.92
Licensed Core Power (MWt)	102% of 1650
System Pressure with Uncertainties (psia)	2280.0
Primary System Fluid Temperatures: T_{HOT} (°F) T_{AVG} (°F) T_{COLD} (°F)	 596.69 562.00 527.31
Fuel Type	Siemens 14x14 Standard Fuel
Fuel Data Source	PAD 3.4 [Ref 9]
Rod Backfill Pressure (psig)	290.0
Total Peaking Factor, F_Q	2.280
Nuclear Enthalpy Rise Peaking Factor, $F_{\Delta H}$	1.550
Peak Linear Power (kw/ft): Appendix K Superbounded Nominal	 14.409 14.409 13.080
Relative Power in the Outer Core Channel	0.45

Table 14.3.2-1 (Page 2 of 2)

Initial Conditions for the Kewaunee Nuclear Plant WCOBRA/TRAC Large Break LOCA Analysis

Parameter	Analysis Value
Loop Volumetric Flow Rate (gpm) Loop Mass Flow Rate (lbm/sec)	83,400 8969
Reactor Coolant Pumps	Running
Steam Generator Tube Plugging (%)	30.0
Steam Generator Secondary Pressure (psia)	642.32
Accumulators in Operation	2 (one injects into the Intact Loop; one spills to containment)
Accumulator Conditions per Accumulator: Water Volume (ft^3) Nitrogen Pressure (psia) Water Temperature (°F)	 1250.0 714.7 90.0 in Appendix K and Nominal; 108.0 in Superbounded
Number of Safety Injection Pumps in Operation	1 LHSI injecting into the Upper Plenum; 1 HHSI injecting, spilling to containment pressure 1 LHSI and 1 HHSI pump assumed to fail
Safety Injection Conditions: Pump Flow Water Temperature (°F) HHSI Delay Time (sec) LHSI Delay Time (sec)	 Degraded 10% for HHSI; Degraded 5% for LHSI 70.0 15.0 after S-signal (offsite power available) 20.0 after S-signal (offsite power available)
Containment Pressure	See Figure 14.3.2-2

Table 14.3.2-2

Assumptions Used in the Appendix K Calculation

Plant Configuration	Value
Pressurizer Location	Intact Loop
Total Peaking Factor, F_Q	2.280
Nuclear Enthalpy Rise Peaking Factor, $F_{\Delta H}$	1.550
Core Power (MWt)	102% of 1650.0
Loop Flow Rate	Thermal Design Minimum Flow
Hot Assembly Burnup	Beginning of Cycle
Modeling Assumptions	**Value**
ECCS Worst Single Failure	Loss of a LHSI Pump
ECCS Spilling Assumption	One HHSI Pump injecting, spilling to containment pressure
Accumulator Nitrogen	Modeled
Reactor Coolant Pump Model	Conservative Two-Phase Model
Pump Rotor	Not Locked During Reflood
Cross-Flow De-Entrainment	Modeled
Limiting Break Discharge Coefficient, C_D	0.4 [References 3 and 4]
Containment Pressure	Lower Bound
Decay Heat Model	ANS 1971 Decay Heat + 20%
Metal-Water Reaction Model	Baker-Just
Swelling and Blockage	Modeled
Clad Burst	Modeled
Nucleate Boiling During Blowdown	Not Allowed
ECCS Bypass Model	Conservative

Table 14.3.2-3

Assumptions Used in the Superbounded Calculation

Plant Configuration	Value
Pressurizer Location	Intact Loop
Total Peaking Factor, F_Q	2.280 (equiv. to Appendix K F_Q of 2.508)
Nuclear Enthalpy Rise Peaking Factor, $F_{\Delta H}$	1.550
Core Power (MWt)	102% of 1650.0
Loop Flow Rate	Thermal Design Minimum Flow
Hot Assembly Burnup	Beginning of Cycle
Modeling Assumptions	**Value**
ECCS Worst Single Failure	Loss of a LHSI Pump
ECCS Spilling Assumption	One HHSI Pump injecting, spilling to containment pressure
Accumulator Nitrogen	Not Modeled
Reactor Coolant Pump Model	Conservative Two-Phase Model
Pump Rotor	Locked During Reflood
Cross-Flow De-Entrainment	Not Modeled
Limiting Break Discharge Coefficient, C_D	0.6 [References 3 and 4]
Containment Pressure	Lower Bound
Decay Heat Model	1979 ANS Decay Heat with 2-σ Unc'y
Metal-Water Reaction Model	Cathcart-Pawel with 2-σ Unc'y
Swelling and Blockage	Not Modeled
Clad Burst	Not Modeled
Nucleate Boiling During Blowdown	Allowed
ECCS Bypass Model	Realistic

Table 14.3.2-4

Assumptions Used in the Nominal Calculation

Plant Configuration	**Value**
Pressurizer Location	Intact Loop
Total Peaking Factor, F_Q	2.111 (equiv. to Appendix K F_Q of 2.280)
Nuclear Enthalpy Rise Peaking Factor, $F_{\Delta H}$	1.435 (equiv. to Appendix K $F_{\Delta H}$ of 1.550)
Core Power (MWt)	100% of 1650.0
Loop Flow Rate	Thermal Design Minimum Flow
Hot Assembly Burnup	Beginning of Cycle
Modeling Assumptions	**Value**
ECCS Worst Single Failure	Loss of a LHSI Pump
ECCS Spilling Assumption	One HHSI Pump injecting, spilling to containment pressure
Accumulator Nitrogen	Modeled
Reactor Coolant Pump Model	Conservative Two-Phase Model
Pump Rotor	Not Locked During Reflood
Cross-Flow De-Entrainment	Modeled
Limiting Break Discharge Coefficient, C_D	0.4 [References 3 and 4]
Containment Pressure	Nominal
Decay Heat Model	1979 ANS Decay Heat w/o 2-σ Unc'y
Metal-Water Reaction Model	Cathcart-Pawel w/o 2-σ Unc'y
Swelling and Blockage	Not Modeled
Clad Burst	Not Modeled
Nucleate Boiling During Blowdown	Allowed
ECCS Bypass Model	Realistic

Table 14.3.2-5

Kewaunee Large Break LOCA Containment Data

(Note that this data is used for minimizing the containment pressure)

Plant Configuration	Analysis Value
Net Free Volume (ft^3)	1.37×10^6
Initial Conditions:	
Pressure (psia)	14.7
Temperature (°F)	90.0
RWST Temperature (°F)	45.0
Service Water Temperature (°F)	32.0
Containment Exterior Shell Temperature (°F)	-20.0
Spray System:	
Number of Pumps Operating	2
Runout Flow Rate, each (gpm)	1,600
Fastest Post-LOCA Initiation of Spray System (sec)	15.0
Safeguards Fan Coolers:	
Number of Fan Coolers Operating	4
Fastest Post-LOCA Initiation of Fan Coolers (sec)	0.0
Fan Cooler Performance	See Table 14.3.2-6
Structural Heat Sink Data	See Table 14.3.2-7

Table 14.3.2-6

Kewaunee Large Break LOCA
Fan Cooler Performance Data (Per Fan Cooler)

Steam / Air Temperature (°F)	Heat Removal (Btu/sec)
120.0	1,800
136.0	5,670
205.0	12,550
244.0	16,970
270.0	20,080

Table 14.3.2-7 (Page 1 of 2)

Kewaunee Large Break LOCA Structural Heat Sink Data

Heat Sink	Material Type of Each Layer	Thickness (ft) of Each Layer	Total Exposed Area (ft^2)
1.5 in steel	Steel Paint (7 mils)	0.125 0.000583	26,381
3/4 in steel	Steel Paint (11 mils)	0.0625 0.000917	17,318
1/4 in steel / 12 in concrete	Steel Concrete Concrete	0.02083 0.5 0.5	1,260
1/4 in steel / 12 in concrete	Stainless Steel Concrete Concrete	0.02083 0.5 0.5	6,600
1.5 in steel	Steel Paint (11 mils)	0.125 0.000917	17,823
3/4 in steel	Steel Paint (3 mils)	0.0625 0.000250	9,877
3/8 in steel	Steel	0.03125	6,800
1/2 in steel	Steel	0.04167	44,000
3/4 in steel	Steel	0.0625	4,823
1/4 in steel	Steel	0.02083	32,000
3/16 in steel	Steel	0.01563	35,125
0.090 in steel	Steel	0.0075	12,400
0.144 in steel	Steel	0.012	1,695
0.100 in steel	Steel	0.00833	6,000
1.440 in steel	Steel	0.12	2,200

Table 14.3.2-7 (Page 2 of 2)

Kewaunee Large Break LOCA Structural Heat Sink Data

Heat Sink	Material Type of Each Layer	Thickness (ft) of Each Layer	Total Exposed Area (ft^2)
12 in concrete	Concrete Concrete	0.5 0.5	3,400
12 in concrete	Concrete Concrete Paint (8 mils)	0.5 0.5 0.000667	37,400
6 in concrete	Concrete	0.5	19,700
6 in concrete	Concrete Paint (8 mils)	0.5 0.000667	5,300
3 in concrete	Concrete	0.25	2,370
3 in concrete	Concrete Paint (8 mils)	0.25 0.000667	5,200

Table 14.3.2-8

Kewaunee Large Break LOCA Mass and Energy Releases

Time (sec)	Mass Release (lbm/sec)	Energy Release (Btu/sec)
0.0	0.	0.
0.5	45700.	23700000.
2.5	30000.	15800000.
7.0	11000.	7800000.
10.0	6000.	4400000.
12.5	7300.	4000000.
20.0	3000.	1000000.
22.0	6200.	1570000.
27.0	0.	20000.
33.0	0.	50000.
35.0	4100.	750000.
40.0	700.	340000.
50.0	100.	100000.
85.0	100.	90000.
135.0	800.	240000.
175.0	200.	115000.
225.0	150.	90000.
275.0	250.	125000.
300.0	250.	125000.

Table 14.3.2-9

Kewaunee Large Break LOCA Sequence of Events

Event	Transient Time (sec)		
	Appendix K	Superbounded	Nominal
Accident Initiation	0.00	0.00	0.00
Reactor Trip Signal	6.40	6.32	6.23
Safety Injection Signal	6.94	6.84	6.75
Blowdown PCT Occurs	7.50	6.50	6.50
Accumulator Injection Begins	7.90	6.55	6.65
End of ECCS Bypass	20.19	-	-
End of Blowdown	20.19	17.49	16.30
High Head Safety Injection Begins	21.94	21.84	21.75
Low Head Safety Injection Begins	26.94	26.84	26.75
Bottom of Core Recovery	27.91	29.49	27.30
Hot Rod Burst Occurs	32.88	-	-
Accumulator Water Empty	34.35	32.52	32.80
Hot Assembly Burst Occurs	48.66	-	-
Reflood PCT Occurs	200.0	75.0	178.0

Table 14.3.2-10

Kewaunee Large Break LOCA Results

Result	Transient		
	Appendix K	Superbounded	Nominal
Calculated Blowdown PCT (°F)	1695	1827 *	1384
Blowdown PCT Location (ft)	7.875	7.625	8.125
Calculated Reflood PCT (°F)	2009	1772 *	1525
Reflood PCT Location (ft)	8.125	7.625	8.125
Maximum Local Zr / H_2O Reaction	4.11 %	< 17.0 %	< 17.0 %
Local Zr / H_2O Location (ft)	7.875	7.625	8.125
Total Corewide Zr / H_2O Reaction	0.005 %	< 1.0 %	< 1.0 %

* In the Superbounded calculation, peak clad temperatures at the 95th percentile probability level with 95 percent confidence is obtained by adding the calculated peak clad temperature to the code bias plus uncertainties. The Superbounded peak clad temperatures listed in the above table already include the uncertainties. The sum of the code bias plus uncertainties has been determined to be 260.5°F in blowdown and 175.0°F in reflood (References 3 and 4).

Figure 14.3.2-1

Large Break LOCA Sequence of Events

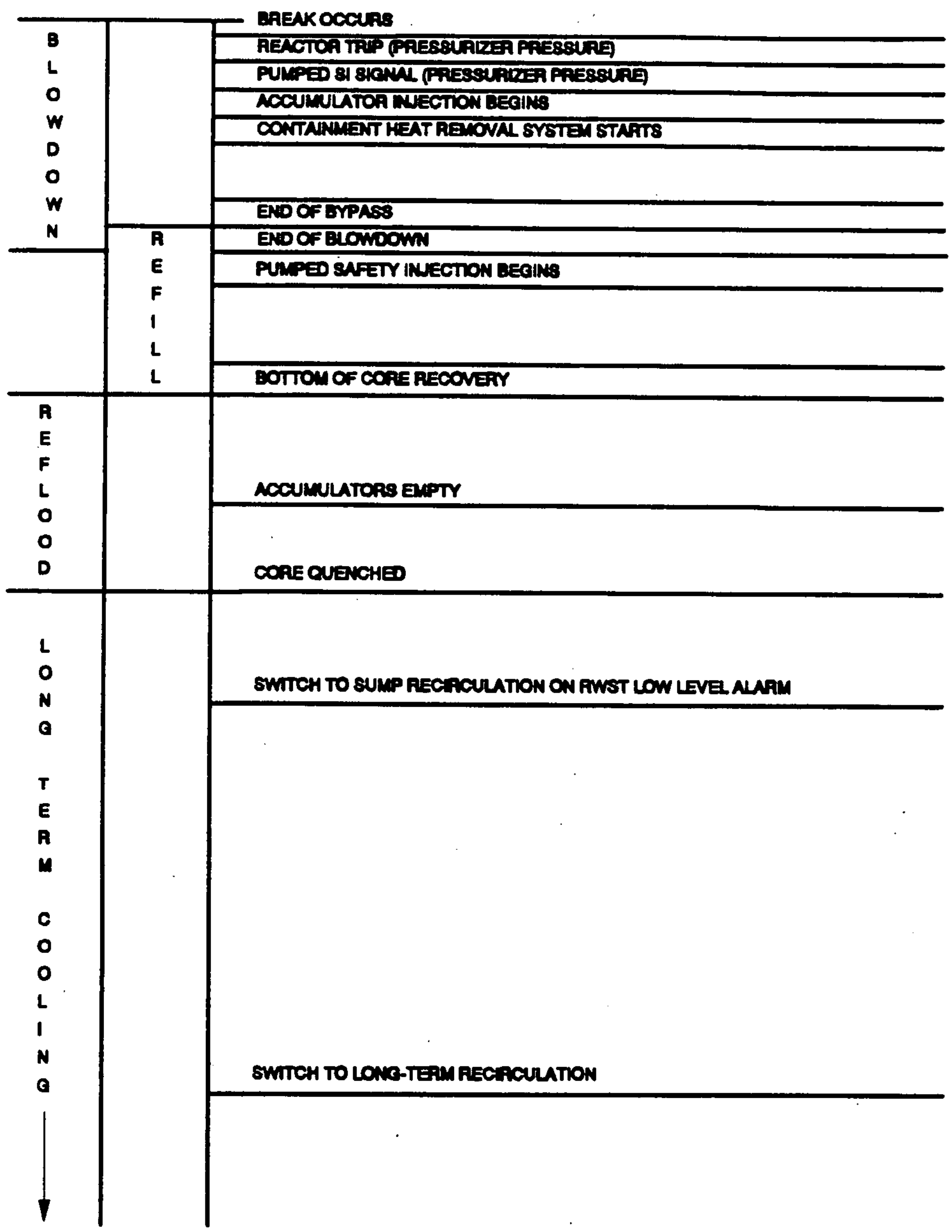

Figure 14.3.2-2

Large Break LOCA Containment Pressure Curve

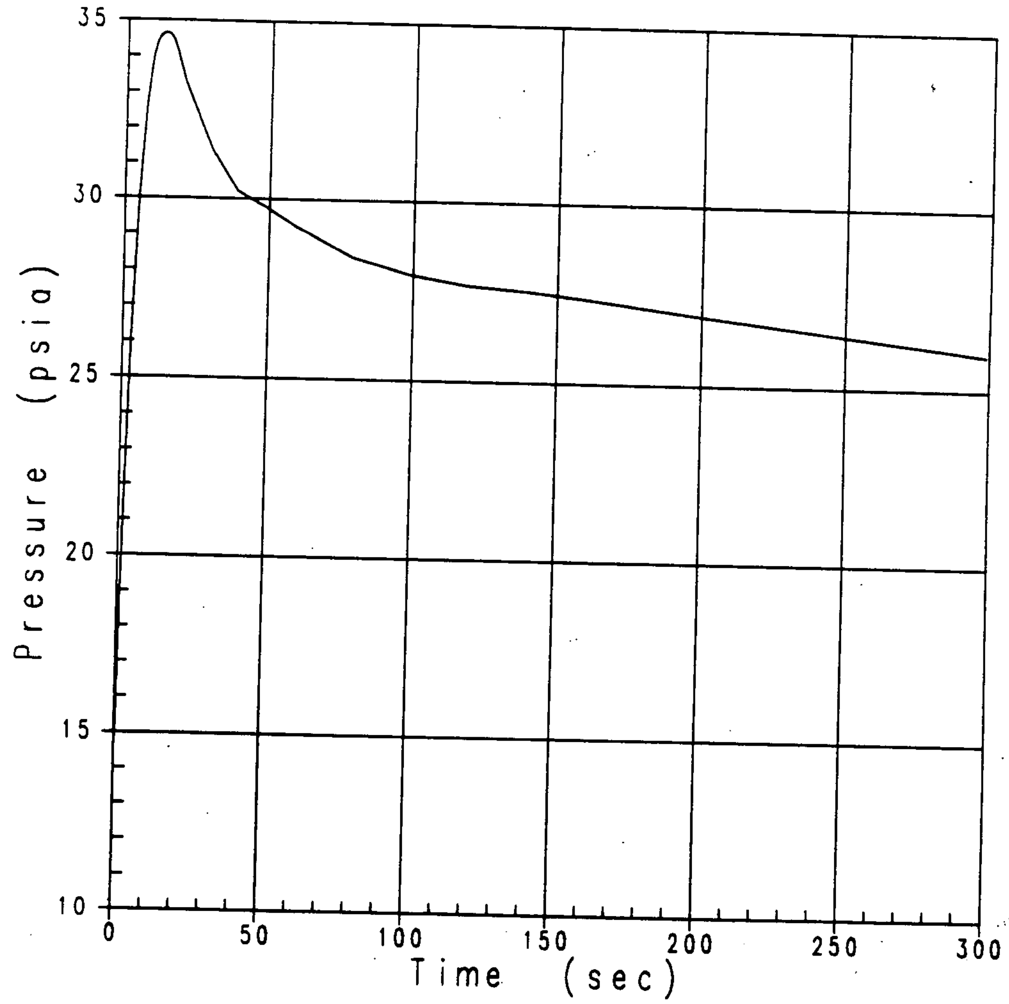

Figure 14.3.2-3

Large Break LOCA Appendix K Power Shape

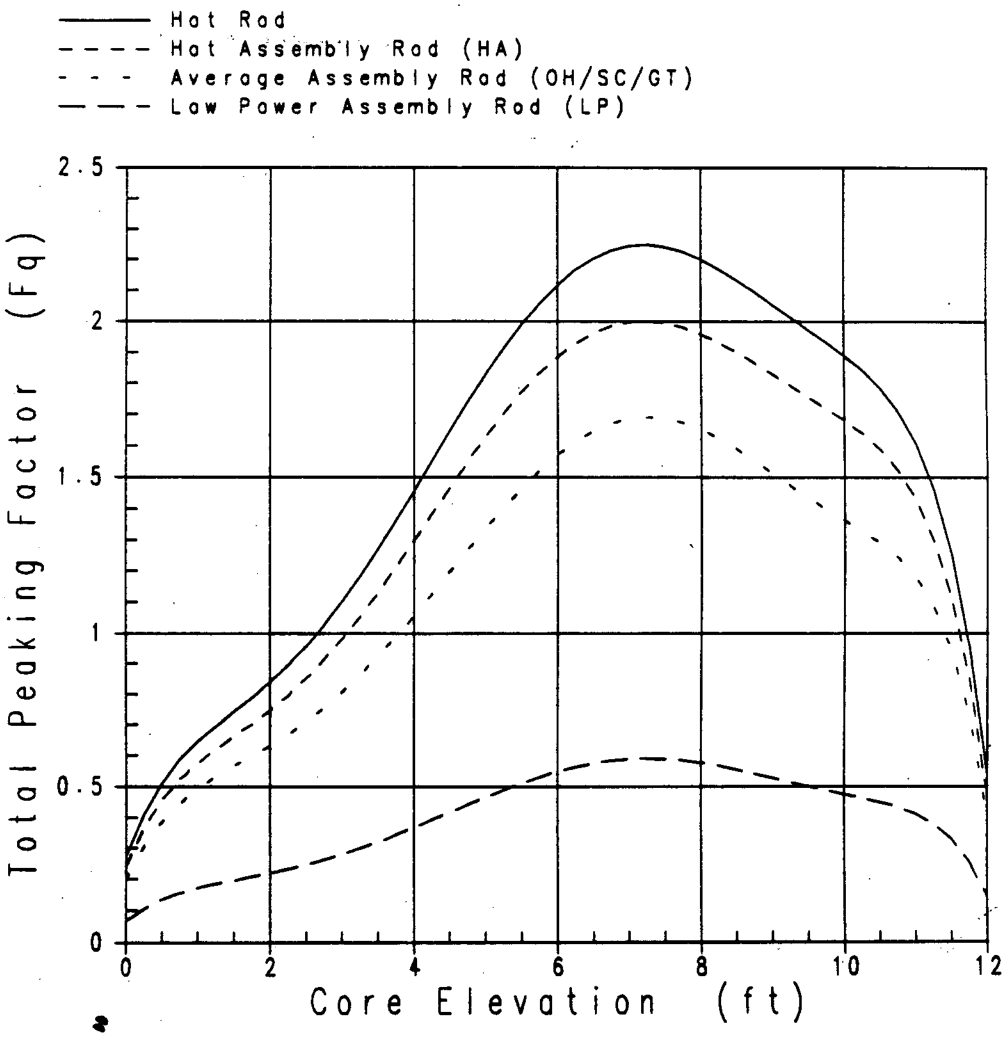

Figure 14.3.2-4

Appendix K Calculation
Peak Cladding Temperature

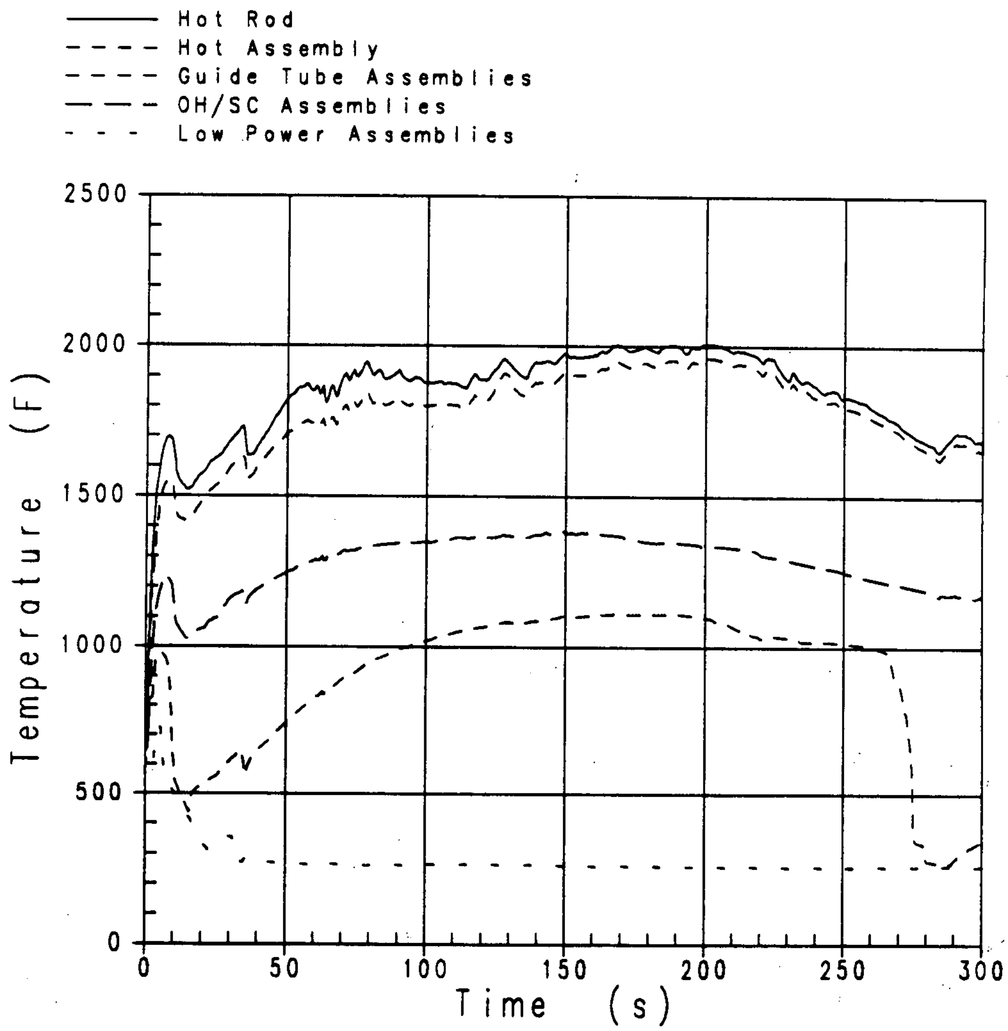

Figure 14.3.2-5

Appendix K Calculation
Core Pressure

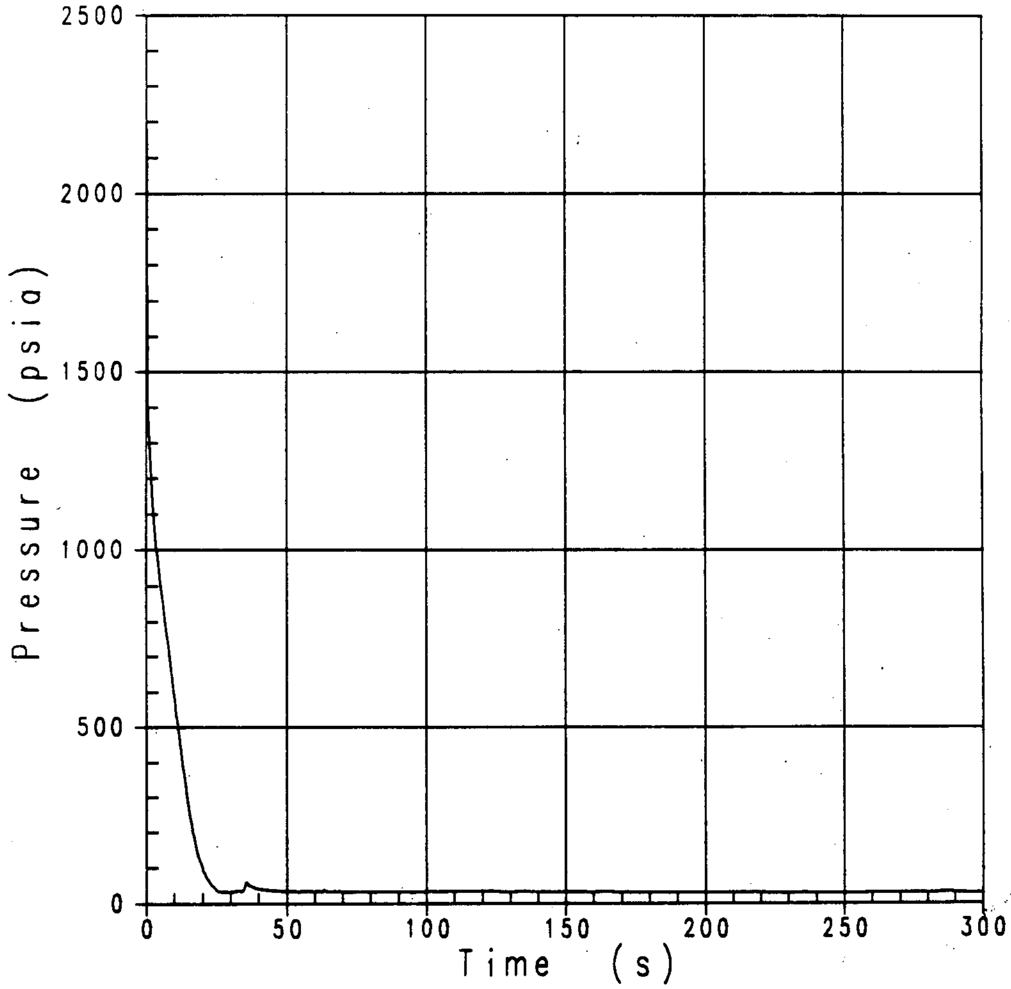

Figure 14.3.2-6

Appendix K Calculation
Vessel Inventory

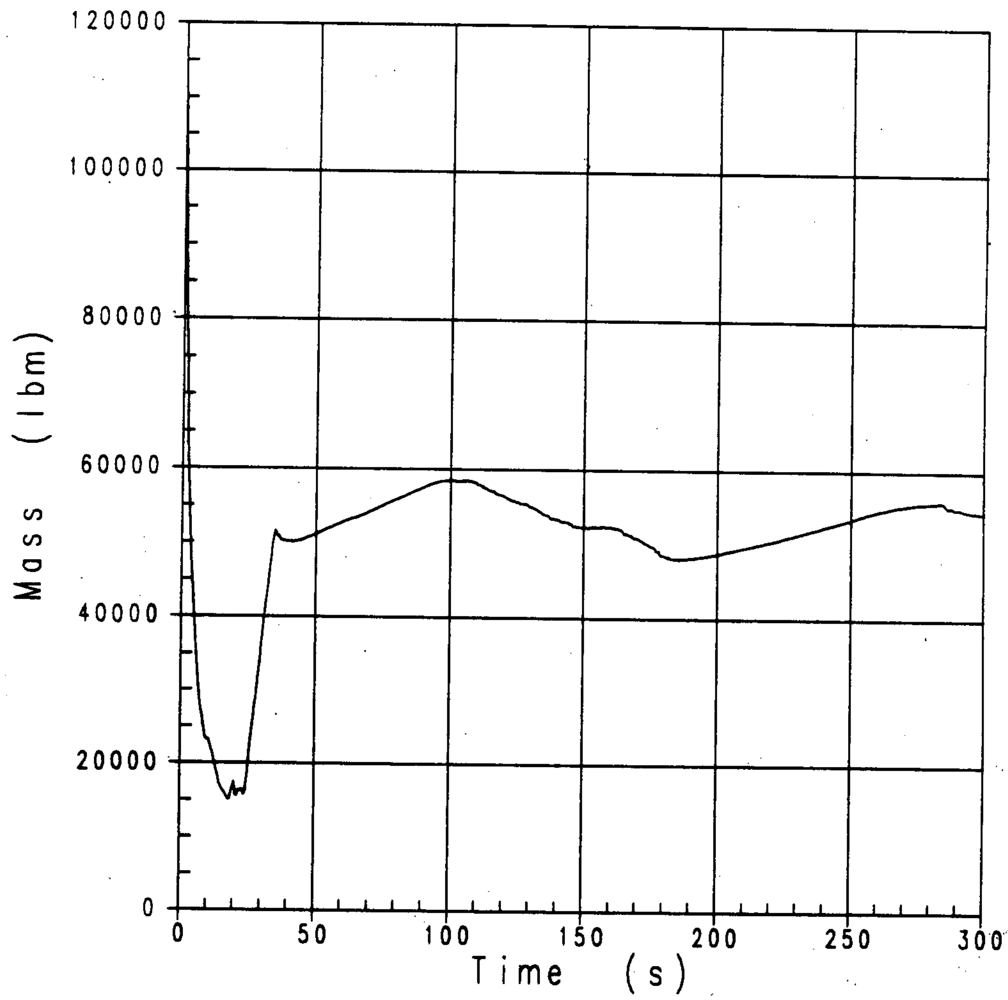

Figure 14.3.2-7

Appendix K Calculation
Loop Side Break Flow

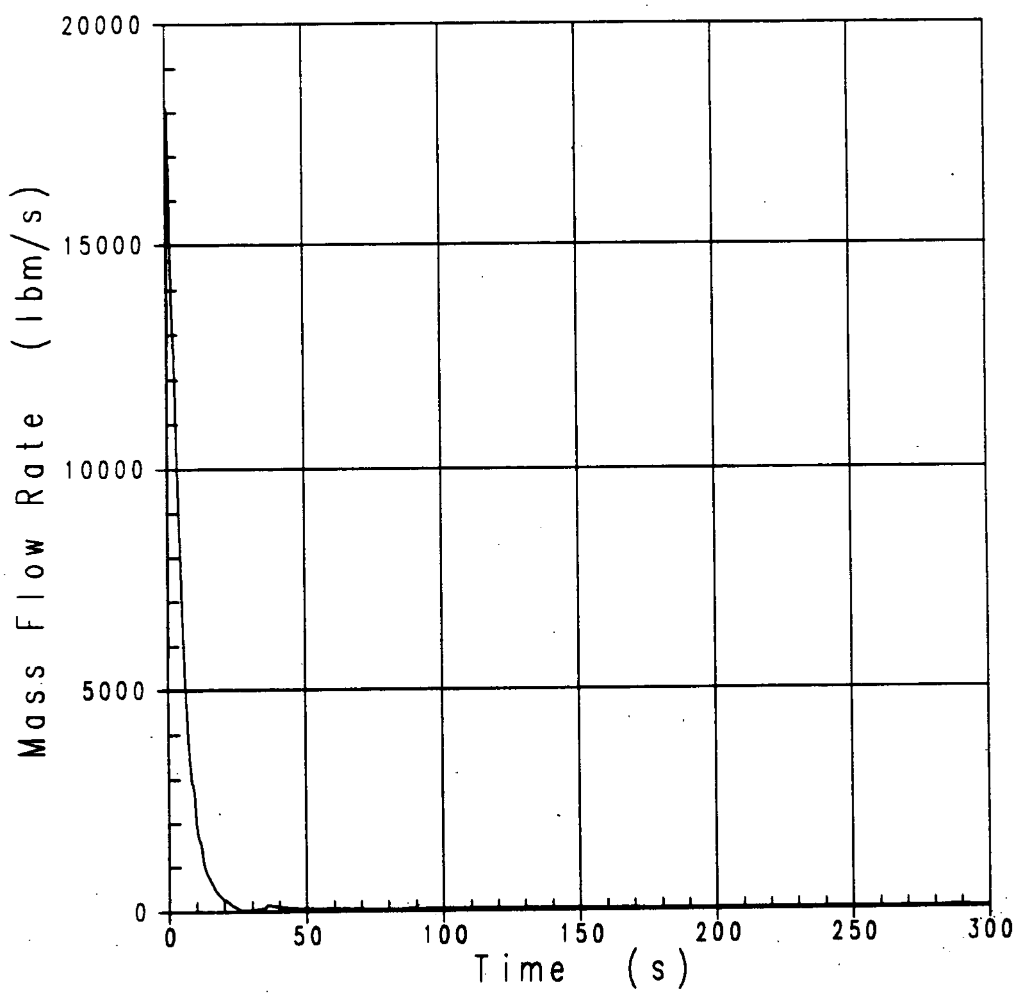

Figure 14.3.2-8

Appendix K Calculation
Vessel Side Break Flow

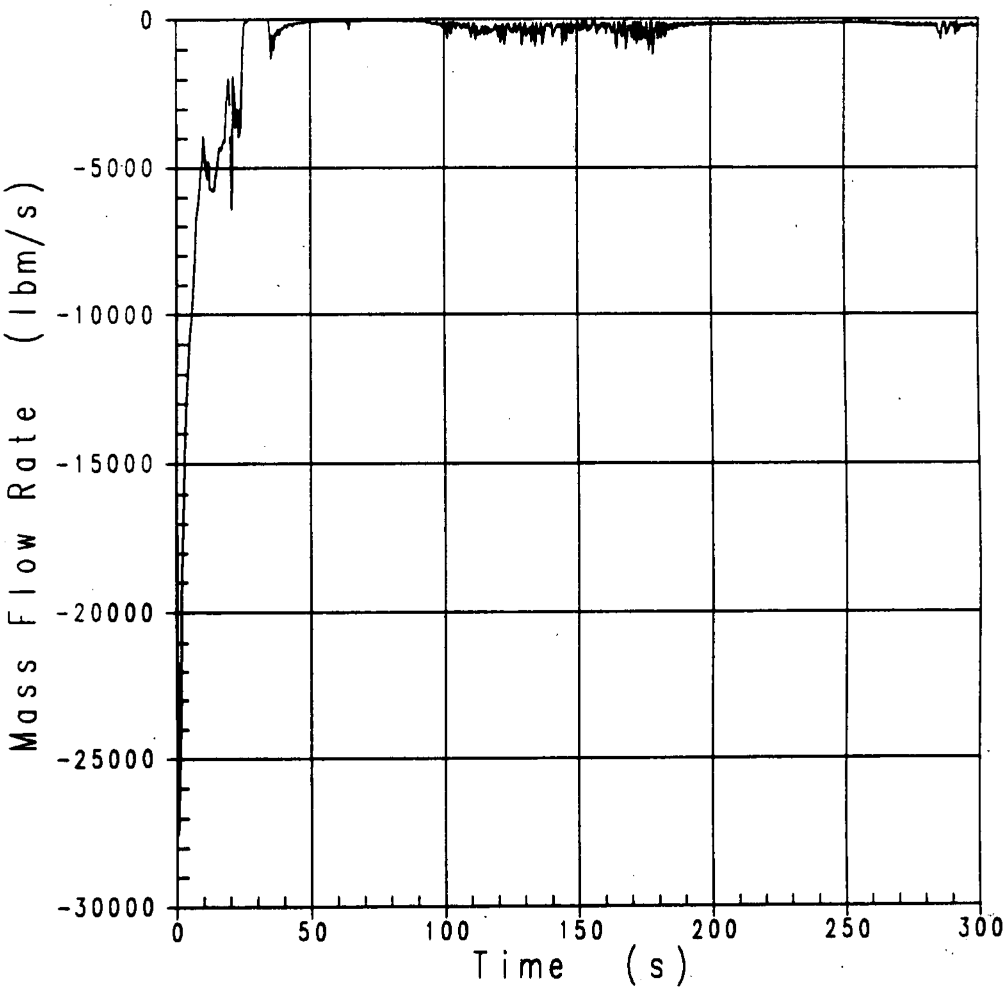

Figure 14.3.2-9

Appendix K Calculation
Accumulator Flow

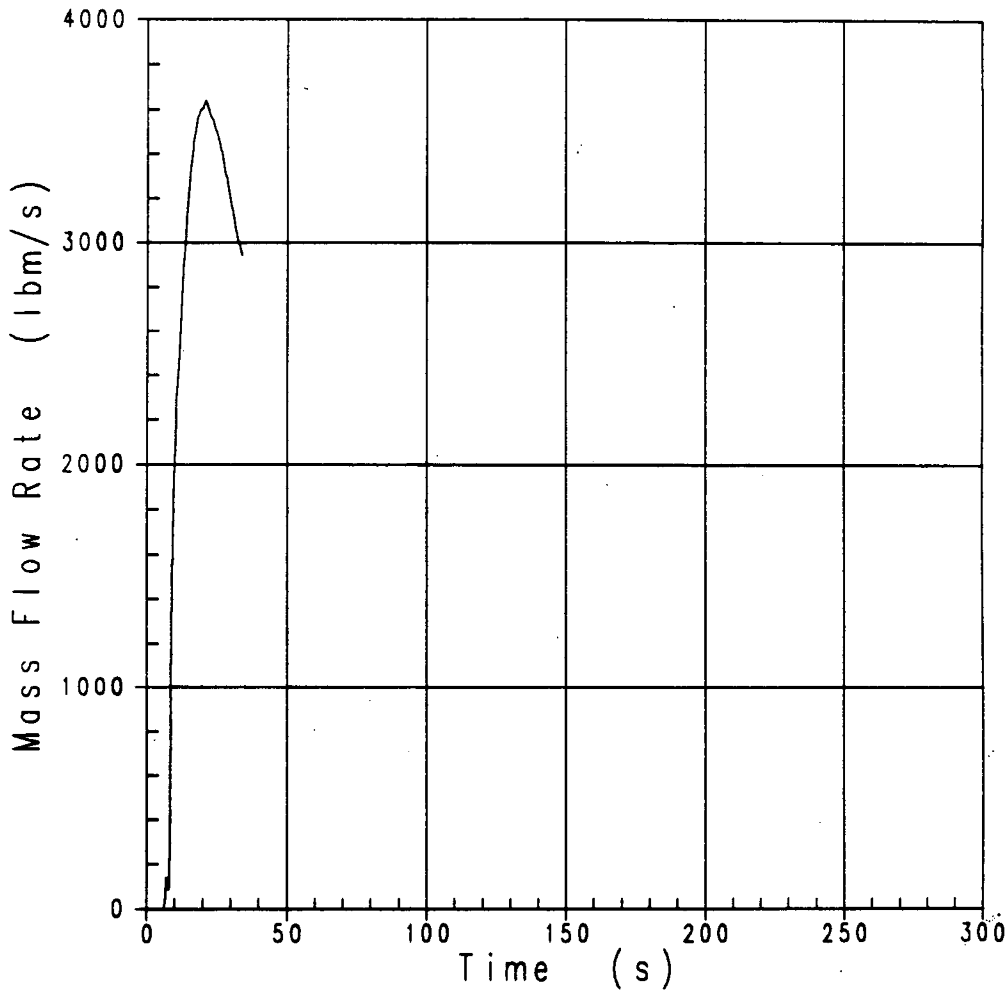

Figure 14.3.2-10

Appendix K Calculation
High Head Safety Injection Flow

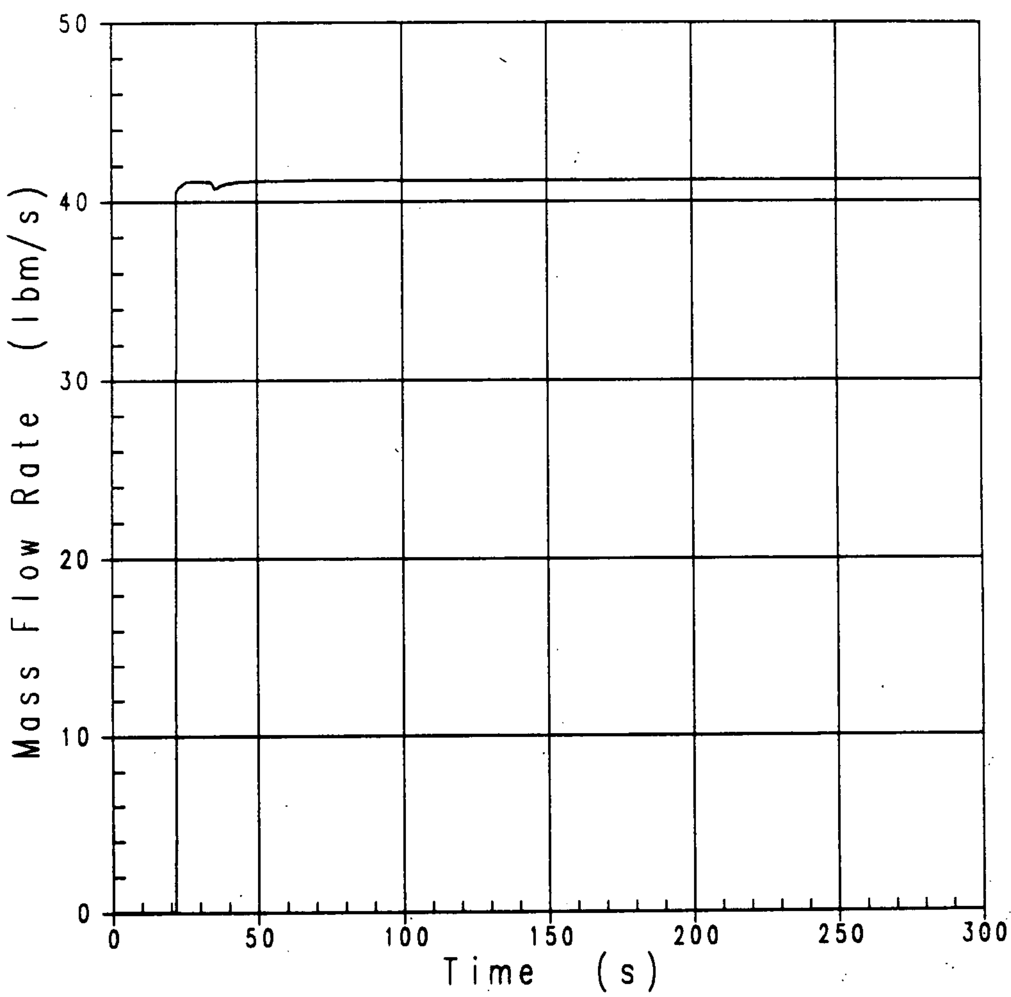

Figure 14.3.2-11

Appendix K Calculation
Low Head Safety Injection Flow

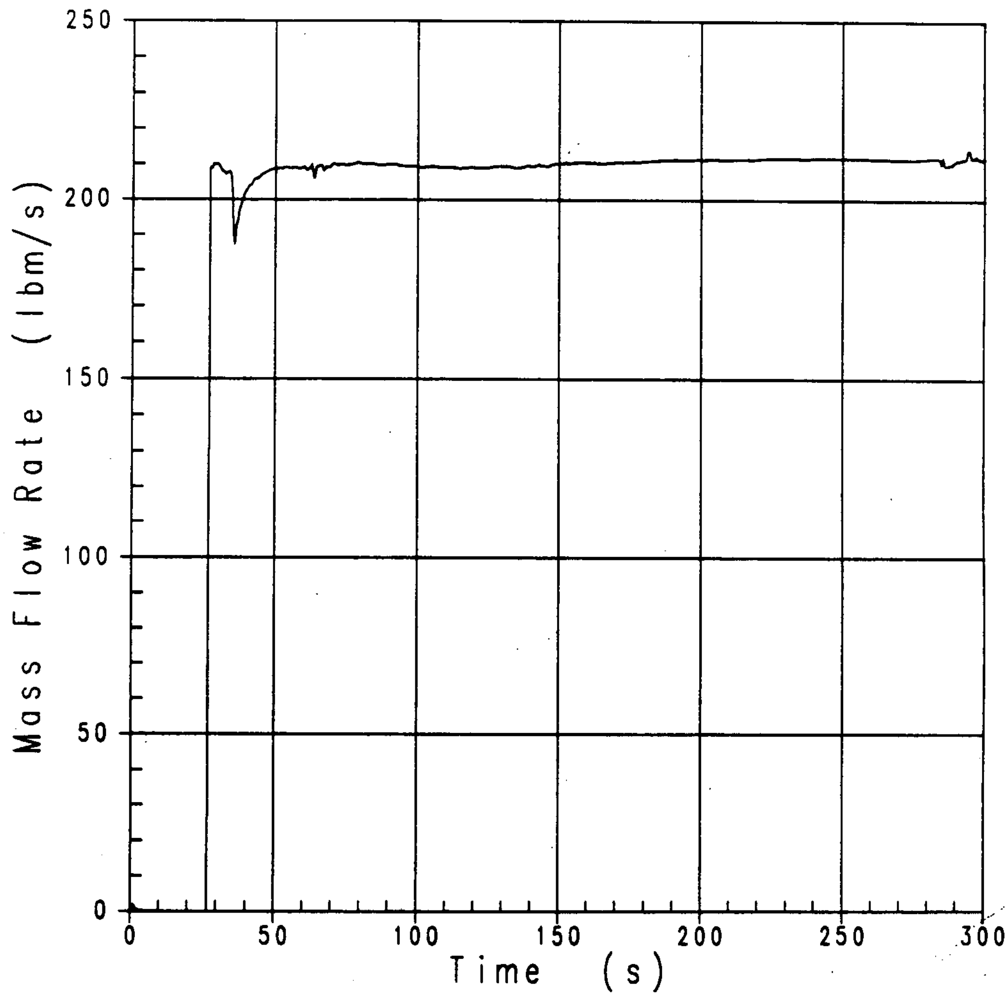

Figure 14.3.2-12

Appendix K Calculation
Liquid Flow at Top of Core Channel 10 (OH/SC)

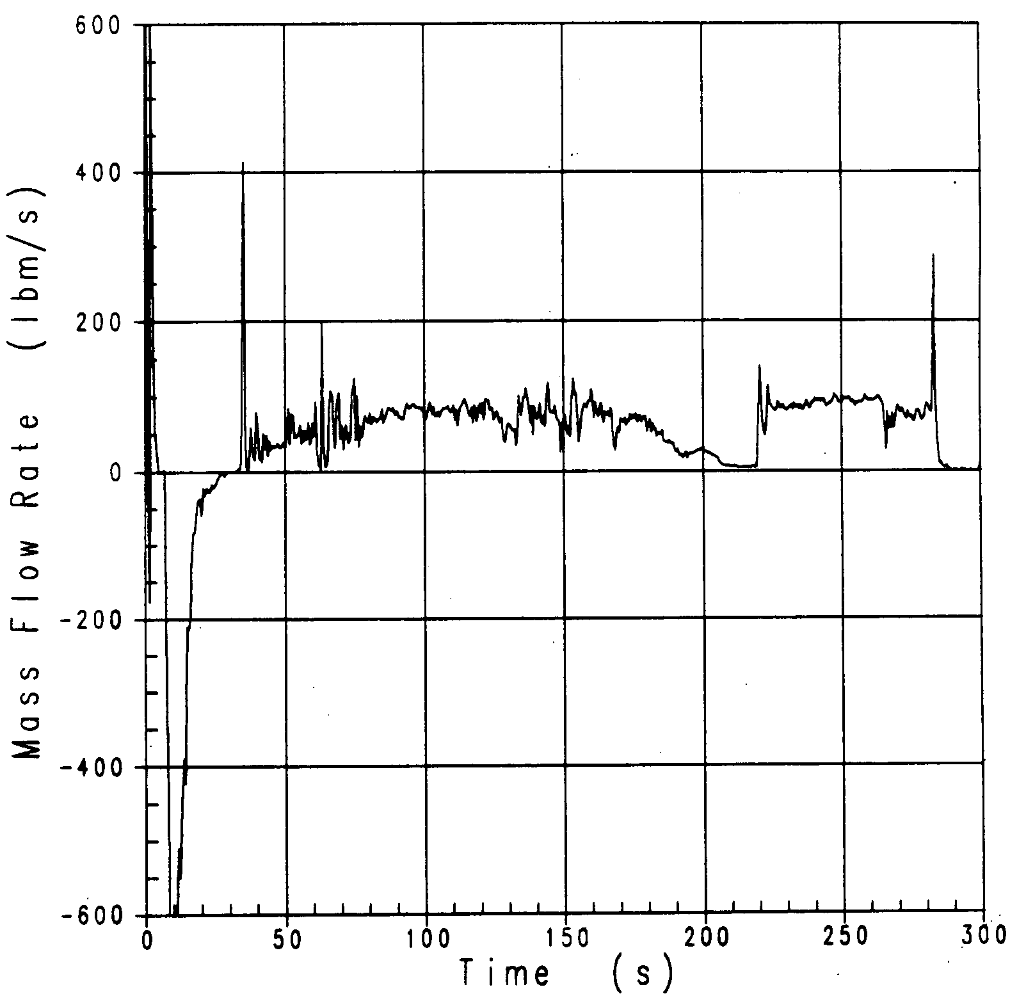

Figure 14.3.2-13

Appendix K Calculation
Vapor Flow at Top of Core Channel 10 (OH/SC)

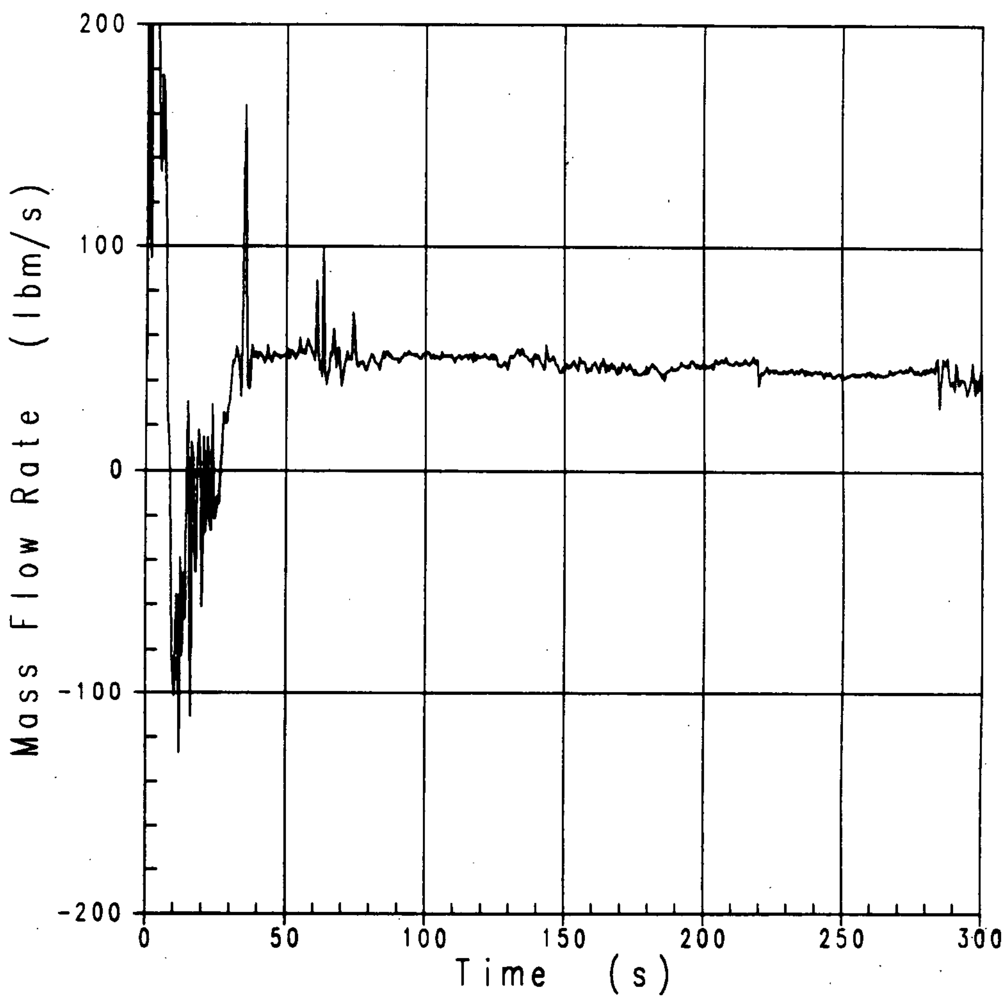

Figure 14.3.2-14

Appendix K Calculation
Liquid Flow at Top of Core Channel 11 (GT)

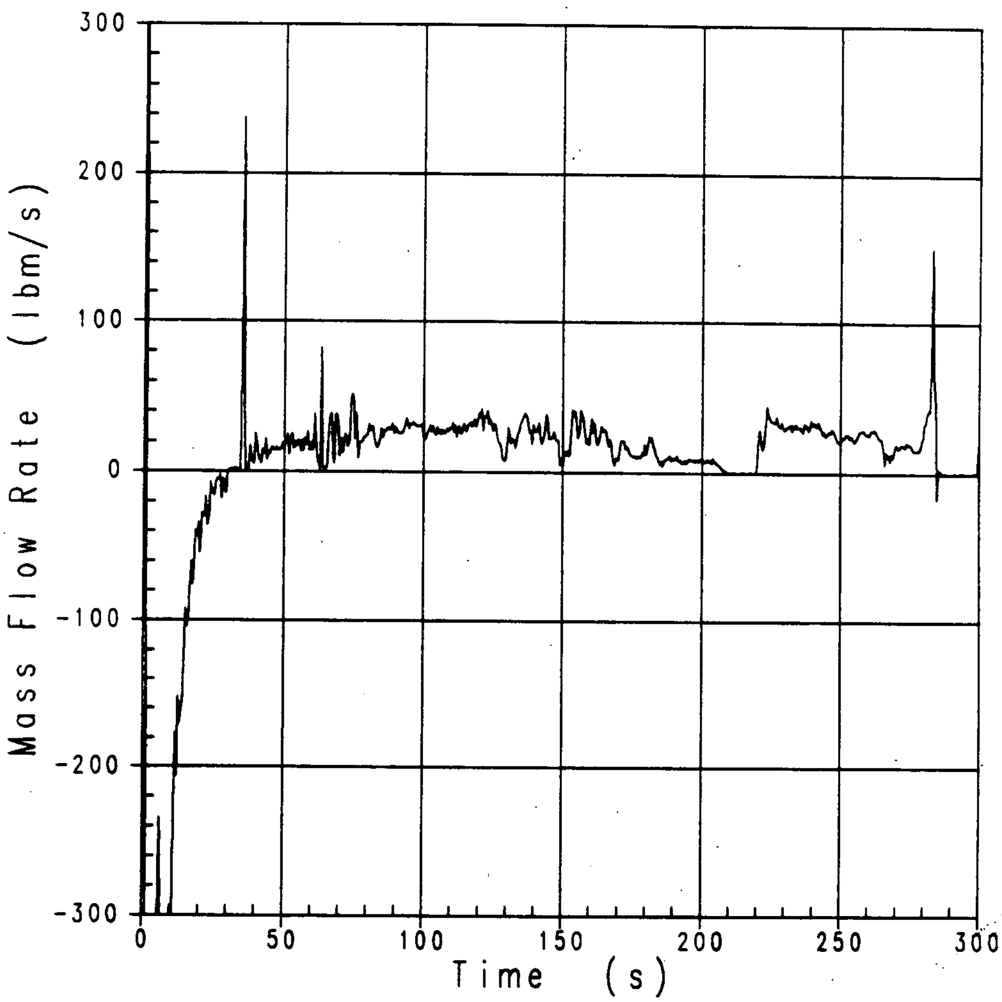

Figure 14.3.2-15

Appendix K Calculation
Vapor Flow at Top of Core Channel 11 (GT)

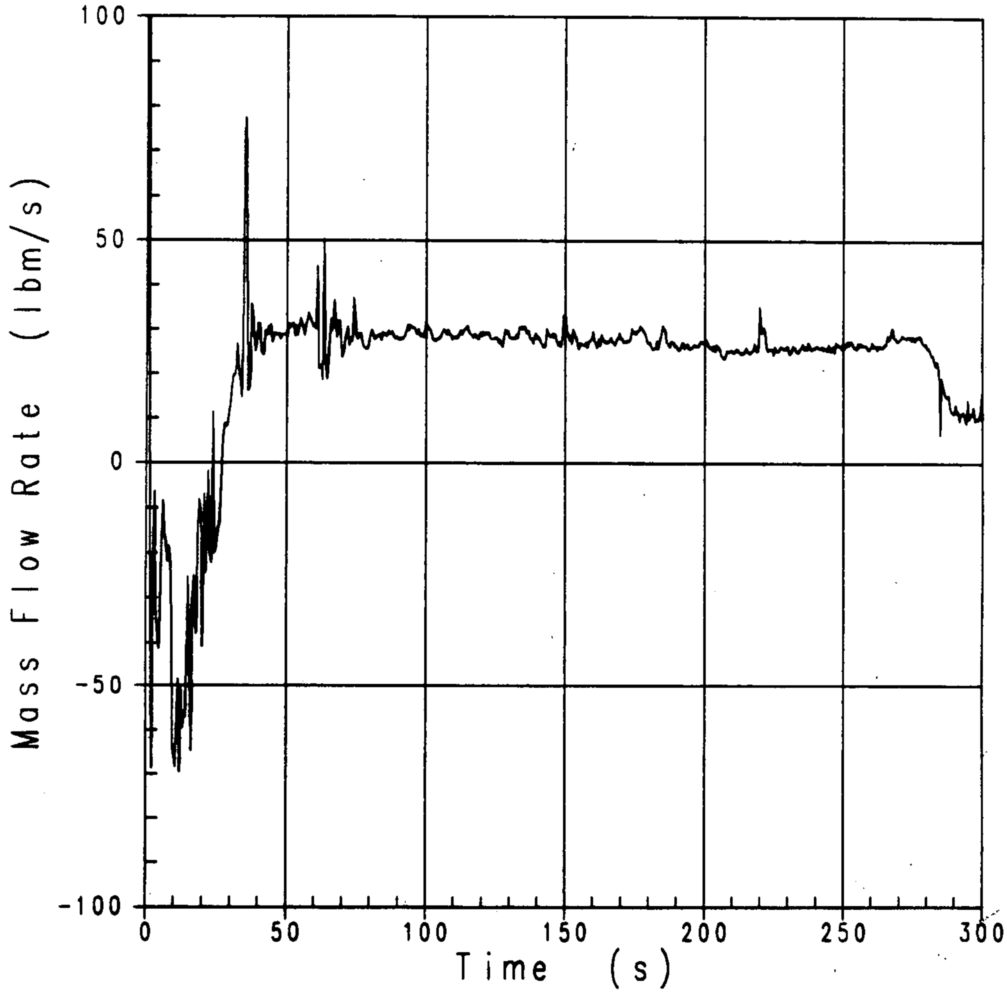

Figure 14.3.2-16

Appendix K Calculation
Liquid Flow at Top of Core Channel 12 (HA)

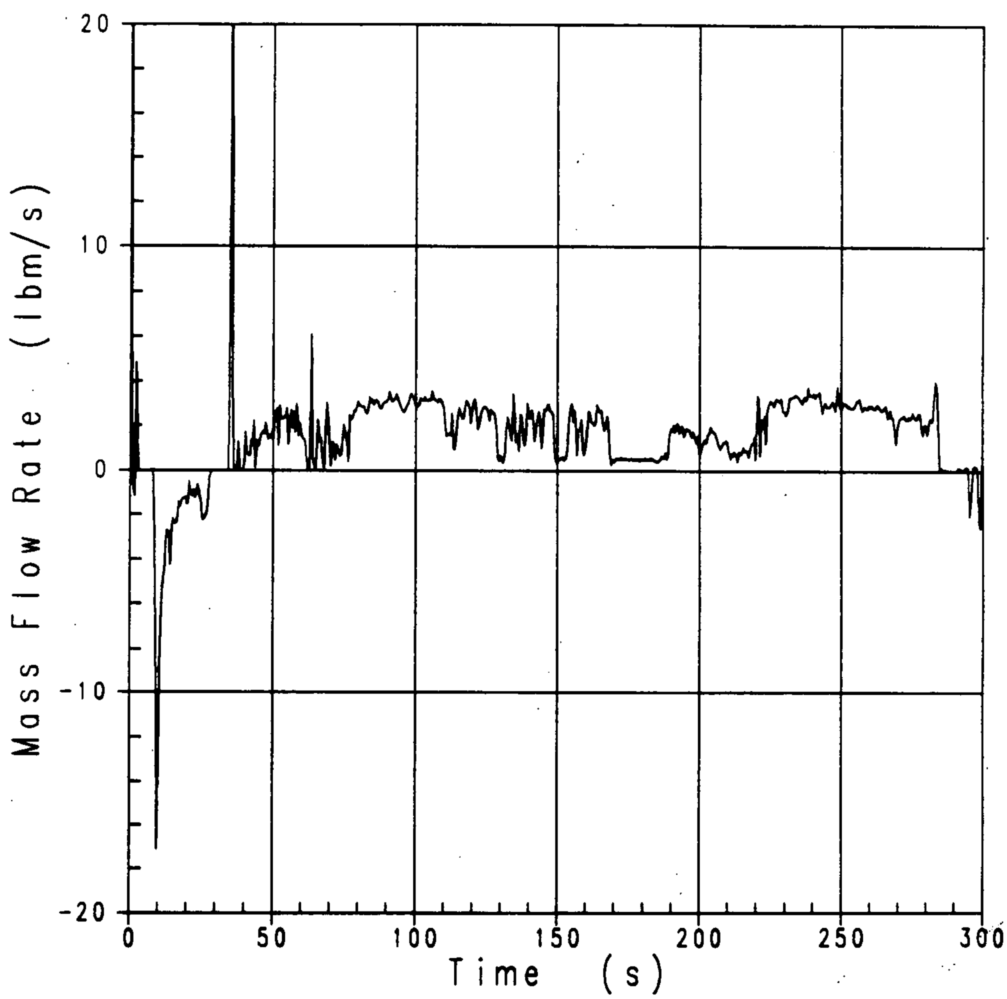

Figure 14.3.2-17

Appendix K Calculation
Vapor Flow at Top of Core Channel 12 (HA)

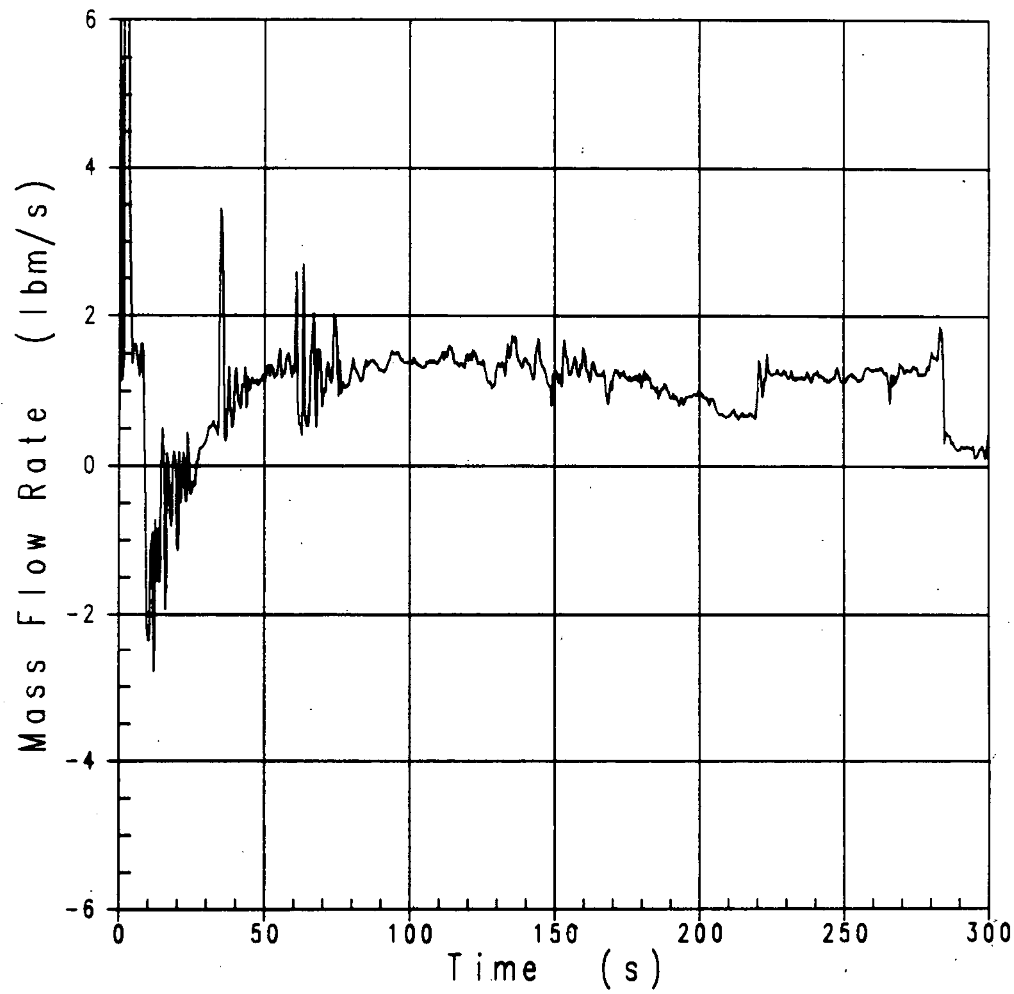

Figure 14.3.2-18

Appendix K Calculation
Liquid Flow at Top of Core Channel 13 (LP)

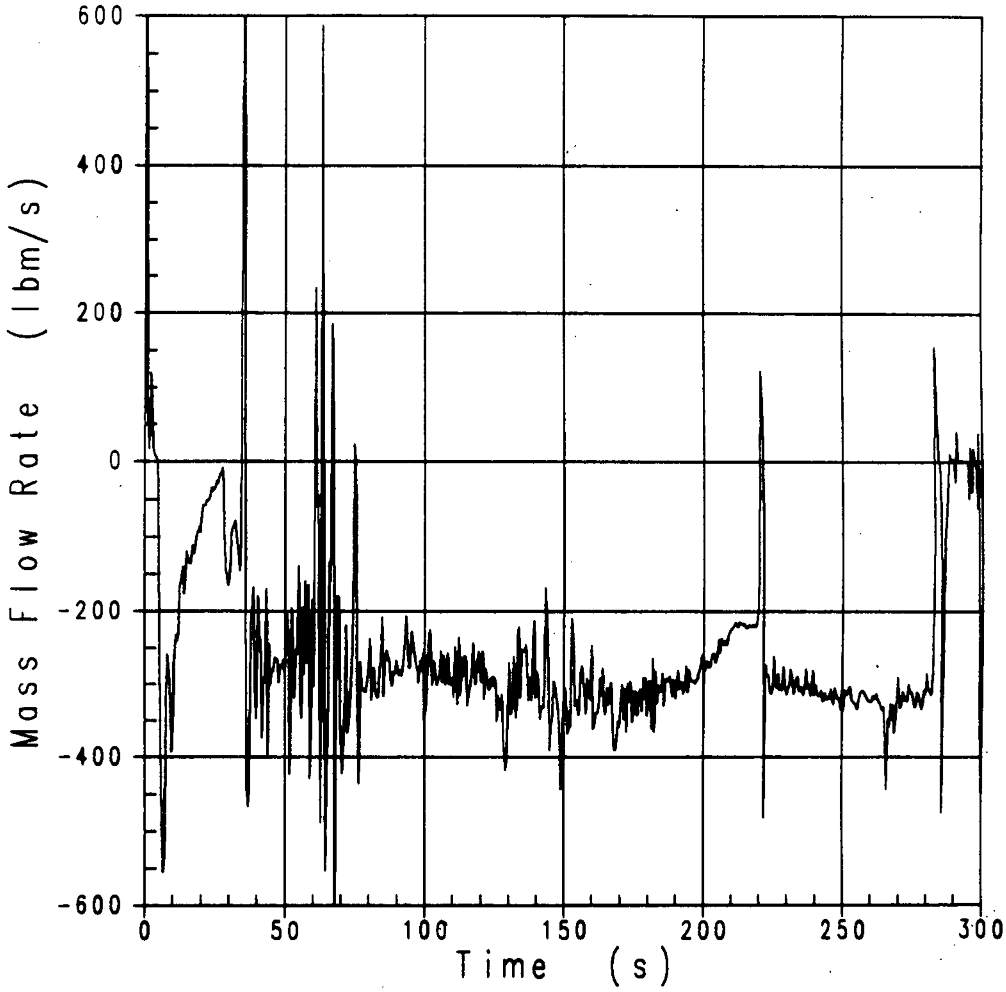

Figure 14.3.2-19

Appendix K Calculation
Vapor Flow at Top of Core Channel 13 (LP)

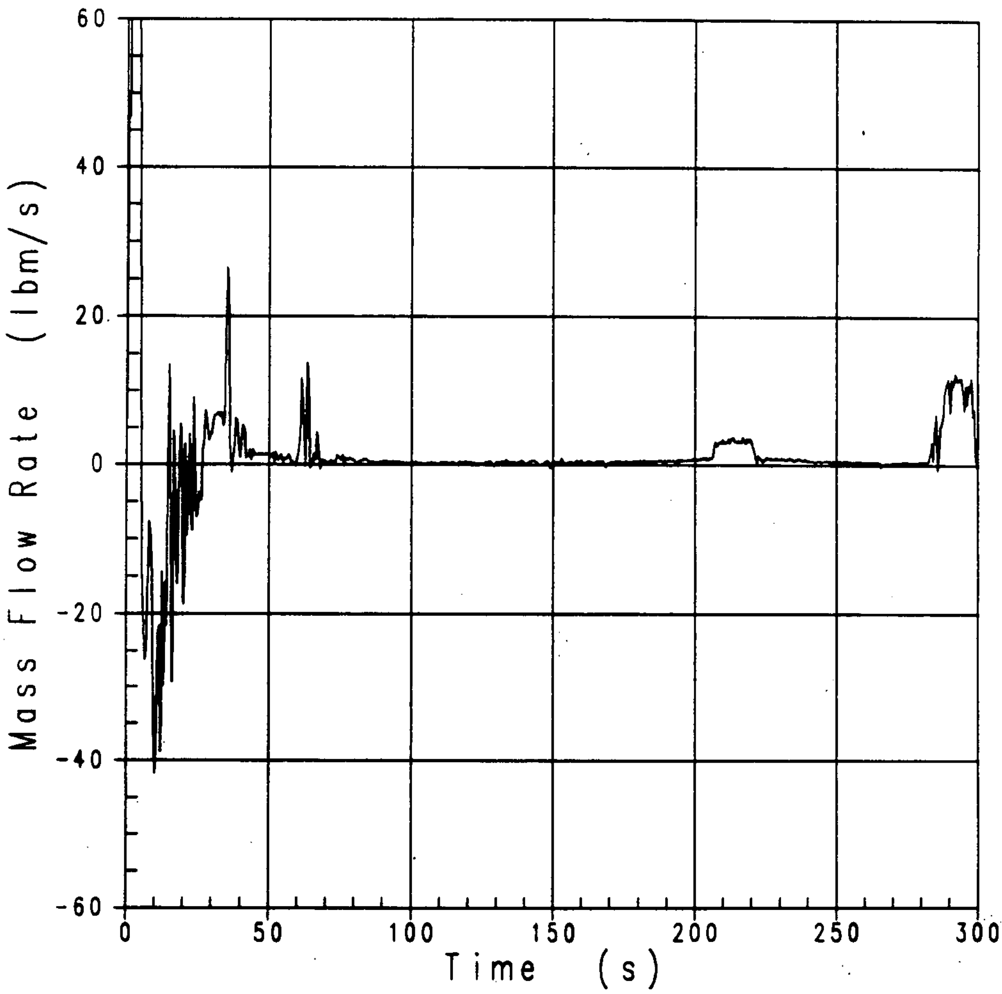

Figure 14.3.2-20

Appendix K Calculation
Downcomer Liquid Level

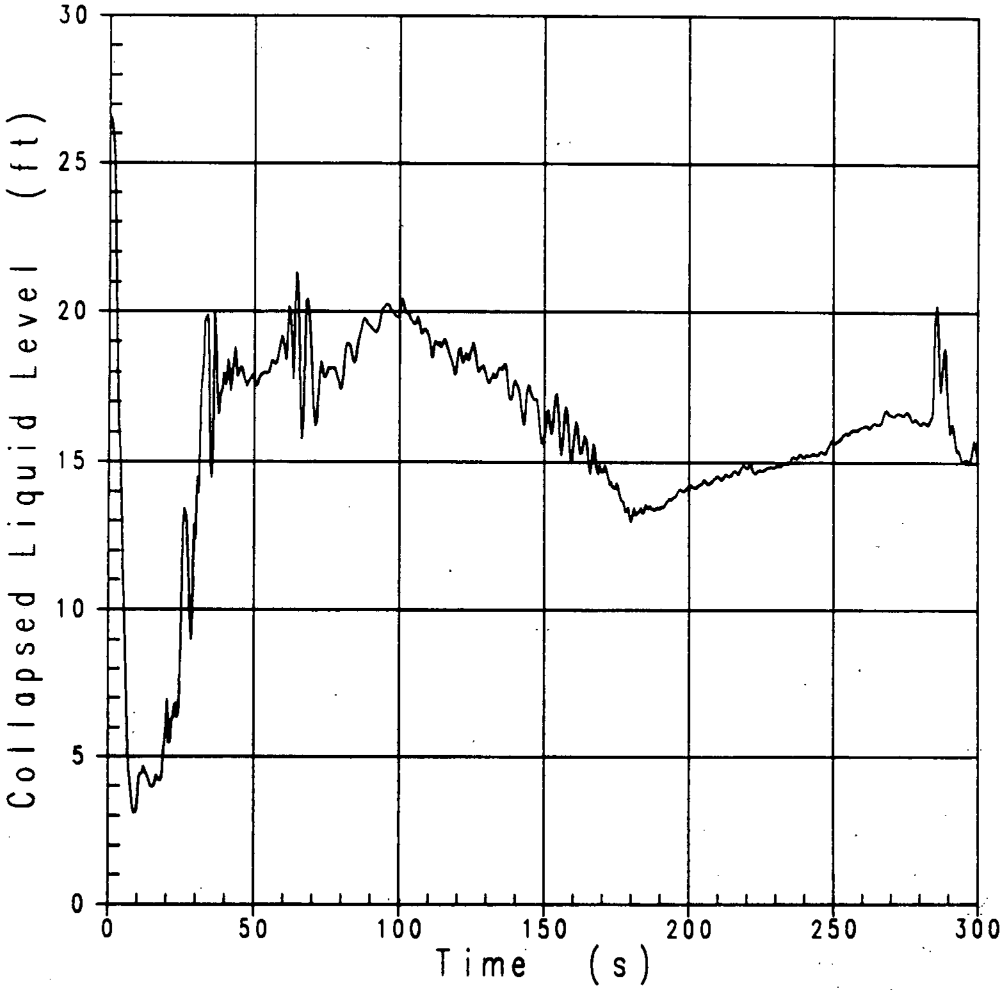

Figure 14.3.2-21

Appendix K Calculation
Lower Plenum Liquid Level

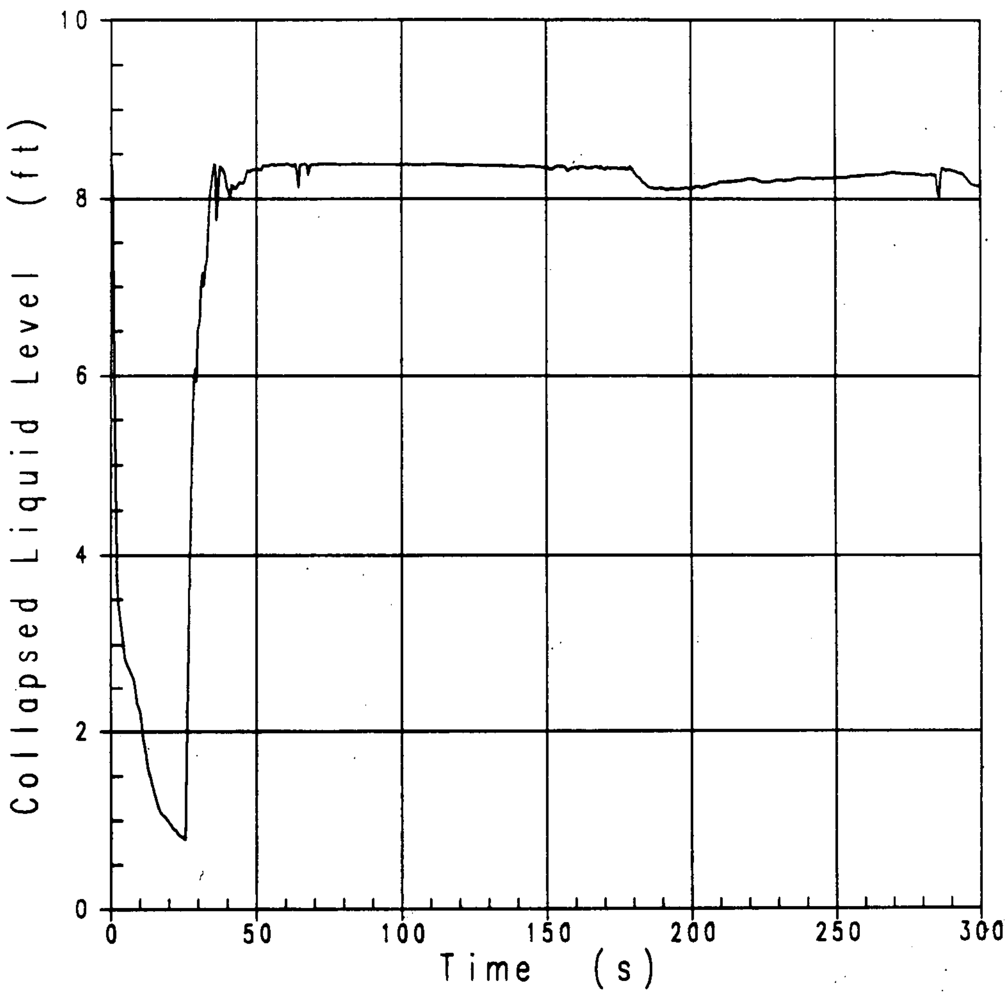

Figure 14.3.2-22

Appendix K Calculation
Core Liquid Level

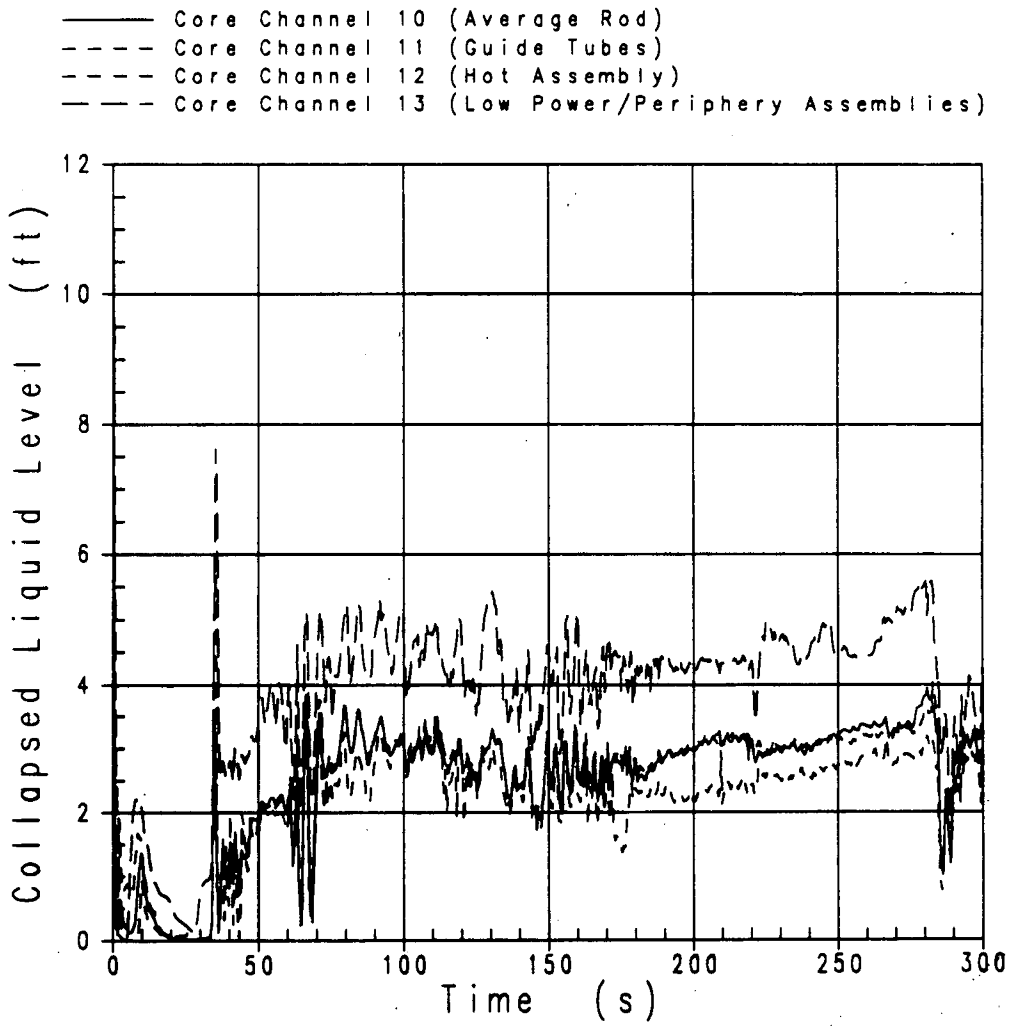

Figure 14.3.2-23

Appendix K Calculation
Void Fraction in Lower Plenum

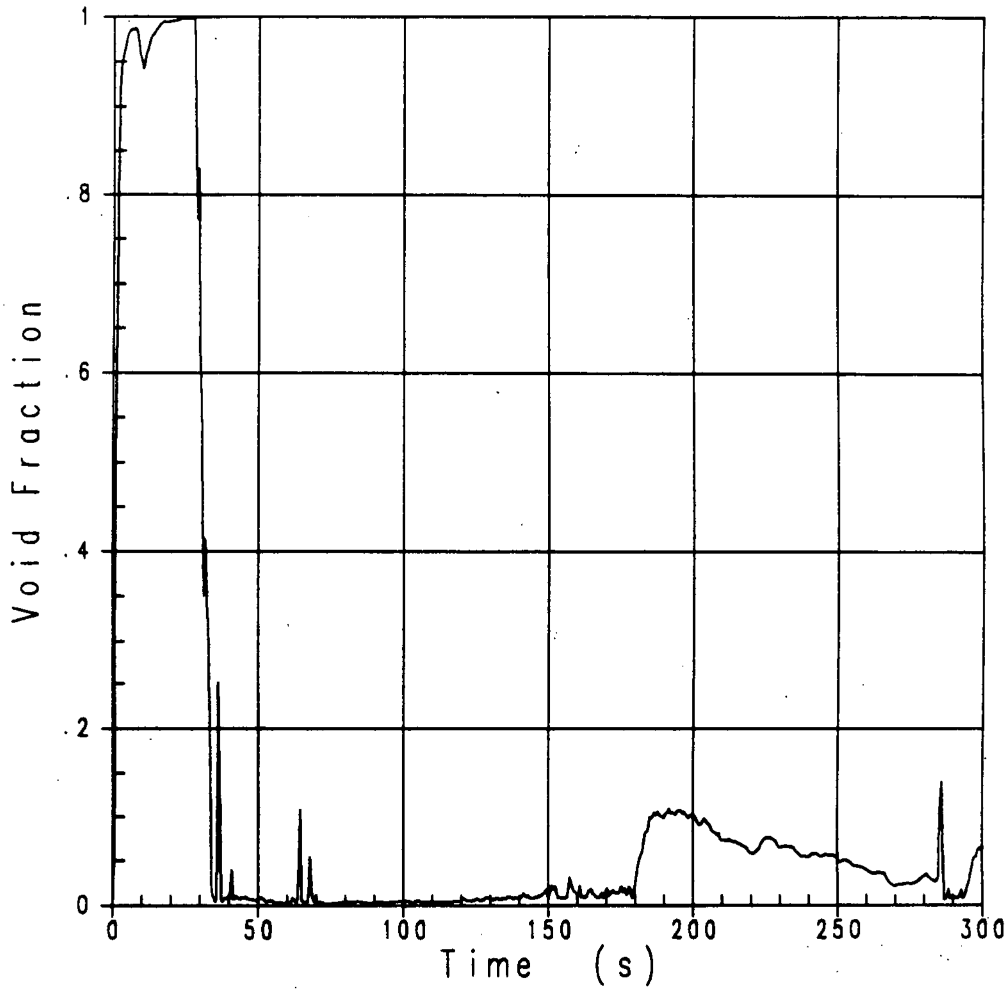

Figure 14.3.2-24

Appendix K Calculation
Void Fraction at Bottom of Core

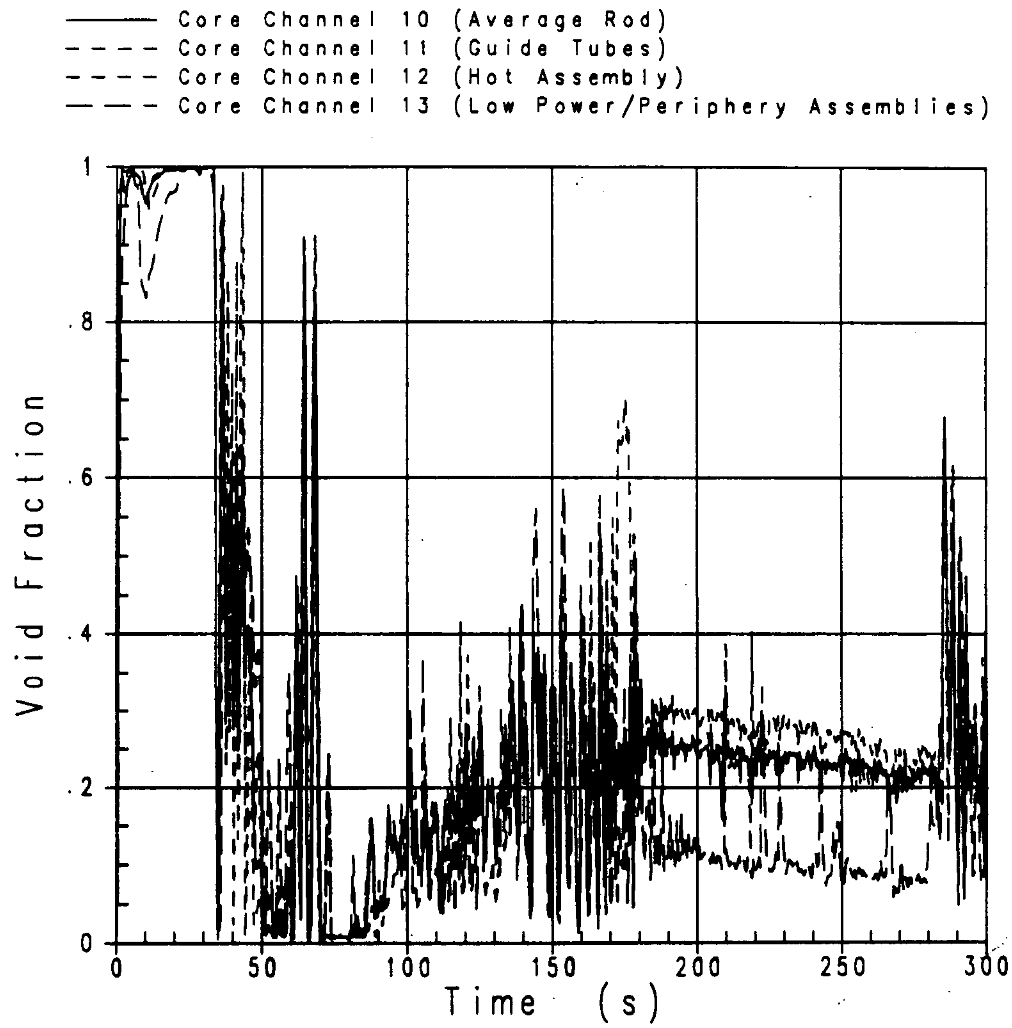

Figure 14.3.2-25

Superbounded Calculation
Peak Cladding Temperature

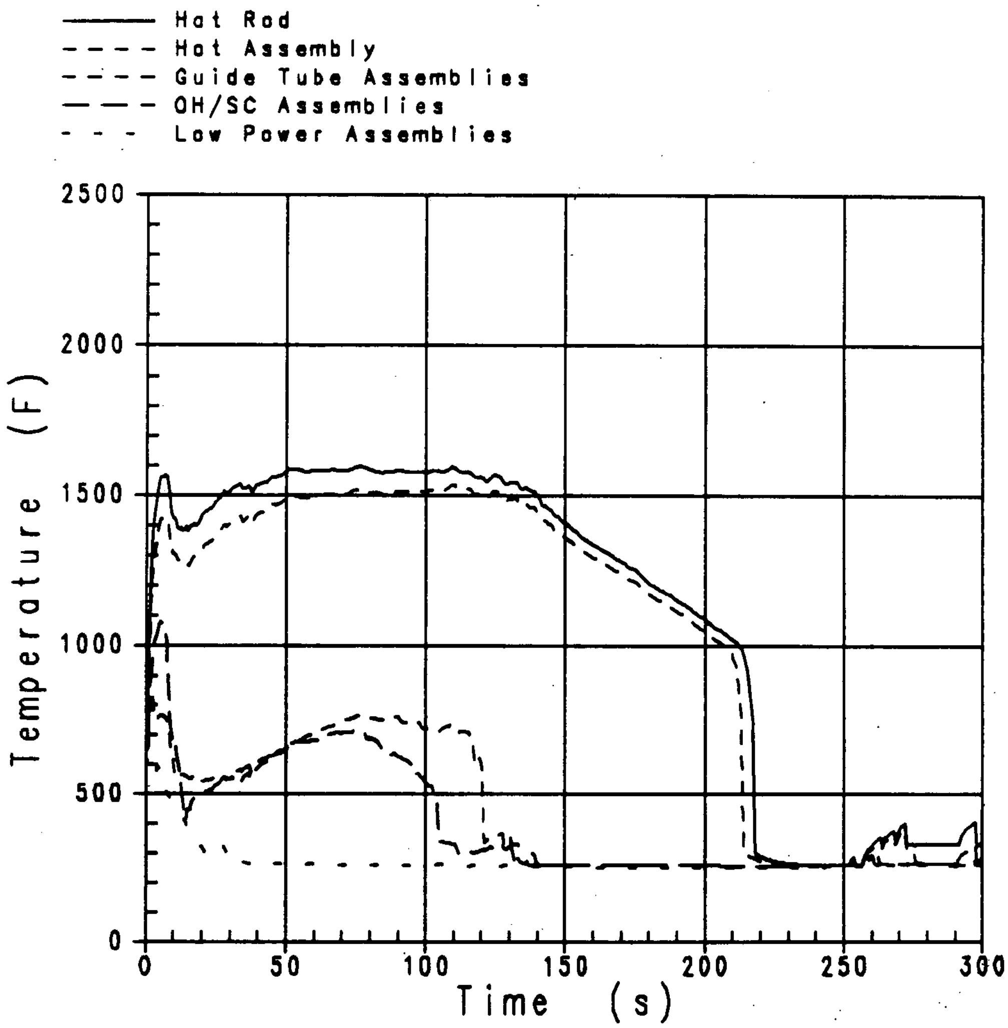

Figure 14.3.2-26

Superbounded Calculation
Core Pressure

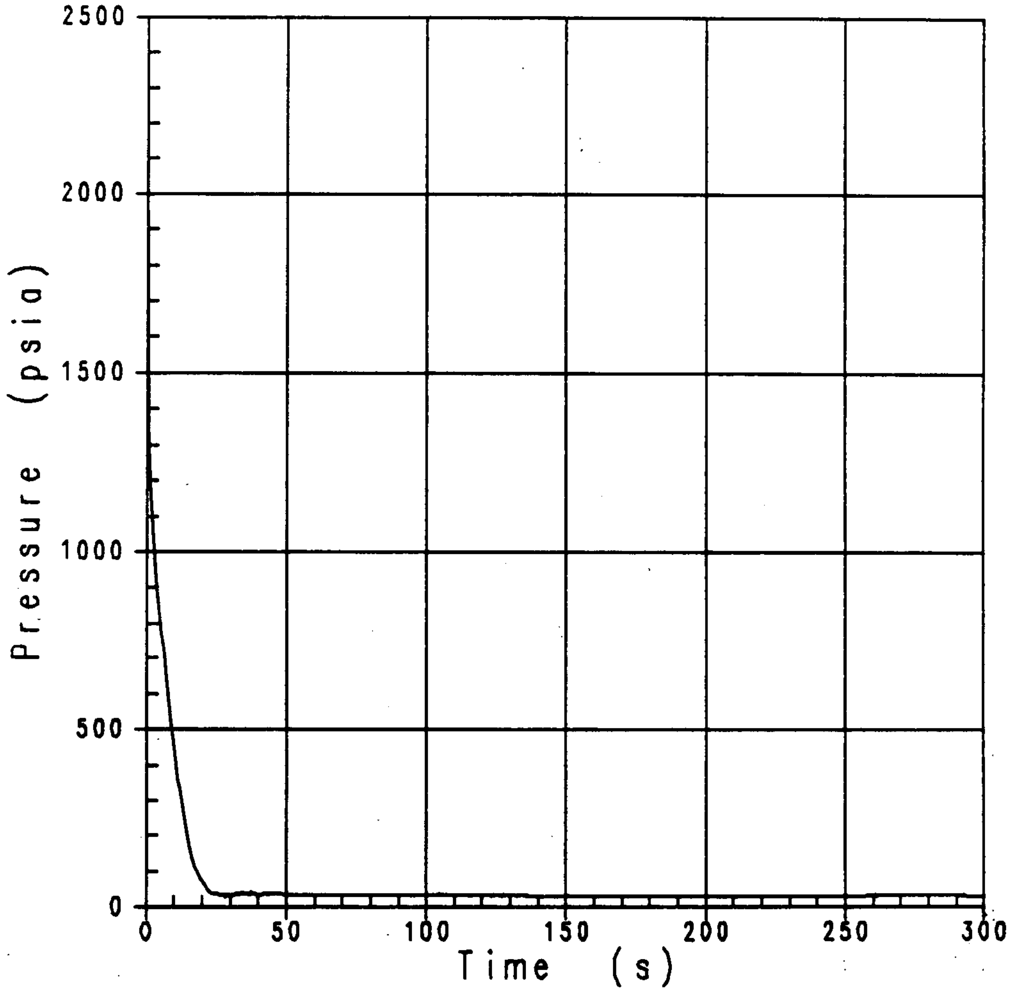

Figure 14.3.2-27

Superbounded Calculation
Vessel Inventory

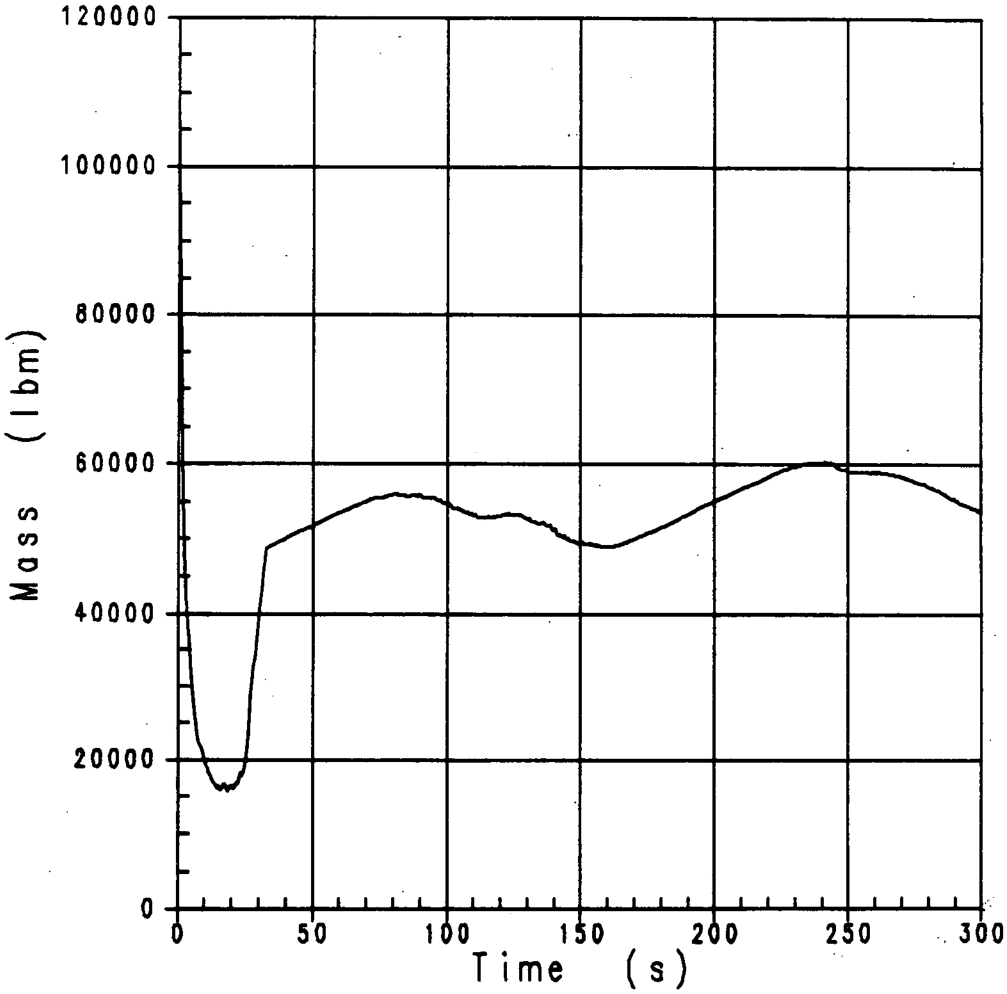

Figure 14.3.2-28

Superbounded Calculation
Loop Side Break Flow

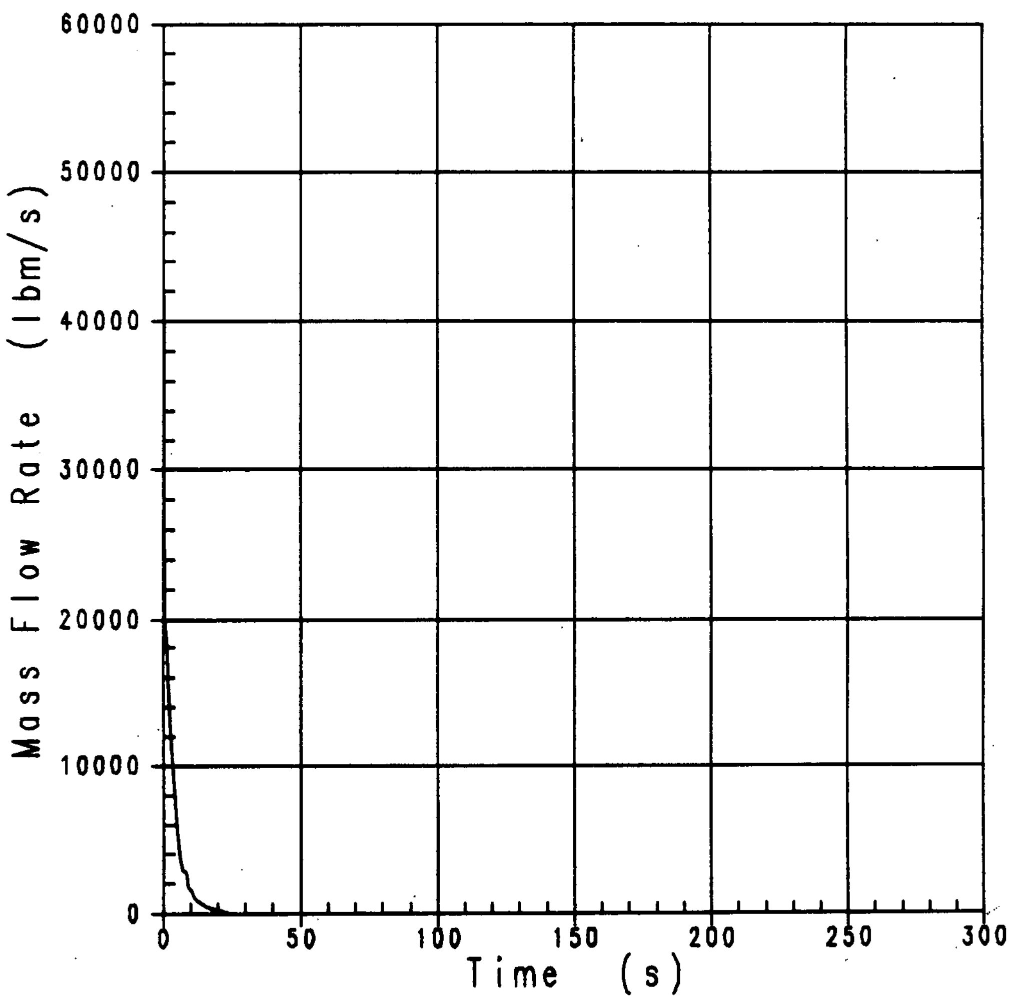

Figure 14.3.2-29

Superbounded Calculation
Vessel Side Break Flow

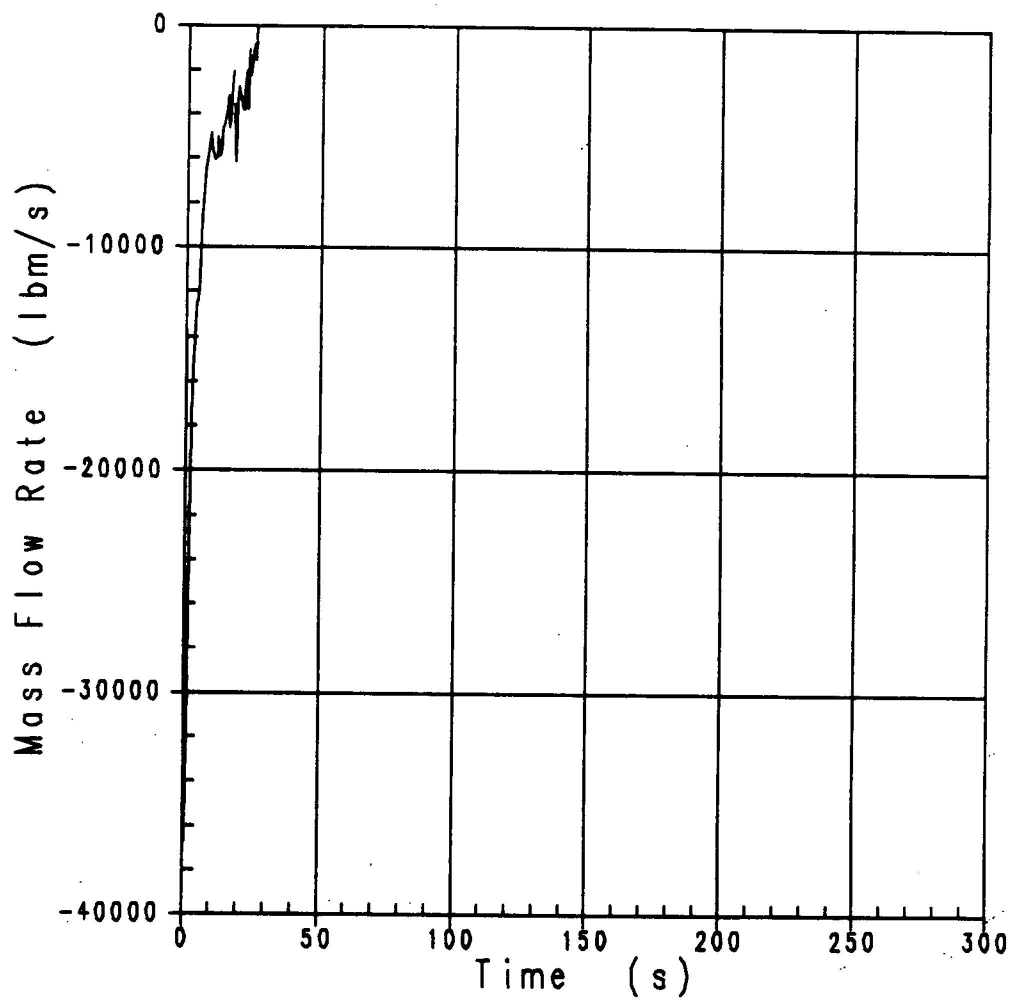

Figure 14.3.2-30

Superbounded Calculation
Accumulator Flow

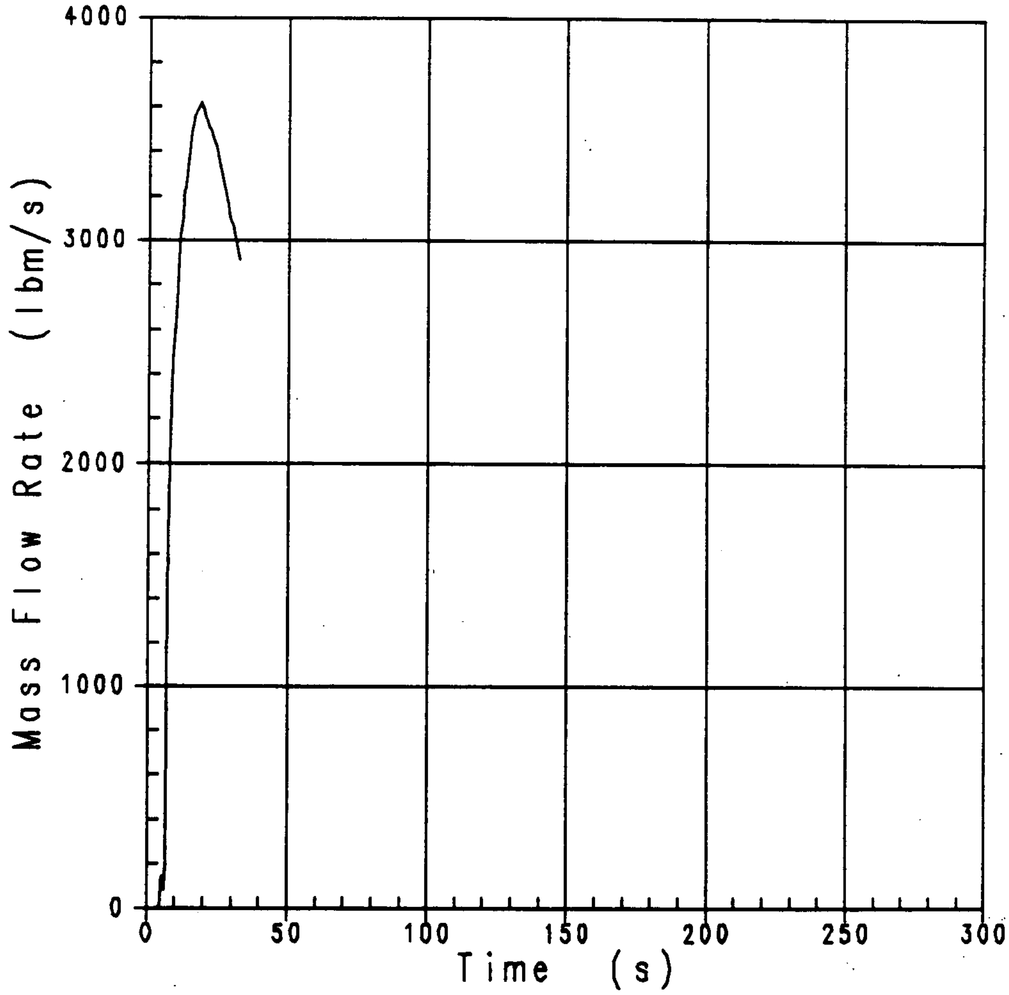

Figure 14.3.2-31

Superbounded Calculation
High Head Safety Injection Flow

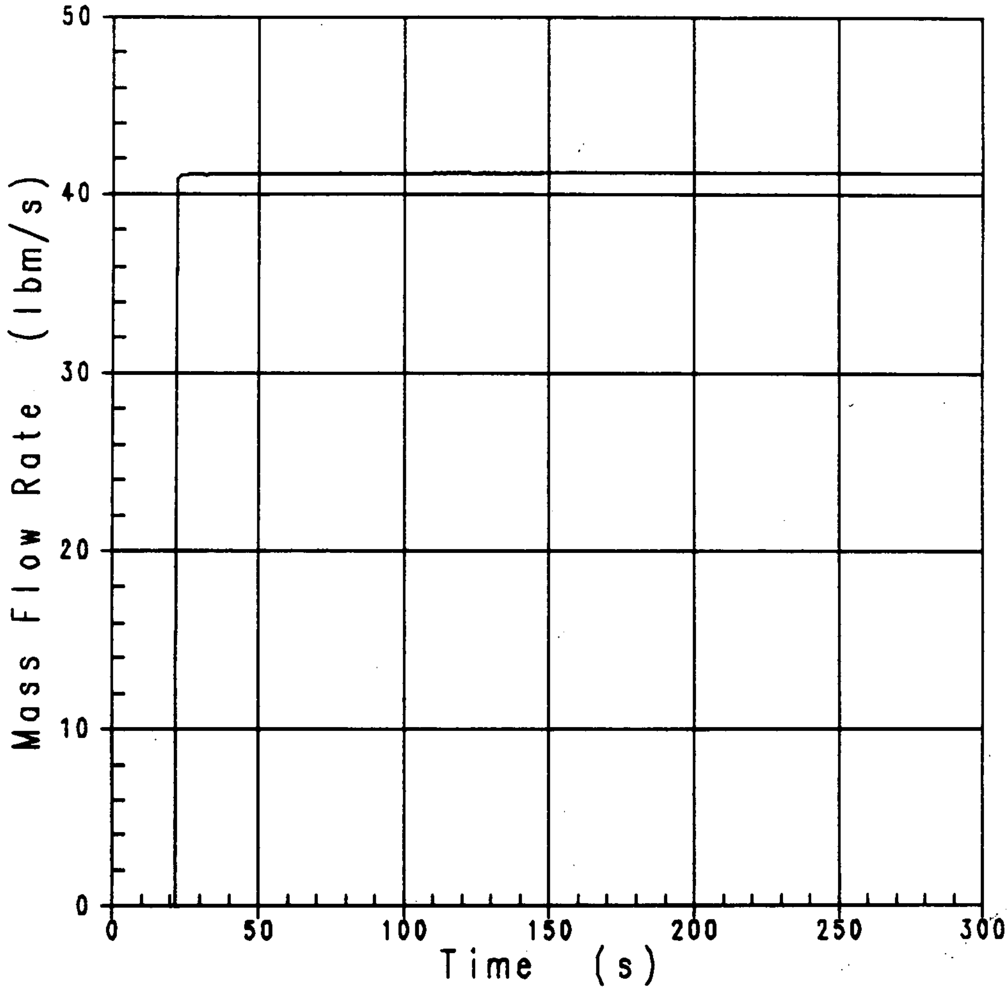

Figure 14.3.2-32

Superbounded Calculation
Low Head Safety Injection Flow

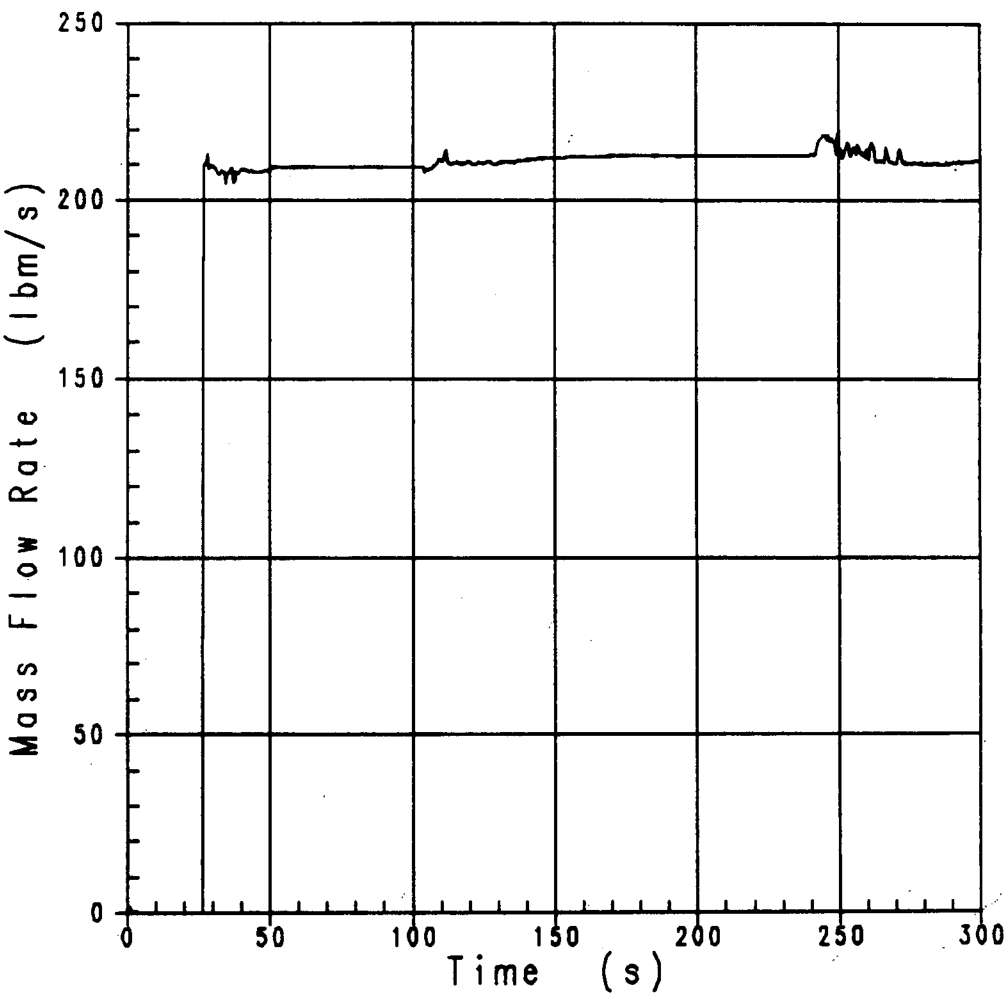

Figure 14.3.2-33

Superbounded Calculation
Liquid Flow at Top of Core Channel 10 (OH/SC)

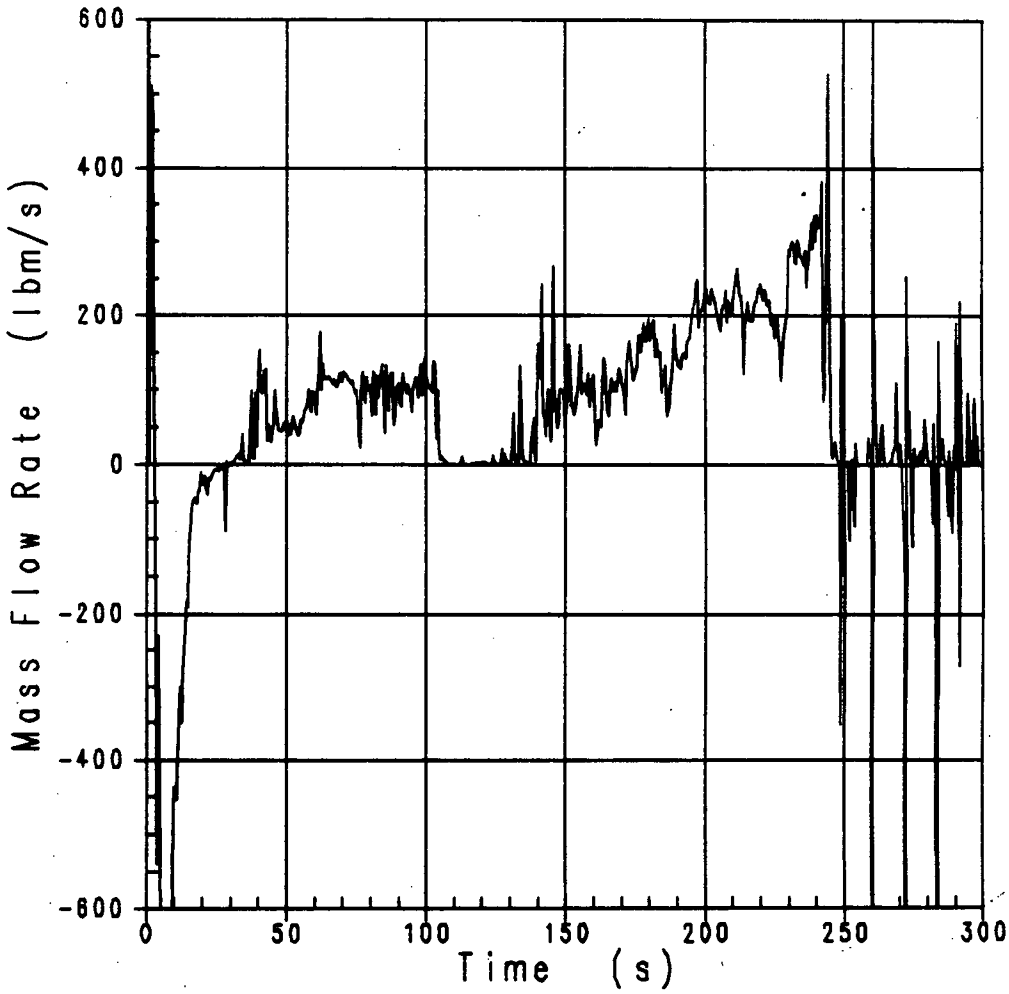

Figure 14.3.2-34

Superbounded Calculation
Vapor Flow at Top of Core Channel 10 (OH/SC)

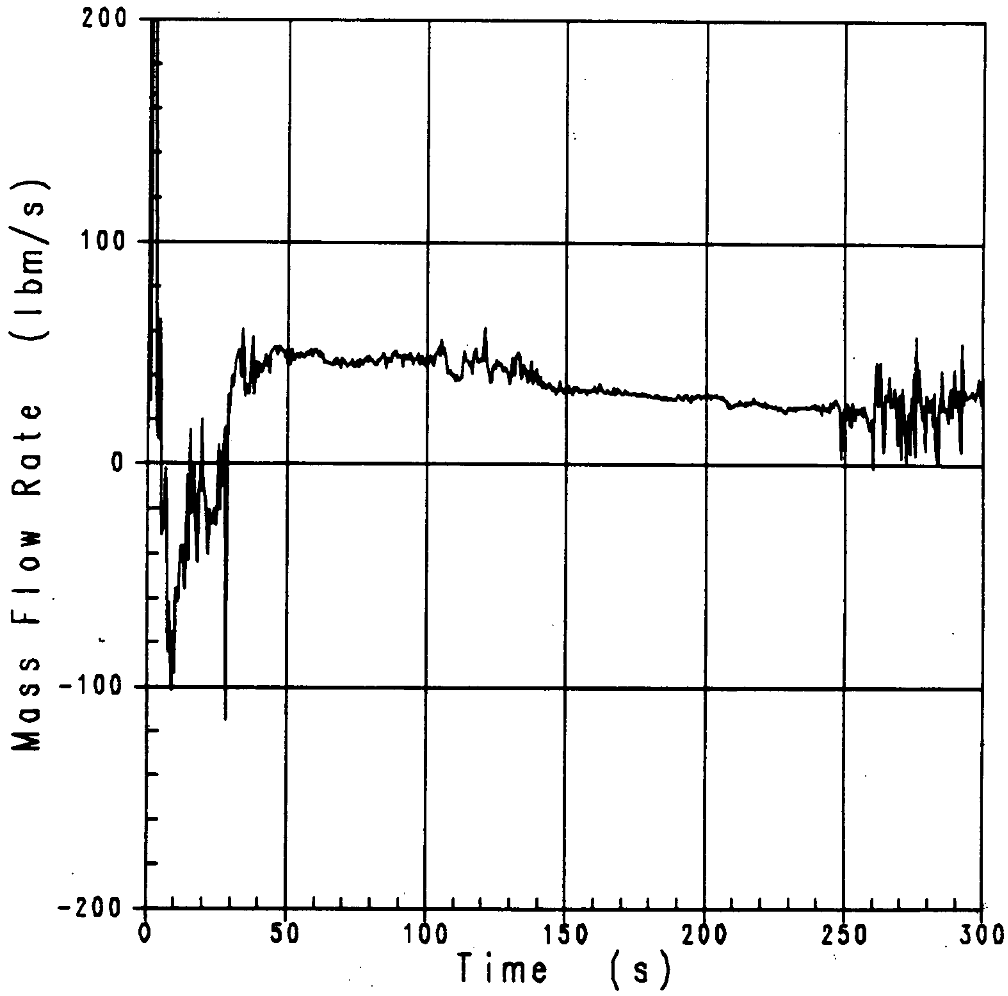

Figure 14.3.2-35

Superbounded Calculation
Liquid Flow at Top of Core Channel 11 (GT)

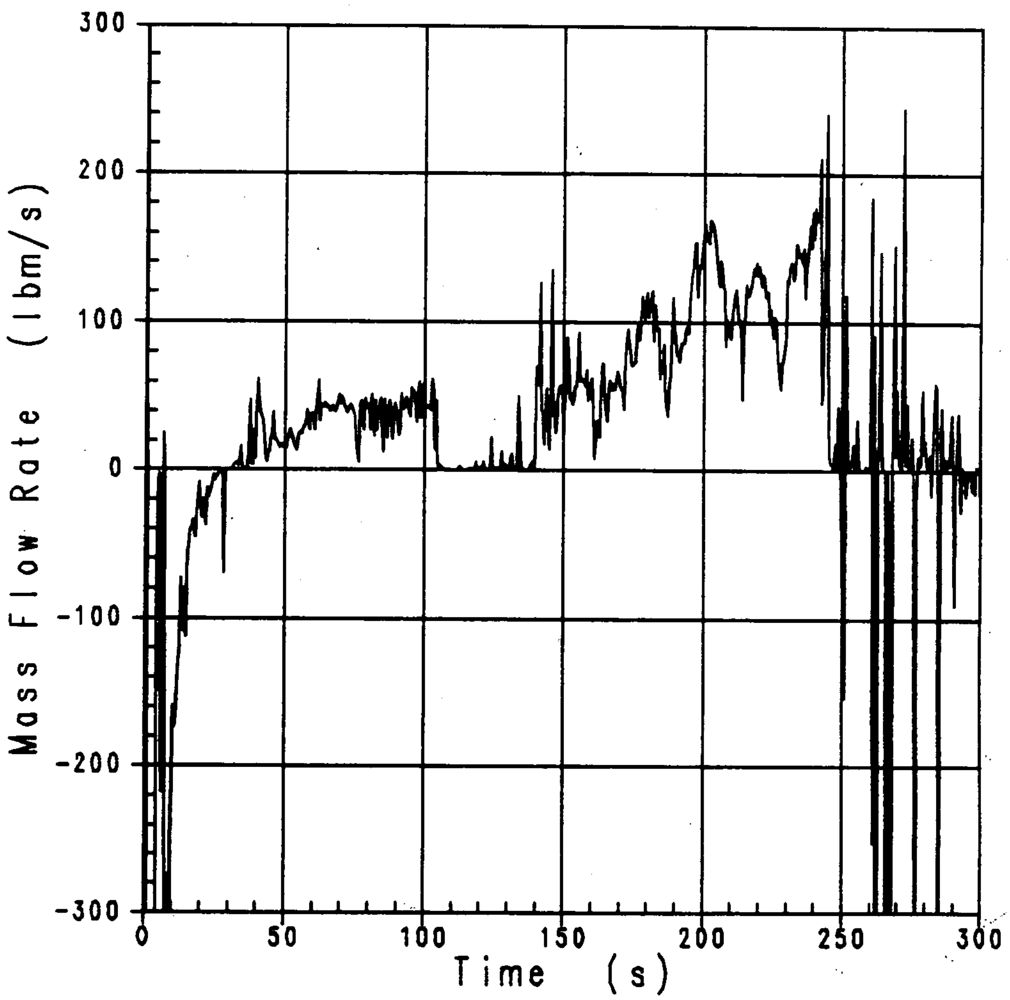

Figure 14.3.2-36

Superbounded Calculation
Vapor Flow at Top of Core Channel 11 (GT)

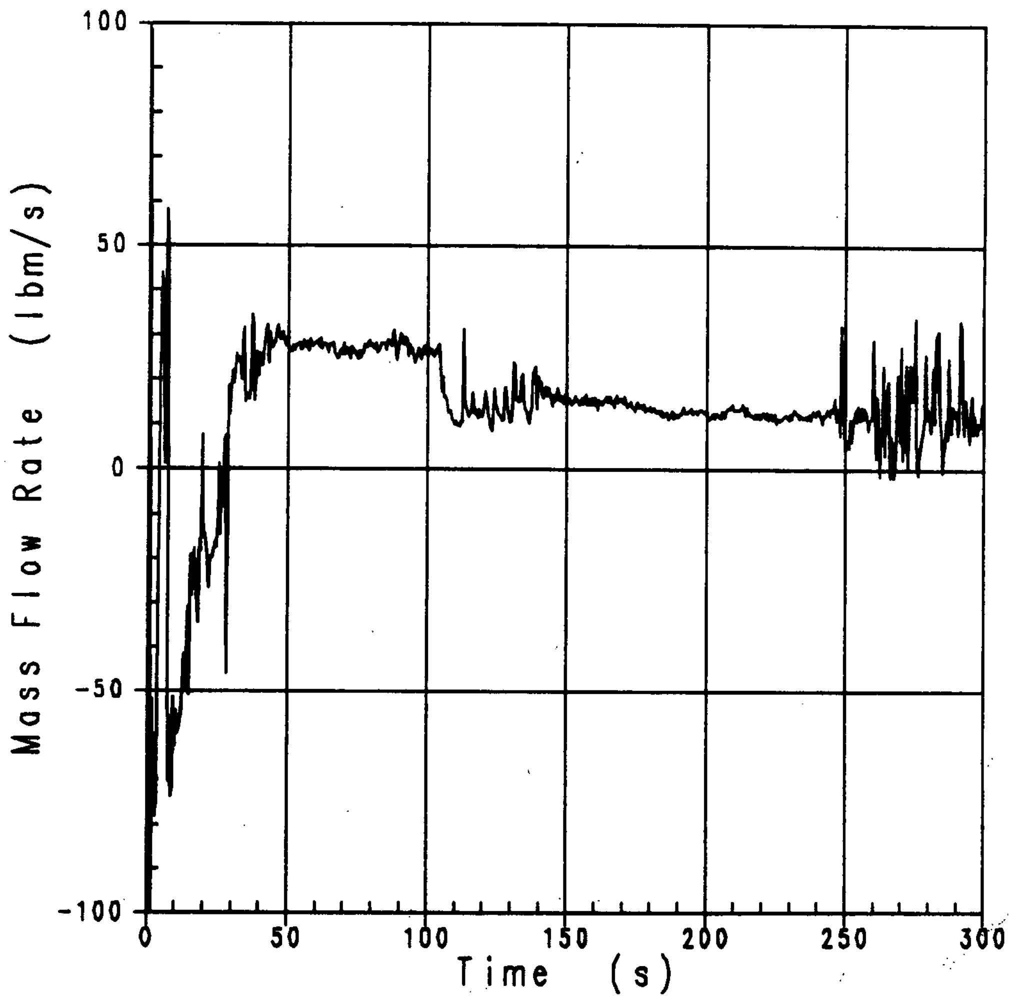

Figure 14.3.2-37

Superbounded Calculation
Liquid Flow at Top of Core Channel 12 (HA)

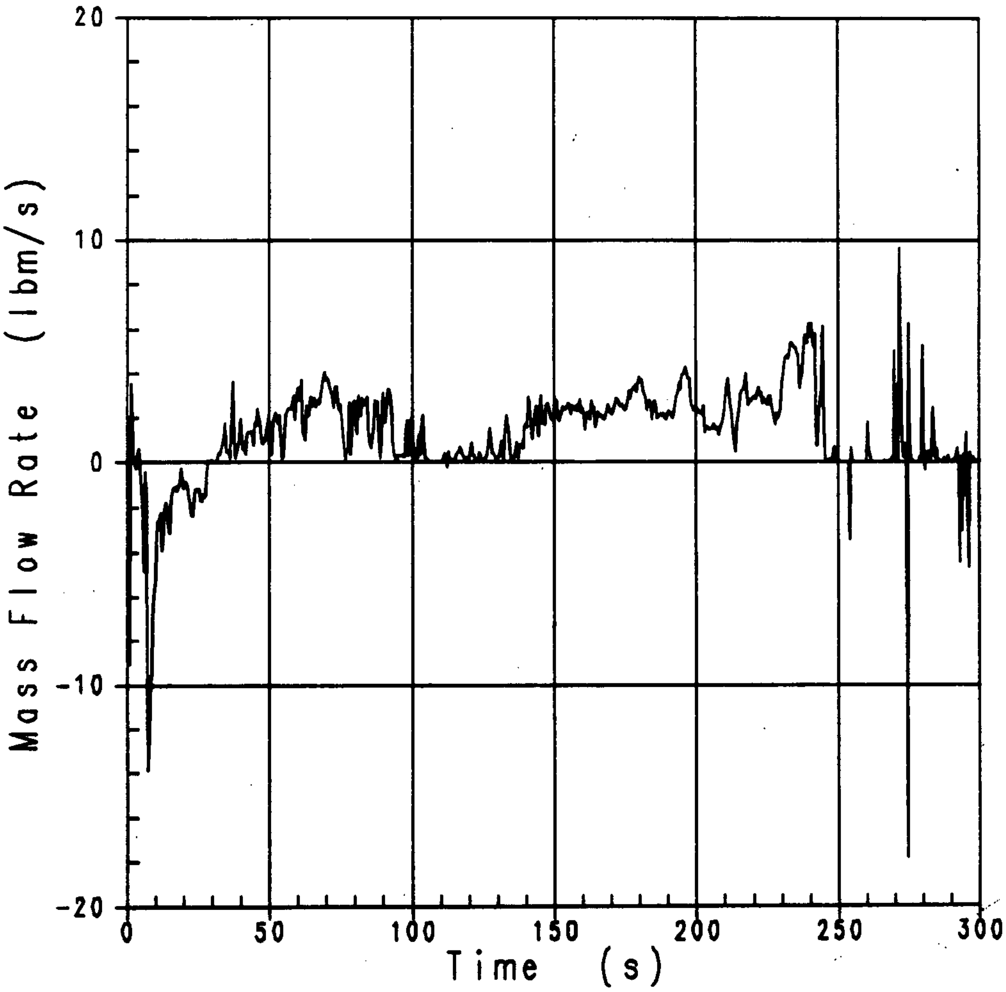

Figure 14.3.2-38

Superbounded Calculation
Vapor Flow at Top of Core Channel 12 (HA)

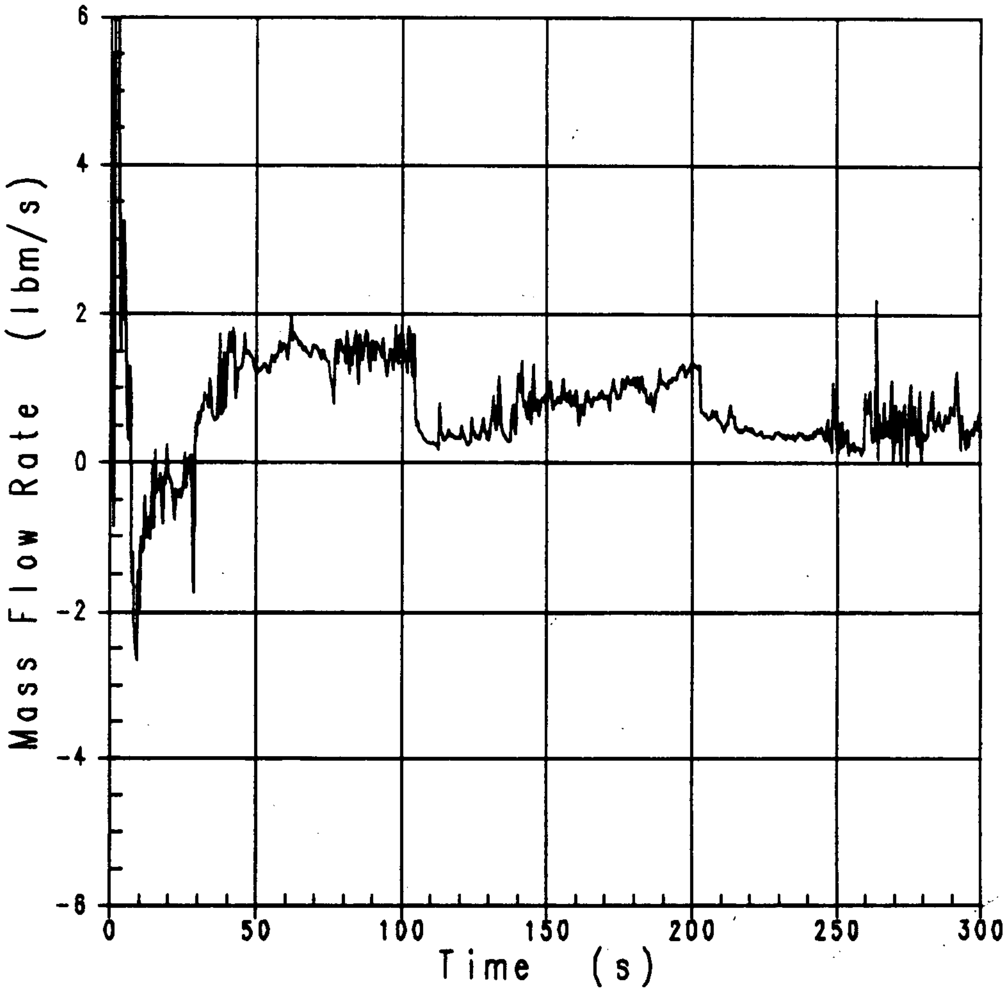

Figure 14.3.2-39

Superbounded Calculation
Liquid Flow at Top of Core Channel 13 (LP)

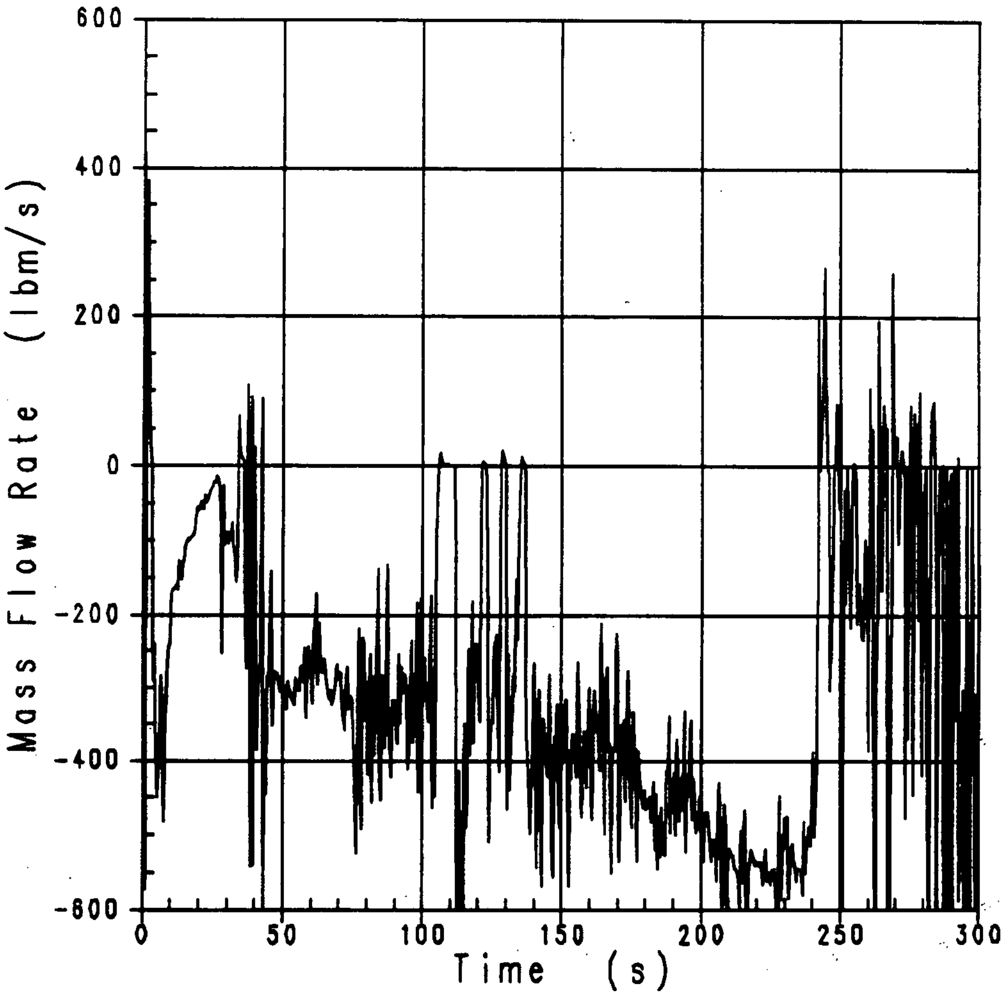

Figure 14.3.2-40

Superbounded Calculation
Vapor Flow at Top of Core Channel 13 (LP)

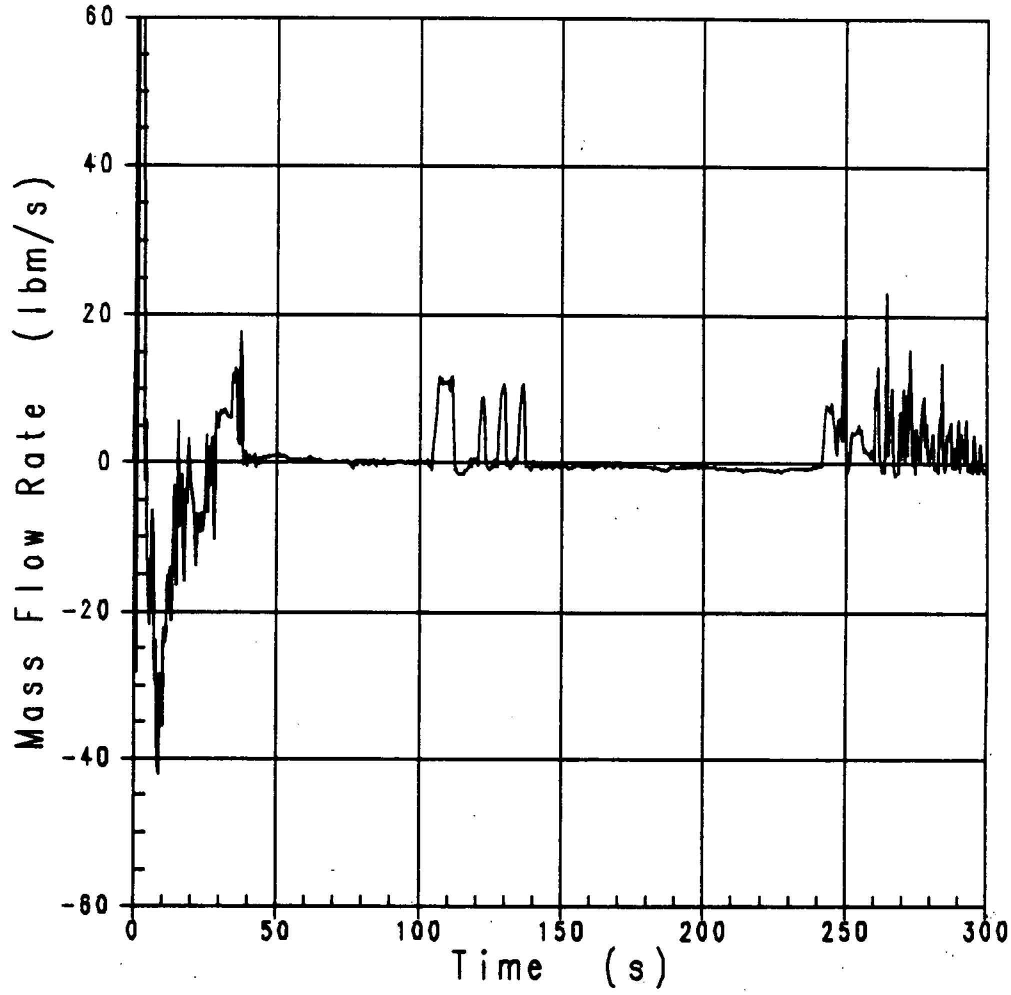

Figure 14.3.2-41

Superbounded Calculation
Downcomer Liquid Level

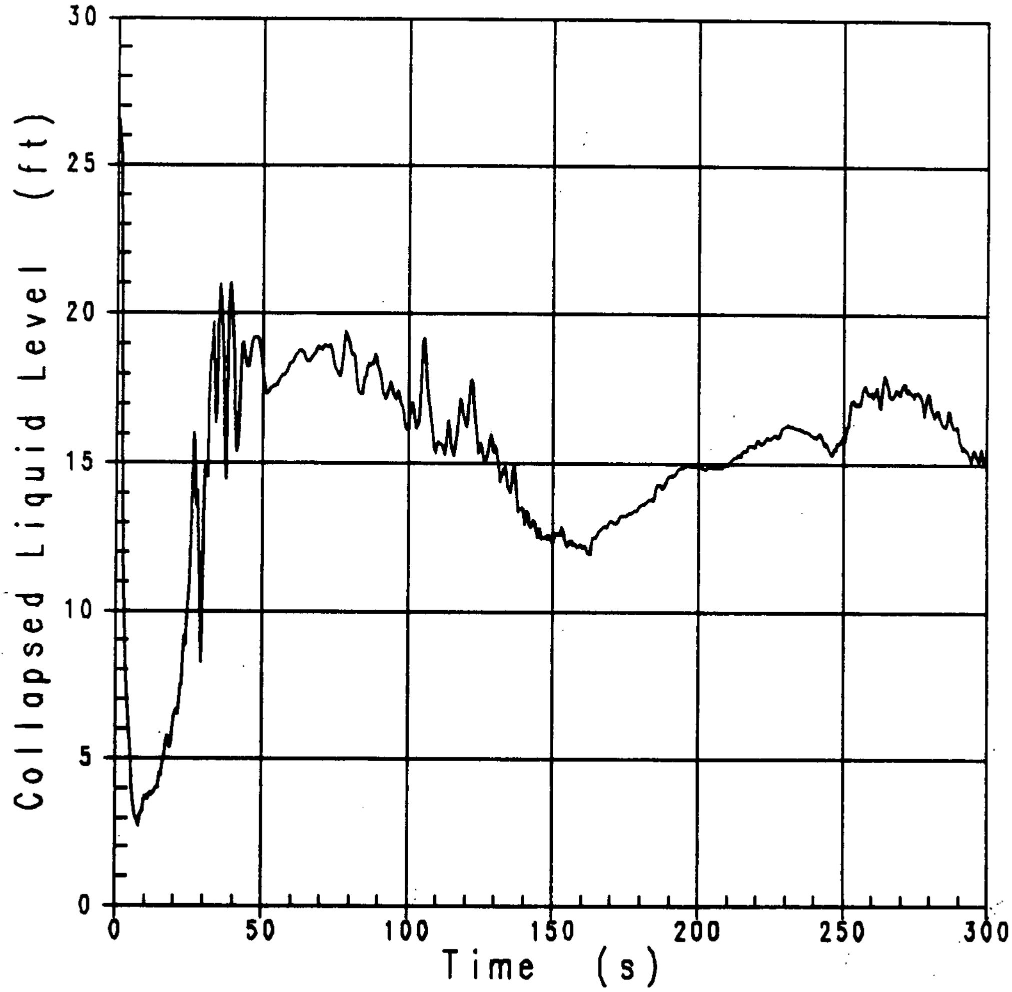

Figure 14.3.2-42

Superbounded Calculation
Lower Plenum Liquid Level

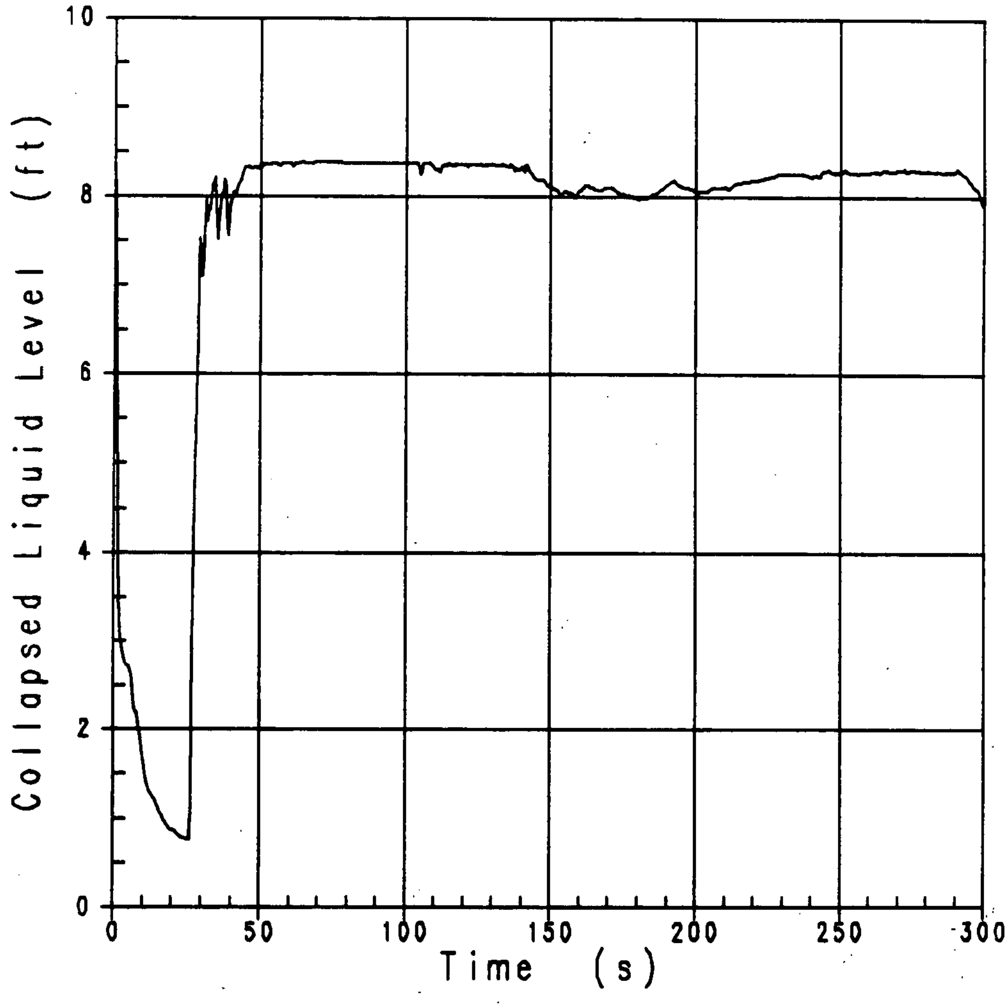

Figure 14.3.2-43

Superbounded Calculation
Core Liquid Level

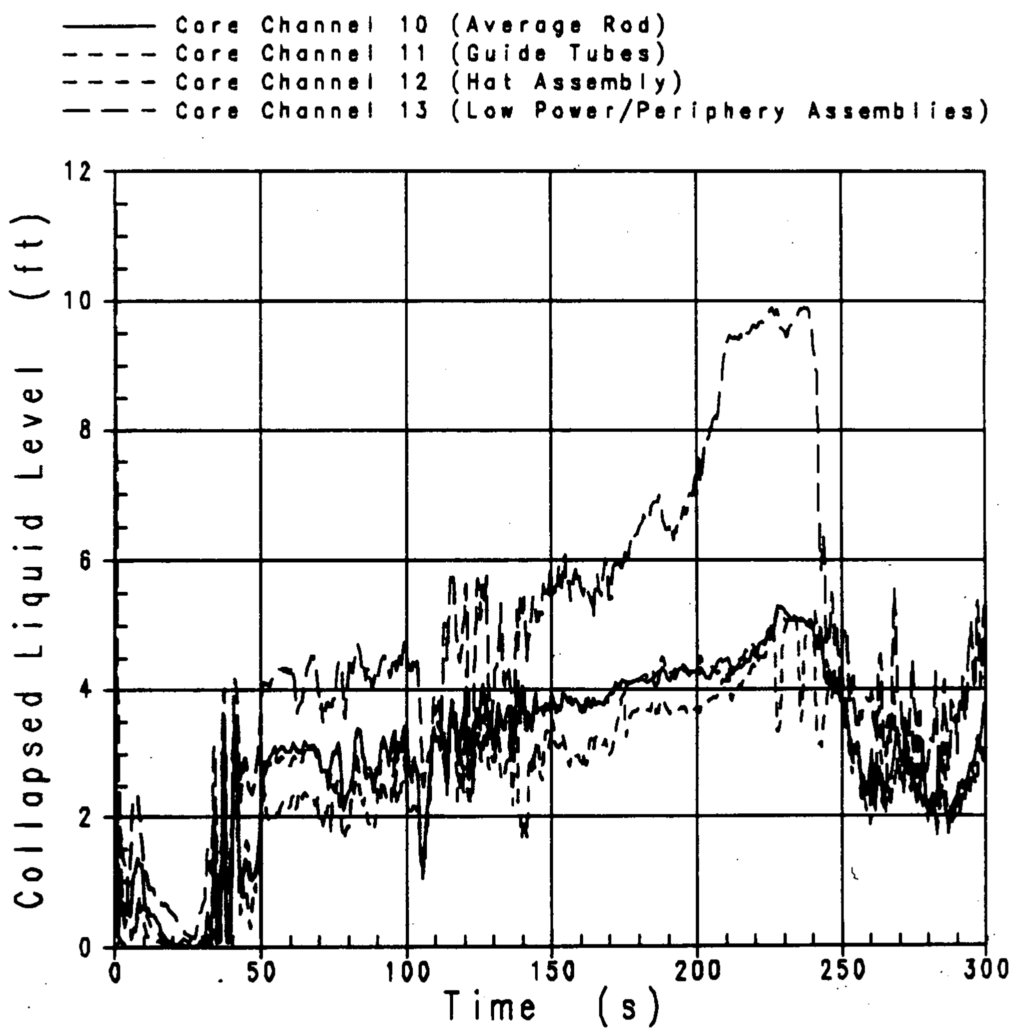

Figure 14.3.2-44

Superbounded Calculation
Void Fraction in Lower Plenum

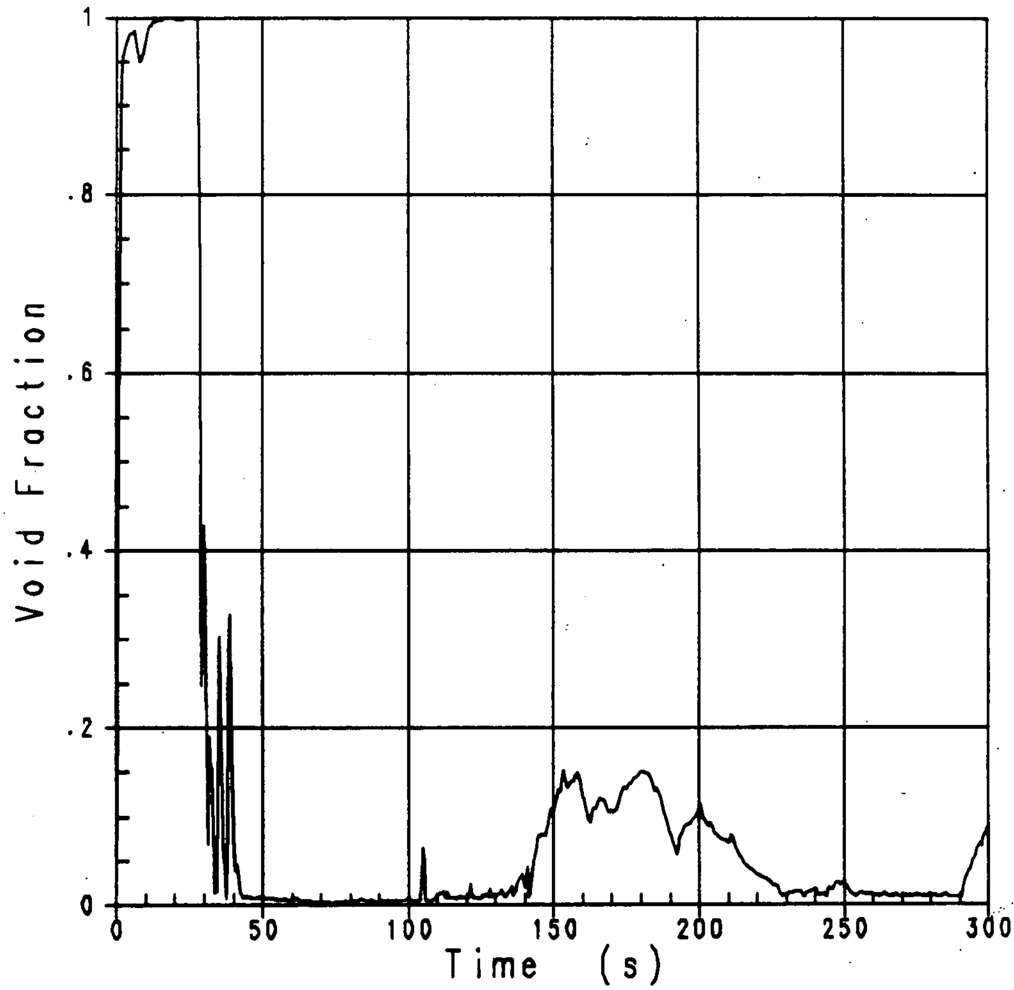

Figure 14.3.2-45

Superbounded Calculation
Void Fraction at Bottom of Core

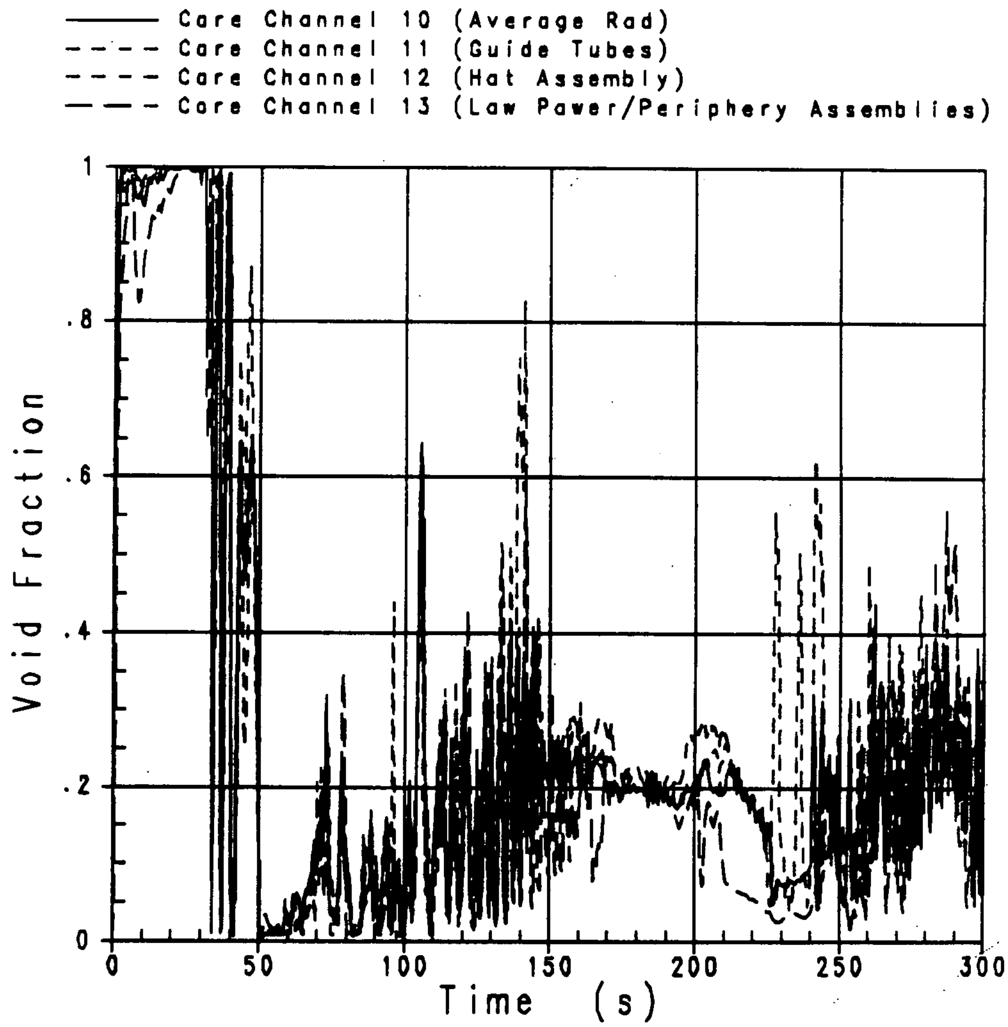

Figure 14.3.2-46

Nominal Calculation
Peak Cladding Temperature

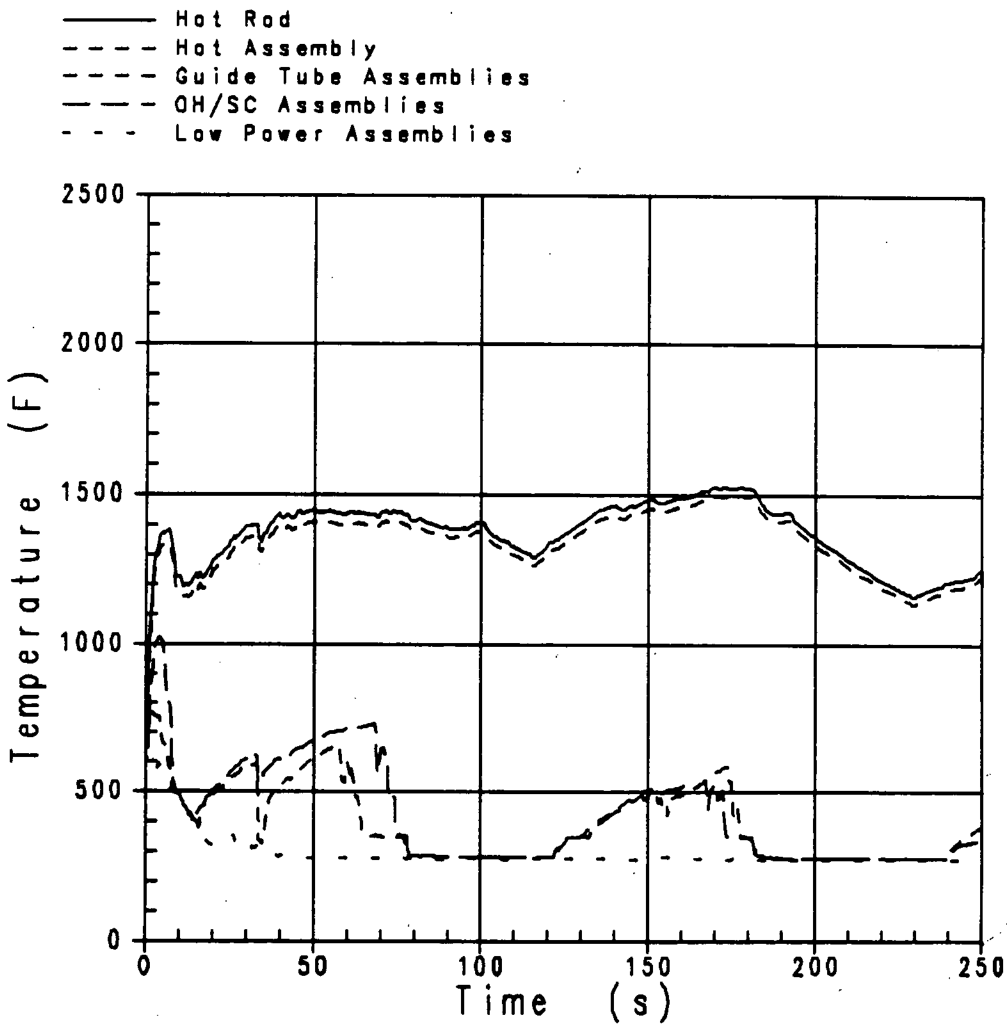

Figure 14.3.2-47

Nominal Calculation
Core Pressure

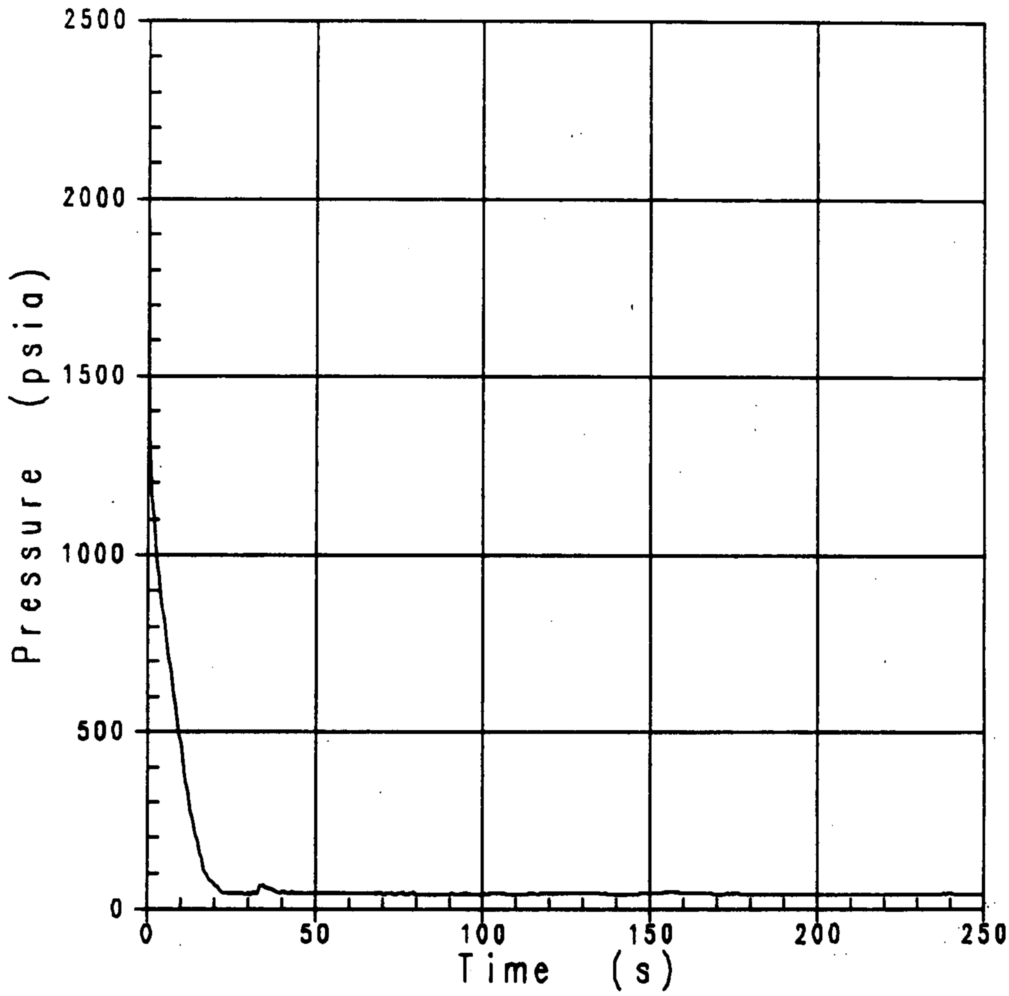

Figure 14.3.2-48

Nominal Calculation
Vessel Inventory

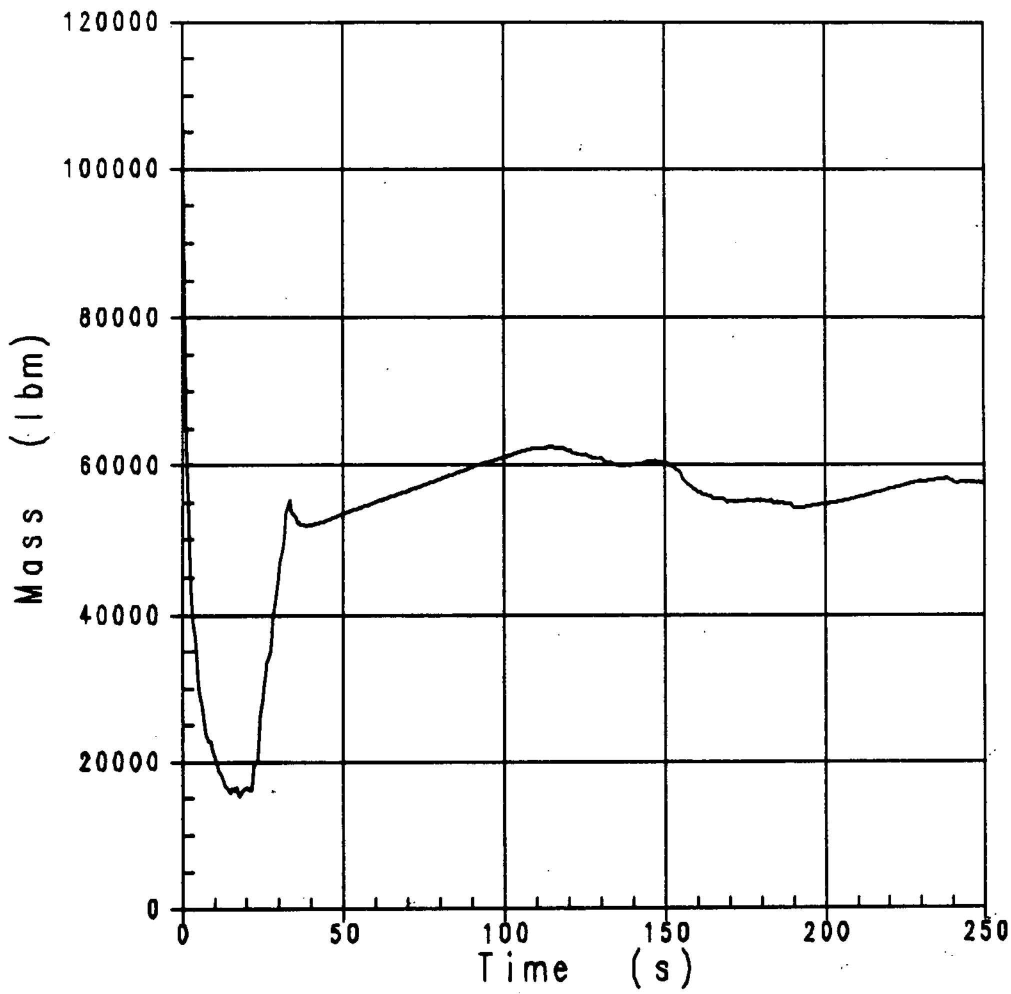

Figure 14.3.2-49

Nominal Calculation
Loop Side Break Flow

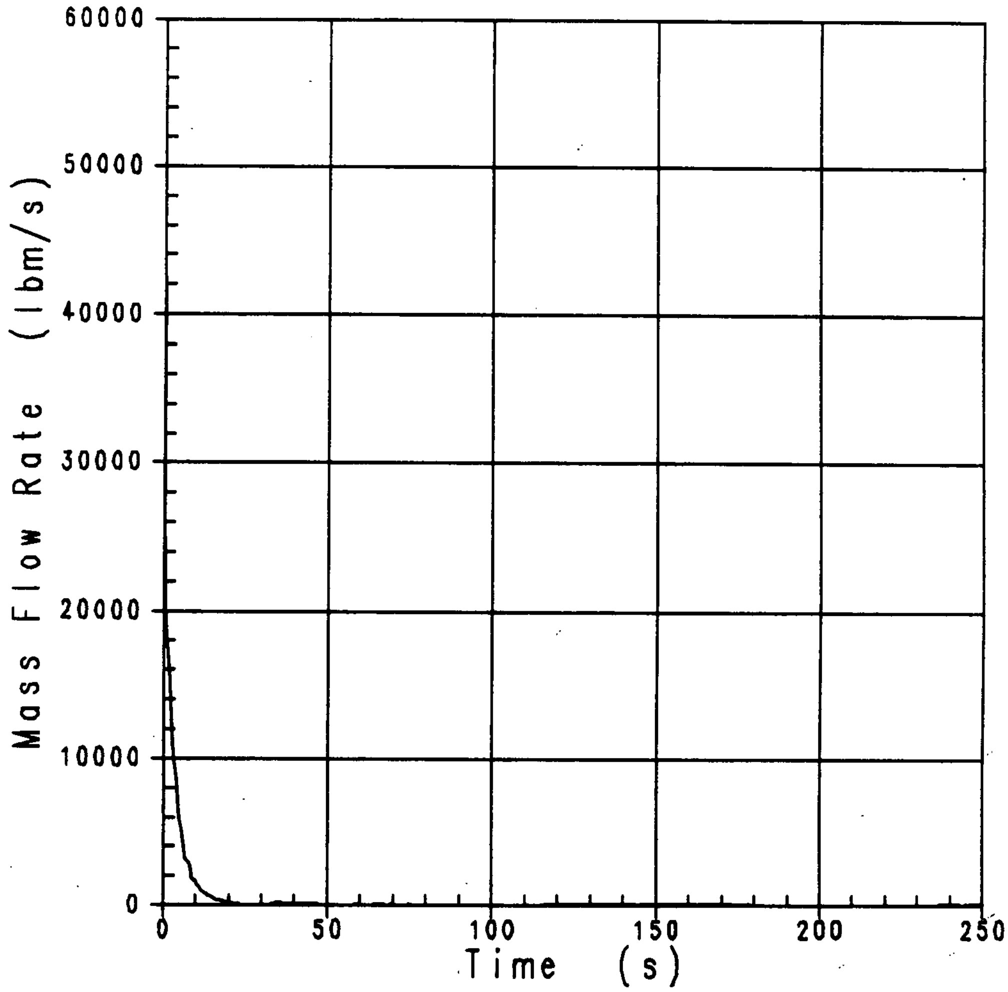

Figure 14.3.2-50

Nominal Calculation
Vessel Side Break Flow

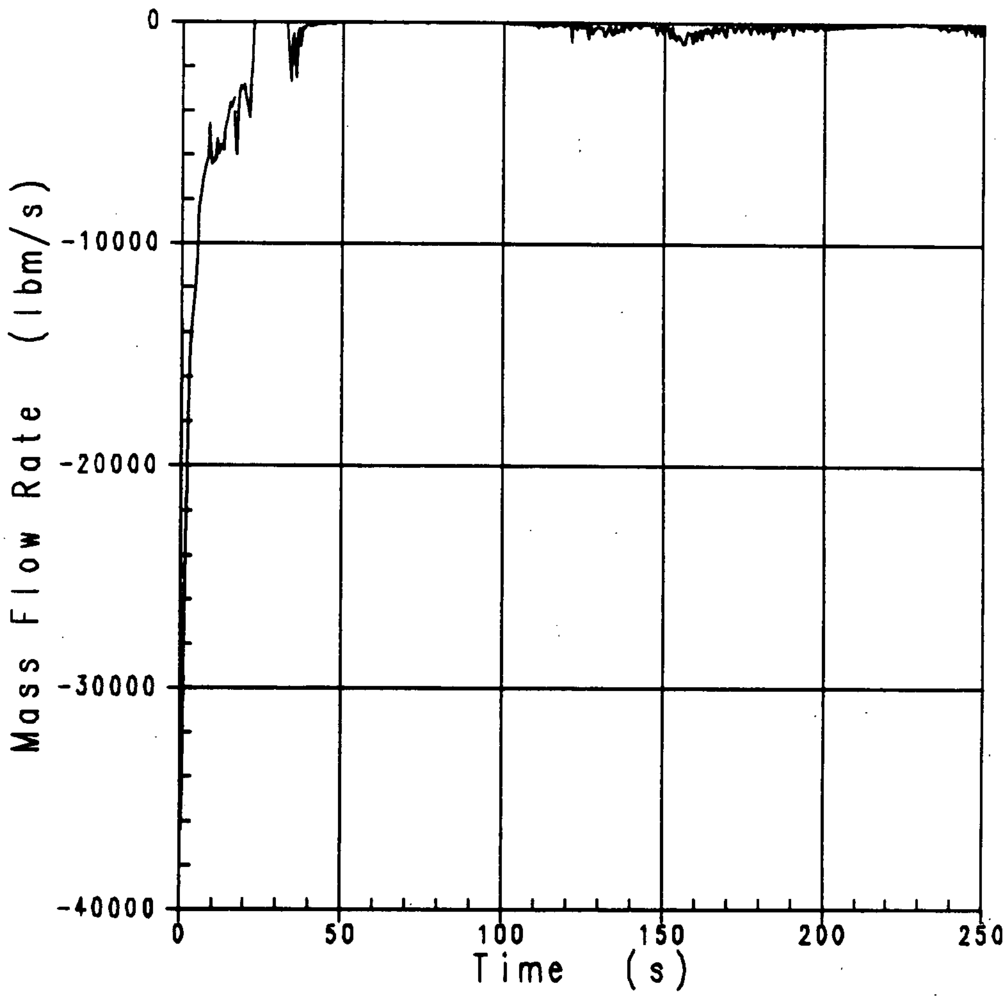

Figure 14.3.2-51

Nominal Calculation
Accumulator Flow

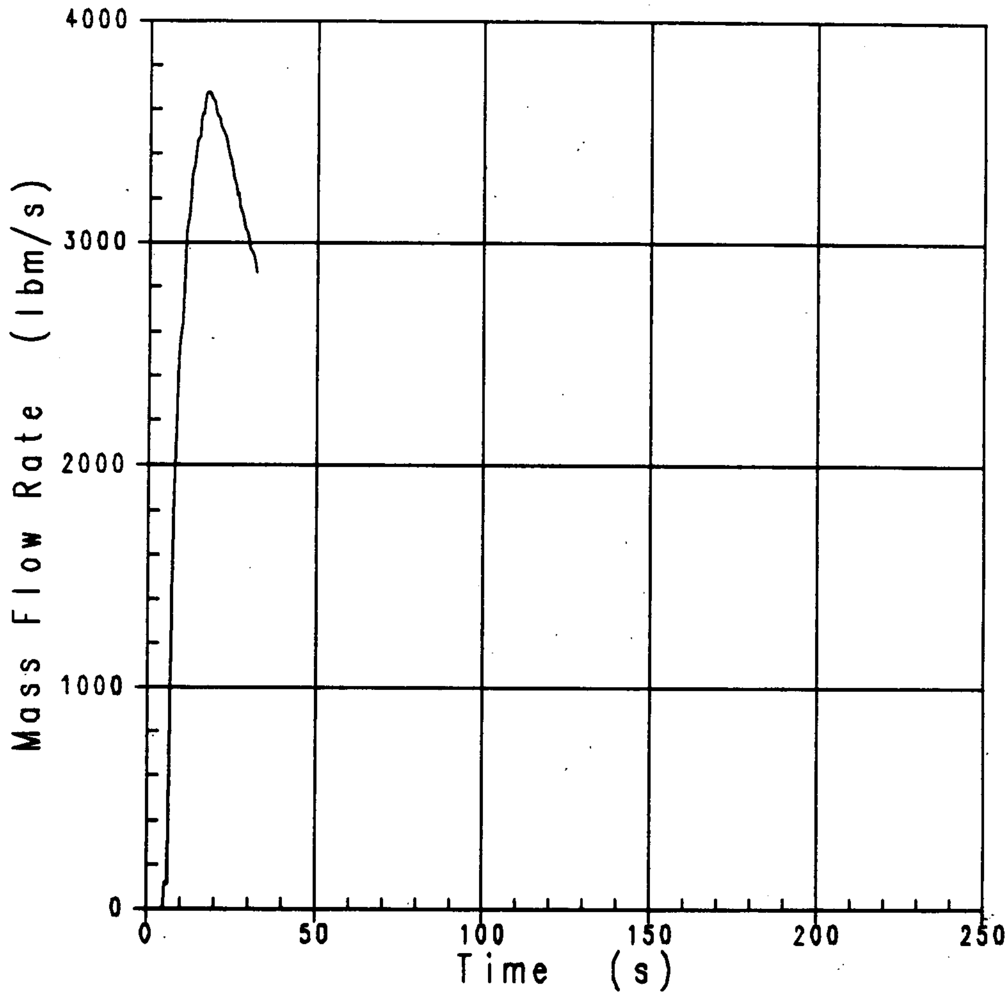

Figure 14.3.2-52

Nominal Calculation
High Head Safety Injection Flow

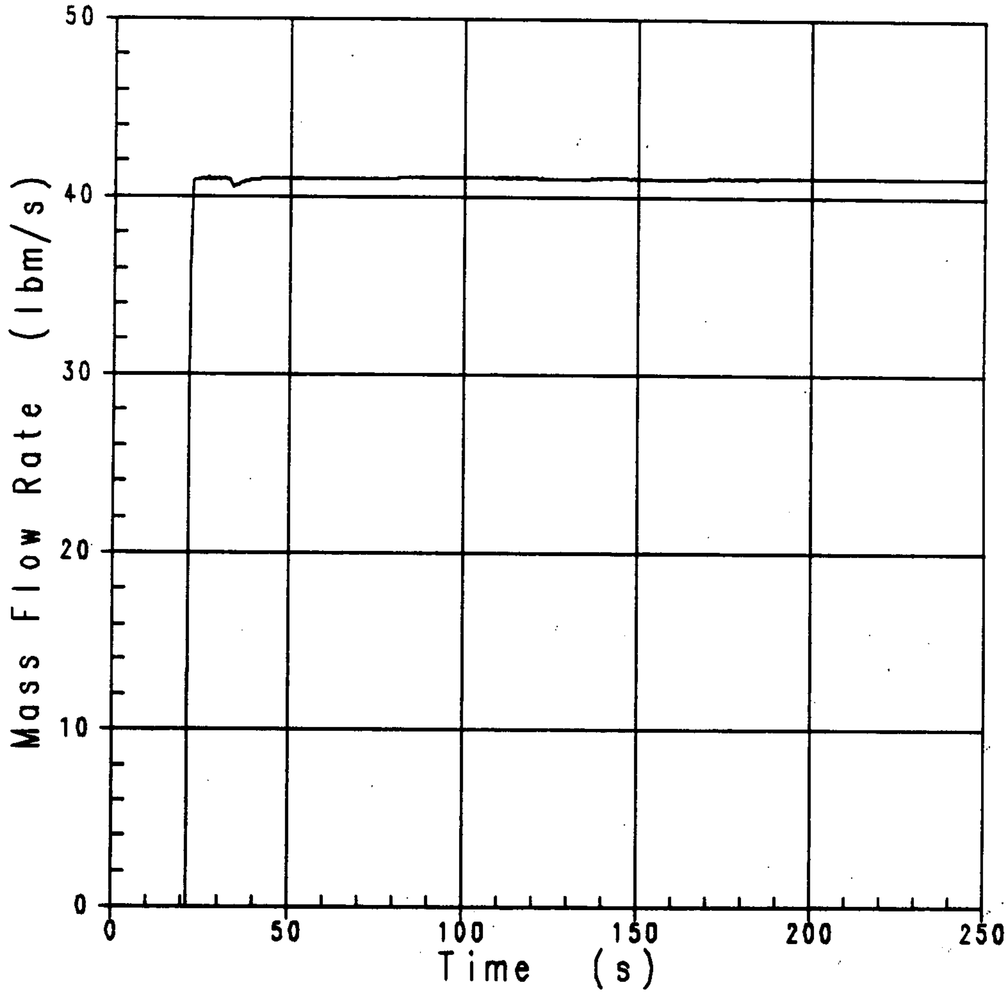

Figure 14.3.2-53

Nominal Calculation
Low Head Safety Injection Flow

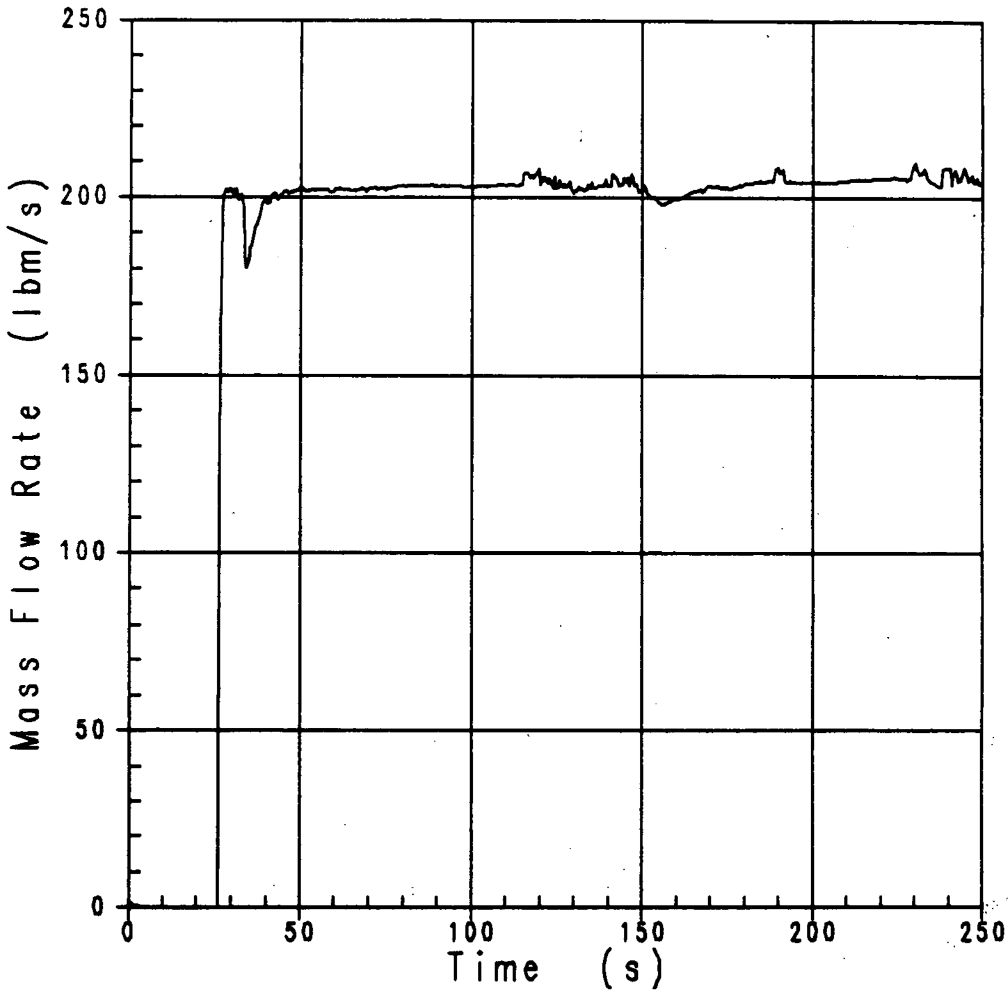

Figure 14.3.2-54

Nominal Calculation
Liquid Flow at Top of Core Channel 10 (OH/SC)

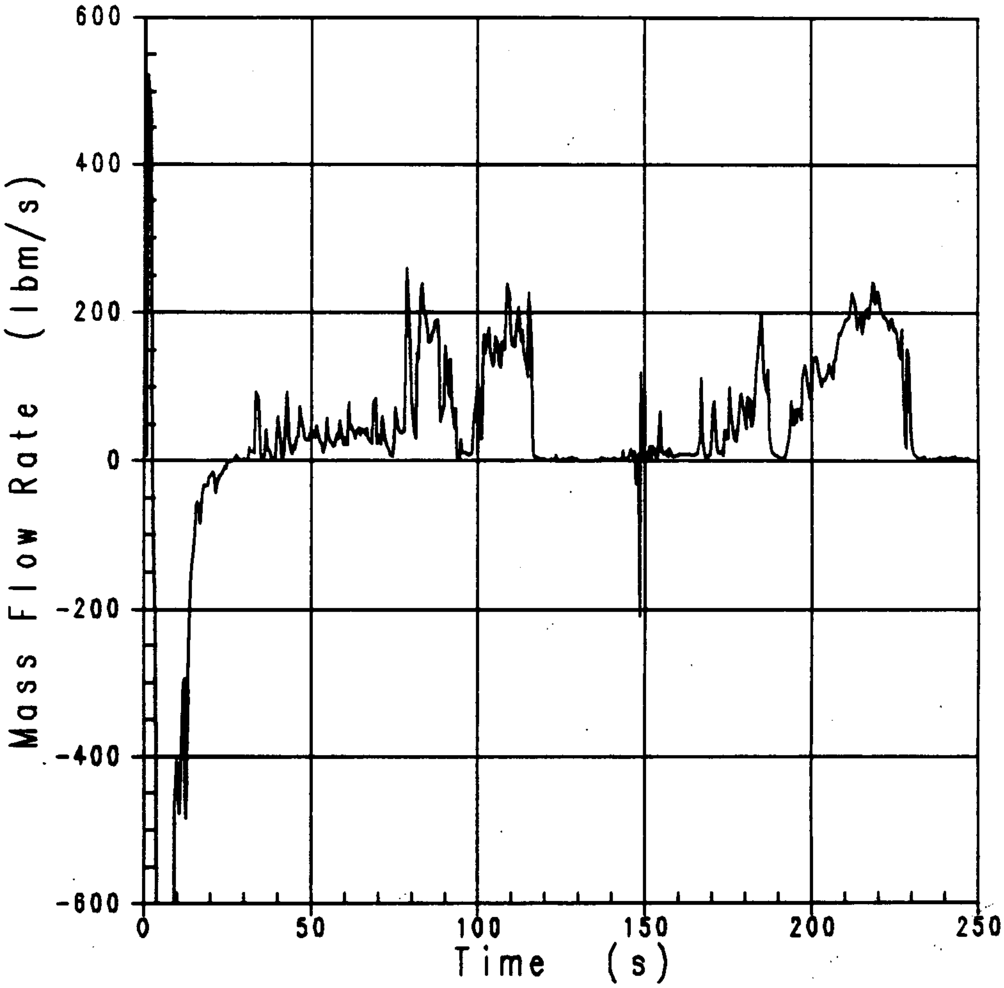

Figure 14.3.2-55

Nominal Calculation
Vapor Flow at Top of Core Channel 10 (OH/SC)

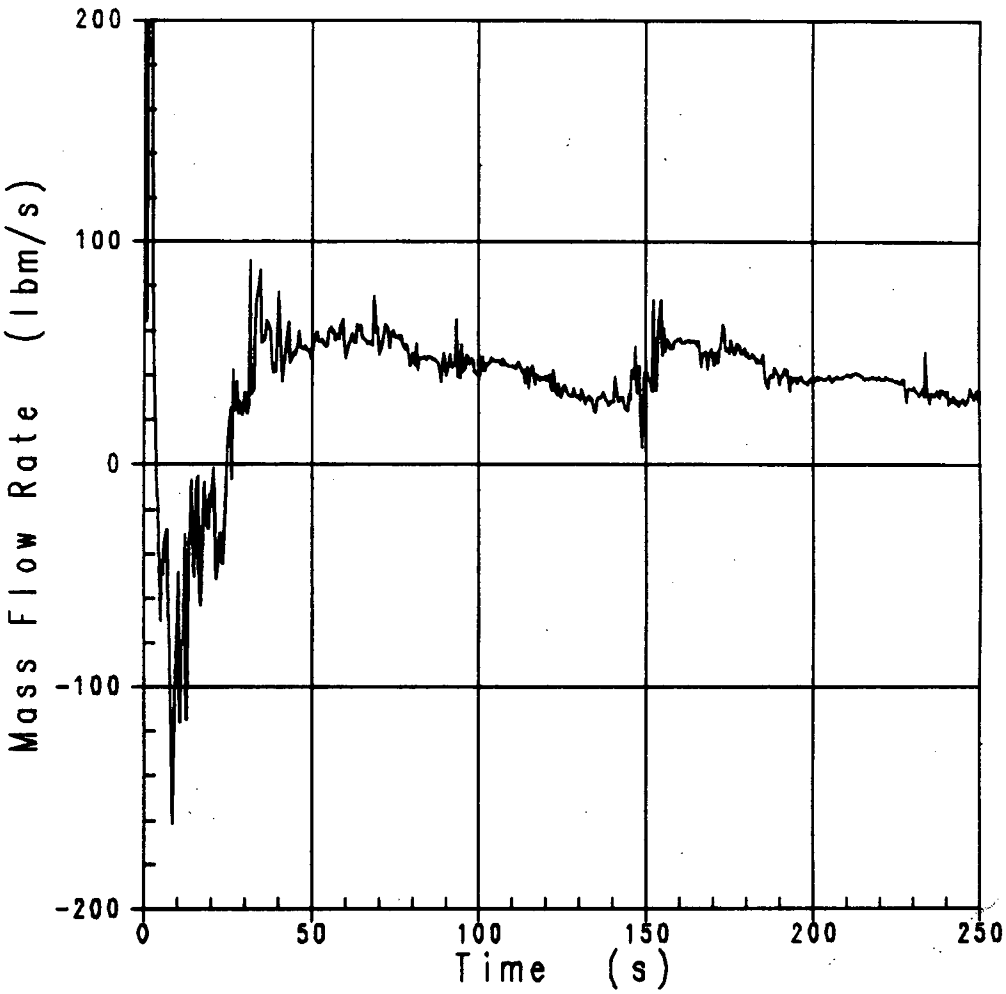

Figure 14.3.2-56

Nominal Calculation
Liquid Flow at Top of Core Channel 11 (GT)

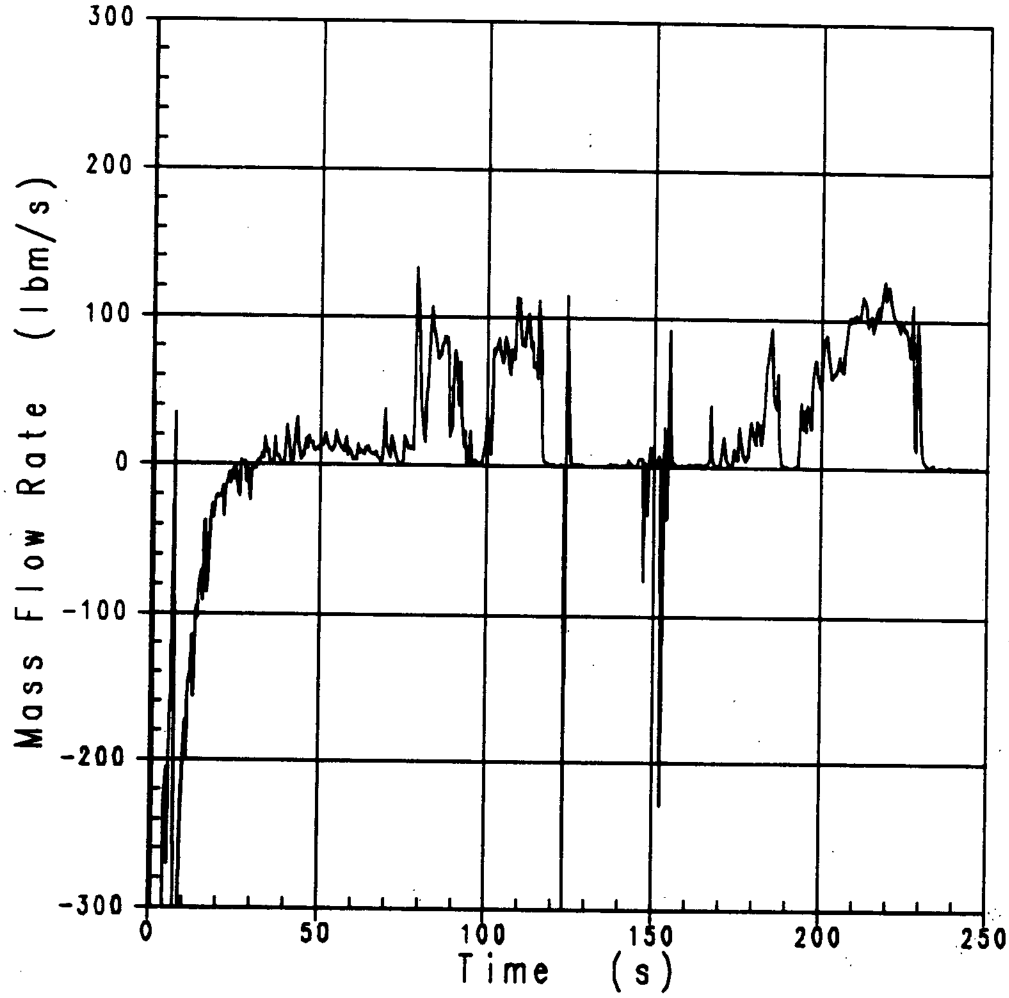

Figure 14.3.2-57

Nominal Calculation
Vapor Flow at Top of Core Channel 11 (GT)

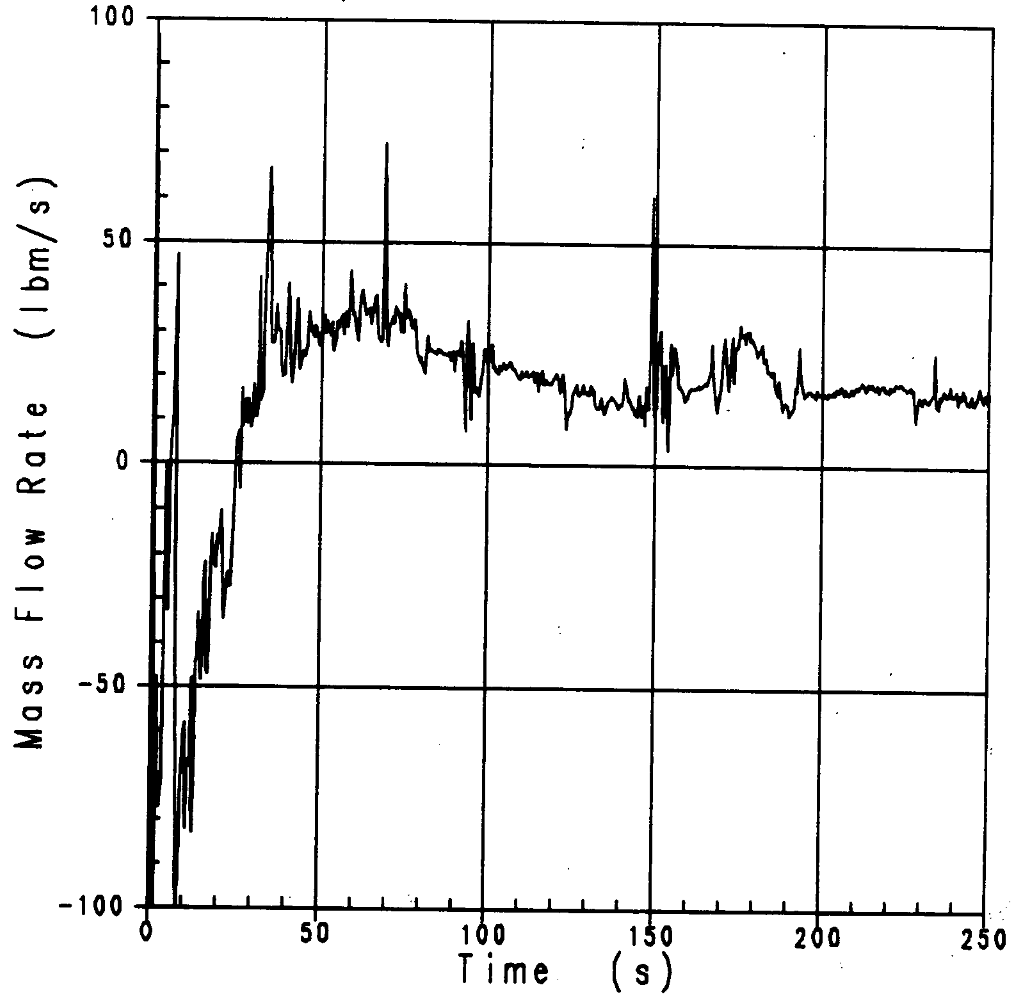

Figure 14.3.2-58

Nominal Calculation
Liquid Flow at Top of Core Channel 12 (HA)

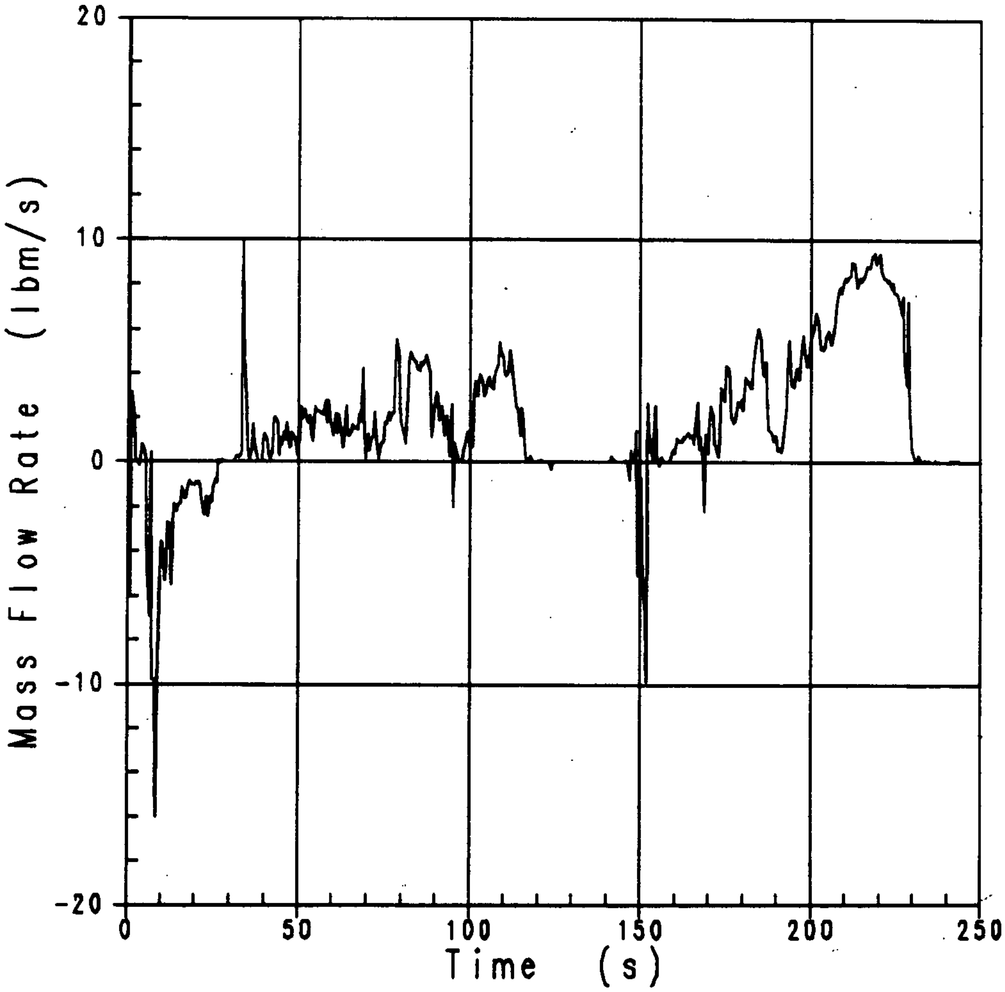

Figure 14.3.2-59

Nominal Calcnlation
Vapor Flow at Top of Core Channel 12 (HA)

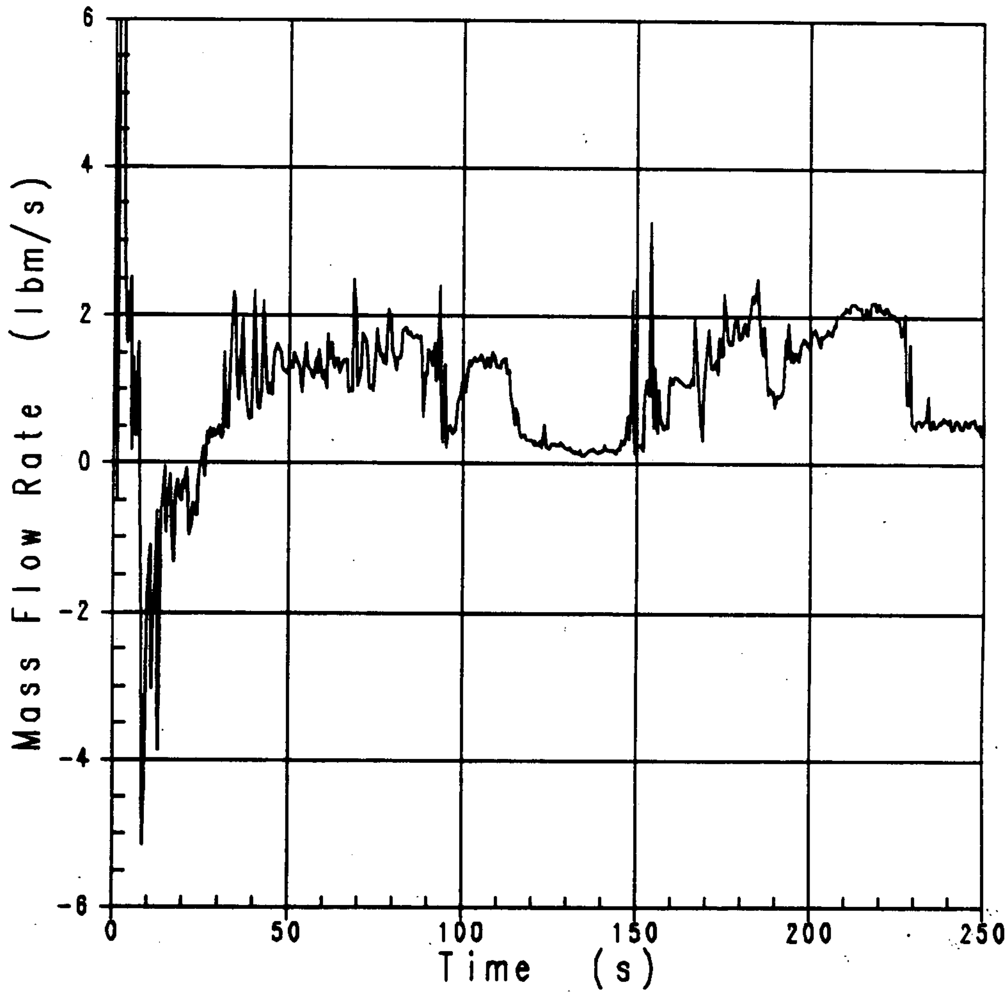

Figure 14.3.2-60

Nominal Calculation
Liquid Flow at Top of Core Channel 13 (LP)

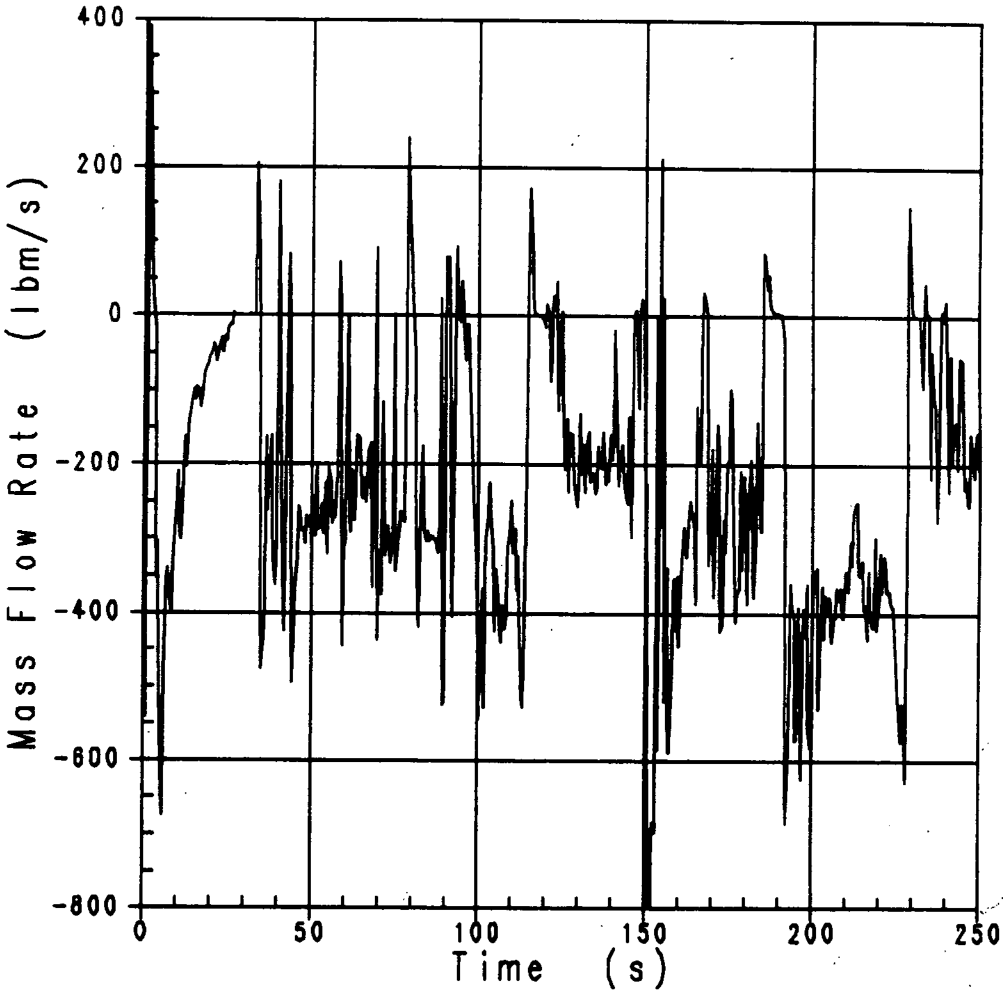

Figure 14.3.2-61

Nominal Calculation
Vapor Flow at Top of Core Channel 13 (LP)

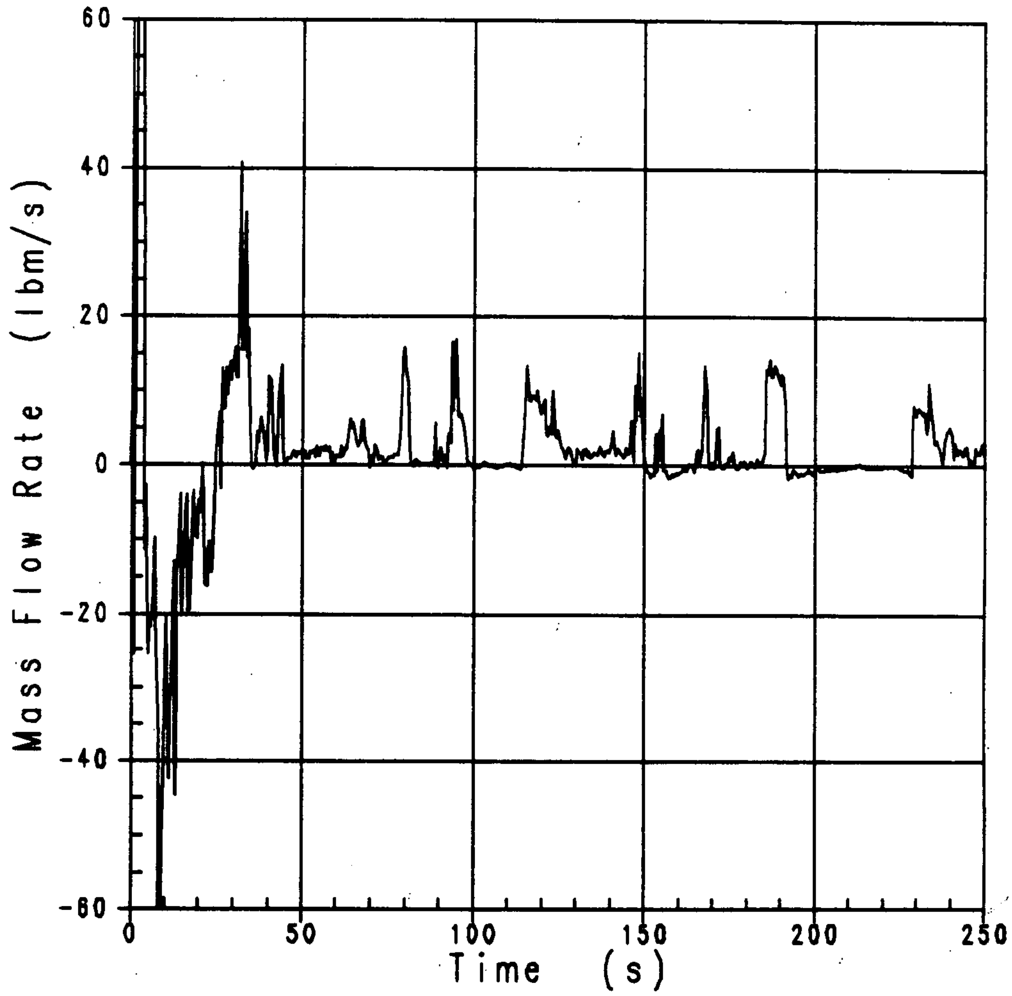

Figure 14.3.2-62

Nominal Calculation
Downcomer Liquid Level

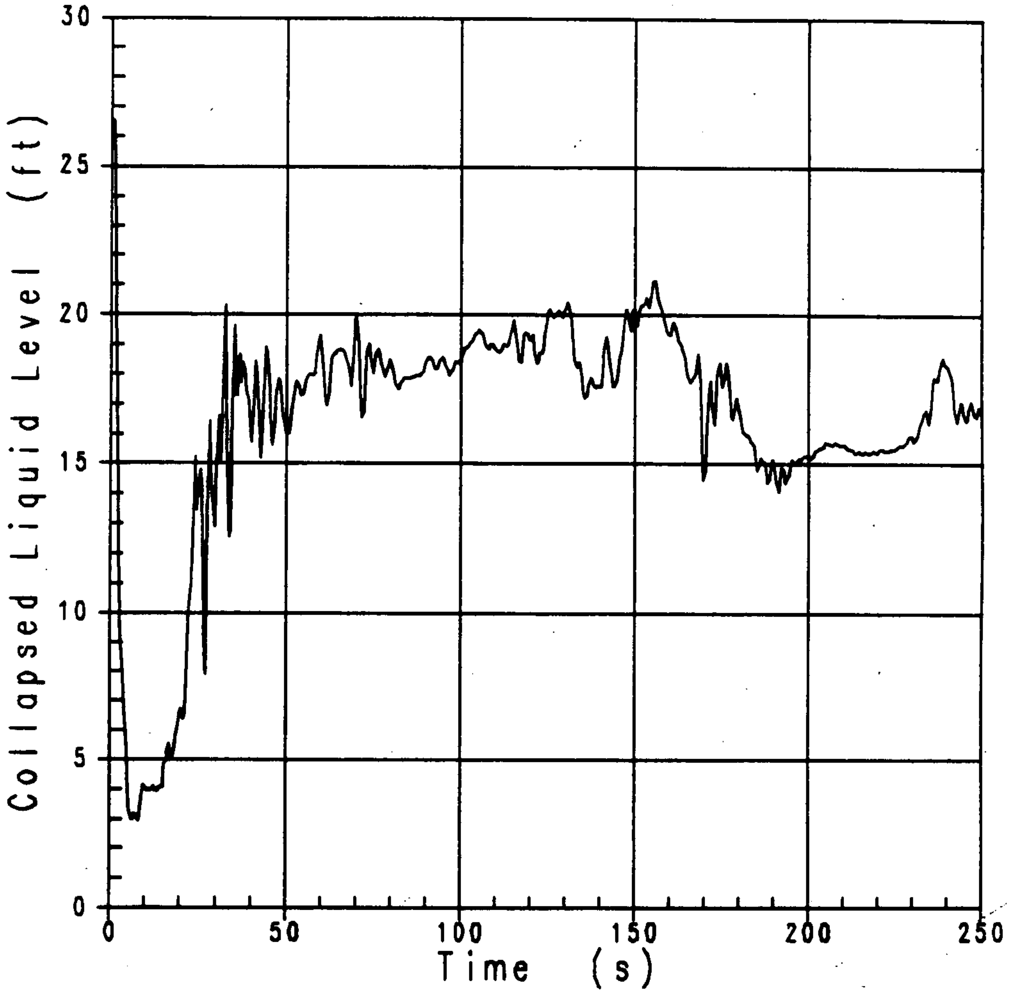

Figure 14.3.2-63

Nonrinal Calculation
Lower Plenum Liquid Level

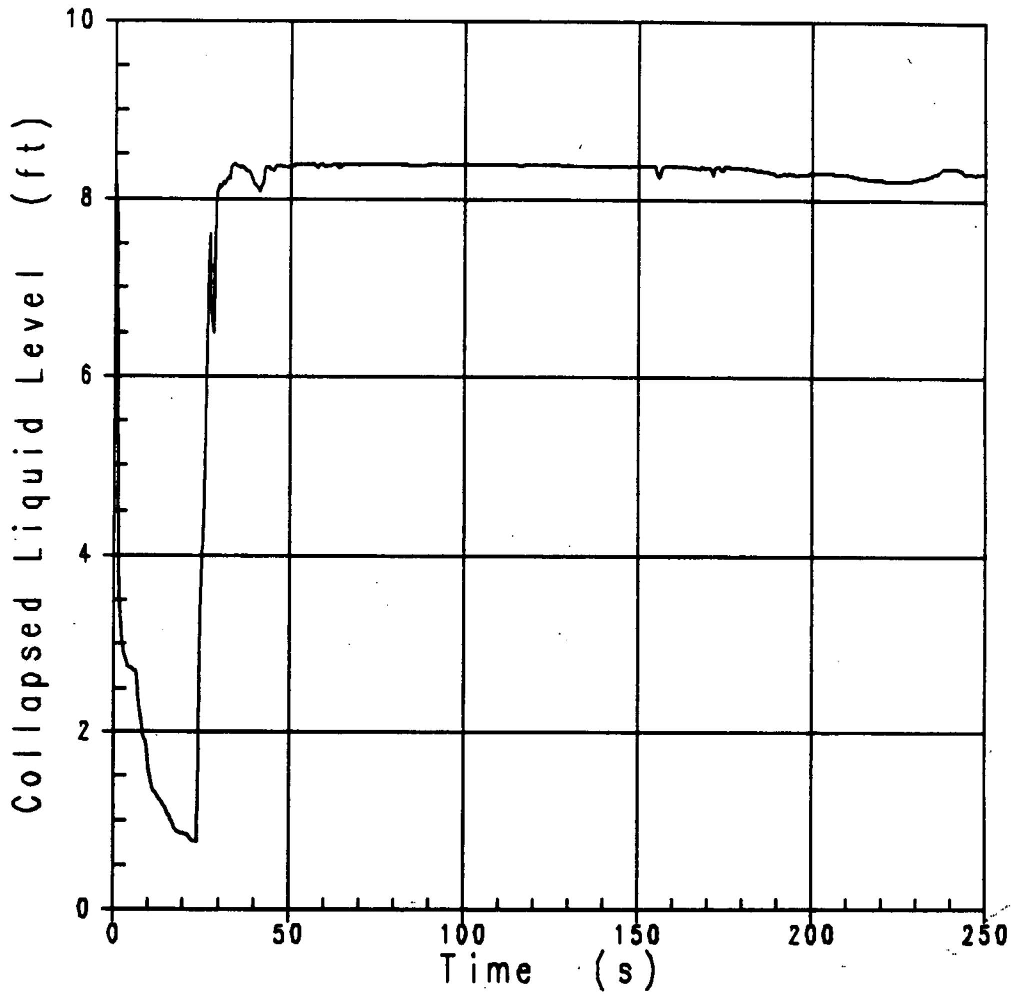

Figure 14.3.2-64

Nominal Calculation
Core Liquid Level

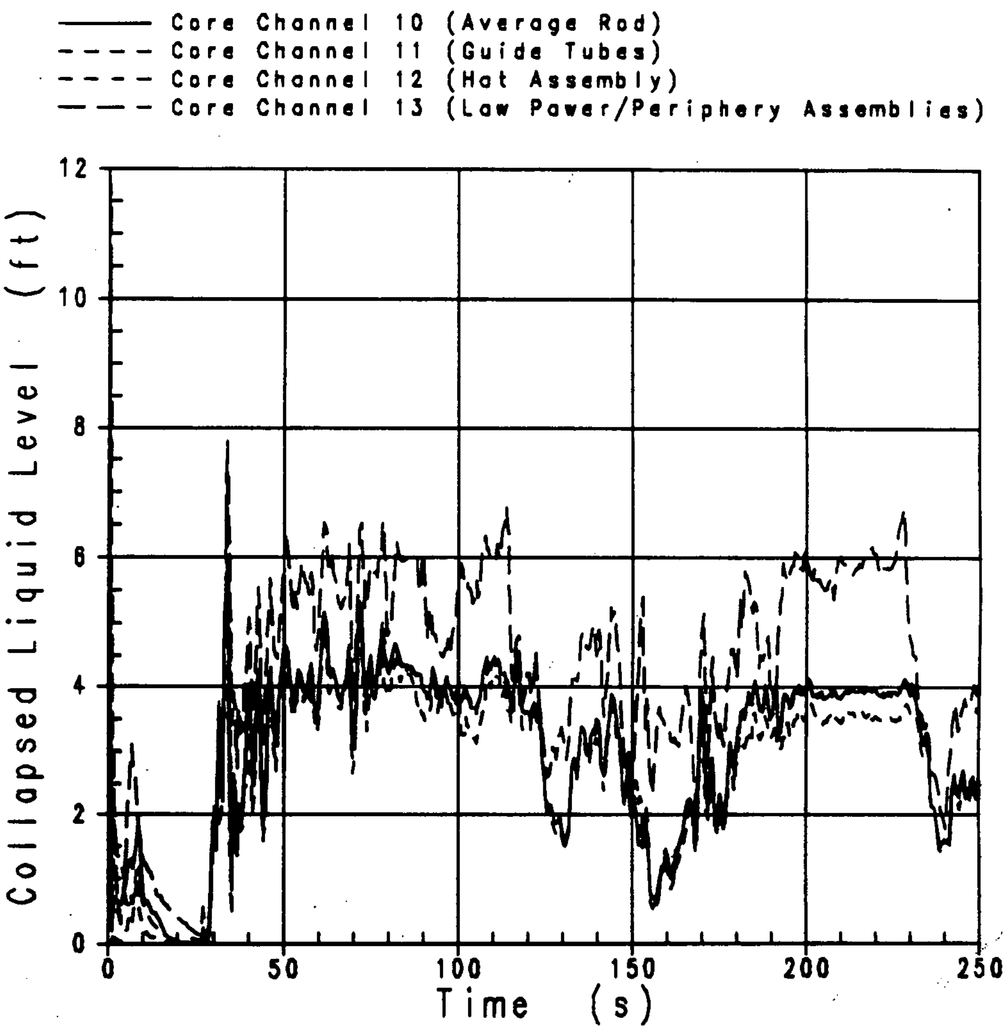

Figure 14.3.2-65

Nominal Calculation
Void Fraction in Lower Plenum

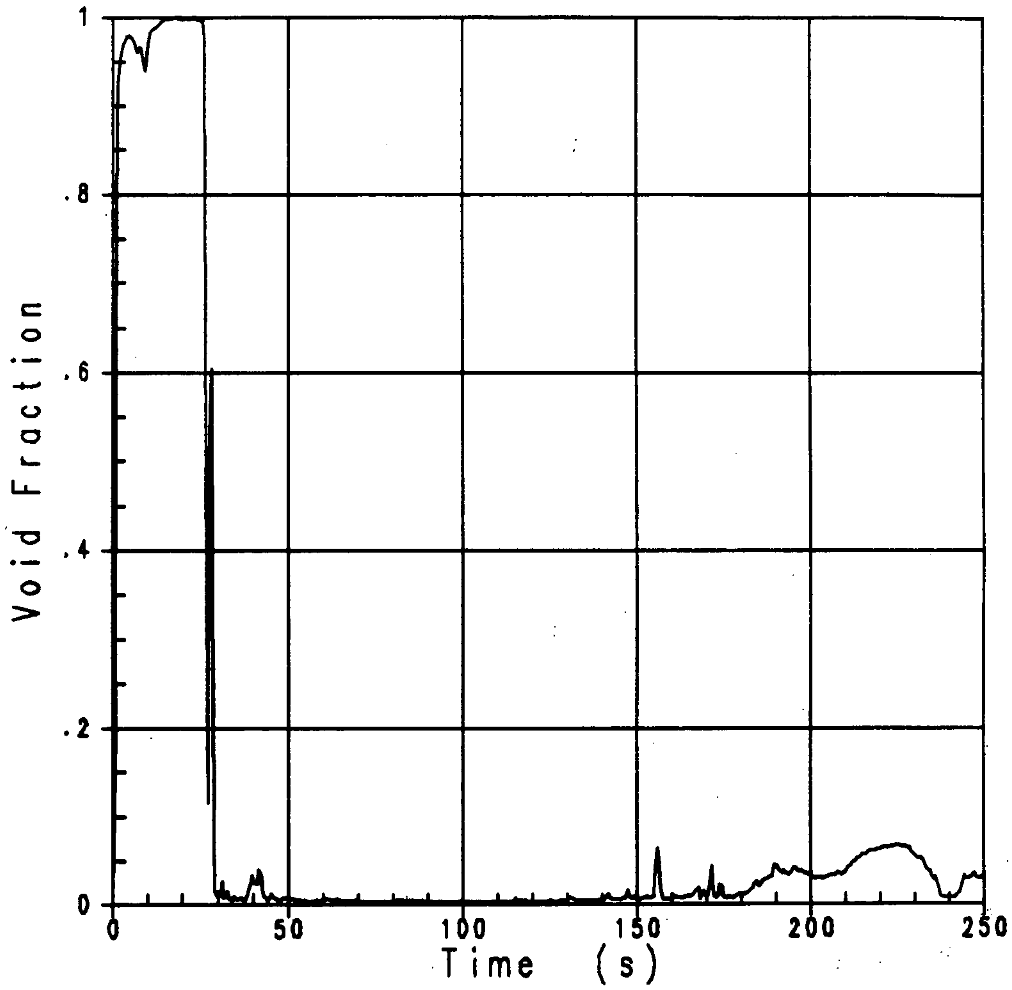

Figure 14.3.2-66

Nominal Calculation
Void Fraction at Bottom of Core