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Ref: #10CFR50.46
#10CFR50, Appendix K

CPSES-200500255
Log # TXX-05023

January 25, 2005

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
SUBMITTAL OF SUPPLEMENT TO THE CPSES LOSS OF COOLANT ACCIDENT (LOCA) ANALYSIS METHODOLOGIES – TOPICAL REPORT #ERX-04-004, REVISION 0

Gentlemen:

As an enclosure to this letter, TXU Generation Company LP (TXU Power) submits Revision 0 of the CPSES Topical Report ERX-04-004; "Replacement Steam Generator Supplement To TXU Power's Large and Small Break Loss Of Coolant Accident Analysis Methodologies." This supplement to the NRC approved methodologies already in place contains the changes to those methodologies that will be necessary to support future licensee submittals related to the replacement of the CPSES Unit 1 Steam Generators.

This supplement is not intended to replace the methodologies used for small or large break LOCA analyses on CPSES Unit 2, but is provided to reflect the physical differences that will exist between the steam generators in each unit.

This topical report supplement contains no proprietary information.

A001

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2. Should you have any questions, please contact Bob Kidwell at (254) 897-5310.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Regulatory Affairs

RJK
Enclosure

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Enclosure to TXX-05023

CPSES TOPICAL REPORT
ERX-04-004, Revision 0

REPLACEMENT STEAM GENERATOR SUPPLEMENT TO TXU POWER'S LARGE AND SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGIES

Dated January, 2005

ERX-04-004

TXU

REPLACEMENT STEAM GENERATOR SUPPLEMENT TO TXU POWER'S LARGE AND SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGIES

January, 2005

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REACTOR ENGINEERING

TXU POWER
COMANCHE PEAK STEAM ELECTRIC STATION

REPLACEMENT STEAM GENERATOR SUPPLEMENT TO TXU POWER'S LARGE AND SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGIES

ERX-04-004, Rev. 0

January, 2005

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ABSTRACT

This report is presented to demonstrate the implementation of replacement steam generator (RSG) models into the current, TXU Power, NRC-approved, Large Break and Small Break, Loss-of-Coolant Accident (LOCA) Emergency Core Cooling System (ECCS) Evaluation Models (EM).

Comanche Peak Steam Electric Station Unit 1 (CPSES-1) will undergo steam generator replacement while Unit 2 (CPSES-2) will retain the existing steam generators, which are modeled in the current methodologies. Therefore, the material presented in this topical report is intended to supplement rather than to replace the methodologies already in place. Thus, LOCA analyses for CPSES-2 will continue to be performed with the already approved LOCA methodologies which include the D-4 and D-5steam generator models.

The methodologies and this supplement are used to perform large and small break LOCA-ECCS licensing analyses that comply with NRC regulations contained in 10 CFR 50.46 and 10 CFR 50, Appendix K. The small break methodology and its supplement also satisfies the requirements of NUREG-0737, TMI Action Item II.K.3.30.

Because this report is a supplement, the methodology description sections present only the differences between the current models, which are applicable to the D-4 and D-5 steam generators and the proposed models, which are applicable to the Δ-76 RSGs. This keeps the report shorter by avoiding repetition of materials already examined by the NRC and circumvents the need to have proprietary and non-proprietary report versions. This is because proprietary information is already included in the proprietary versions of previously submitted reports. The information in this report is non-proprietary.

In order to demonstrate proper implementation of the methodologies for the replacement steam generators, a spectrum of large and small breaks, were examined in each respective section.

Two additional types of sensitivity studies were performed for the large break LOCA. The first was a single failure study to confirm that the most limiting single failure is used. The second was a convergence criterion study, demonstrating that the value used for this parameter is adequate to produce converged results.

Similarly for the small break model, two additional types of sensitivity studies were performed. The first was a time step study demonstrating that all break spectrum results are converged. The second was a cross-flow parameter study required by the methodology.

These demonstration analyses are of the same type as those submitted with the original methodologies.

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CHAPTER 1

INTRODUCTION

TXU Power currently performs its own large and small break Loss-of-Coolant Accident (LOCA) Emergency Core Cooling Systems (ECCS) licensing analyses to support the operation of Comanche Peak Steam Electric Station Unit 1 and Unit 2 (CPSES-1 and -2).

TXU Power's ECCS Evaluation Models (EM) of References 1 and 5 are based on Framatome ANP, Inc.'s (Framatome, formerly Siemens Power Corporation) methodologies (References 2 and 6). The methodologies have been approved by the NRC to perform the large and small break LOCA ECCS licensing analyses in compliance with NRC regulations contained in 10 CFR 50.46 and 10 CFR 50 Appendix K. TXU Power's large and small break LOCA methodologies are both supplemented by Reference 3.

At the end of Cycle 12, CPSES-1 will undergo steam generator replacement from the D-4 model to the Δ-76 model, while CPSES-2 will retain the existing D-5 model steam generators.

Features of the Δ-76 replacement steam generators (RSGs) need to be incorporated into TXU Power's large and small break LOCA Evaluation Models. Therefore, the objective of this report is to obtain NRC approval for changes to TXU Power's already approved ECCS Evaluation Models (References 1 and 5) so they may be used to analyze CPSES-1 with the Δ-76 RSGs.

CPSES-1 alone will undergo steam generator replacement. CPSES-2 will retain the existing D-5 steam generators, which are modeled in the current methodologies. (The D-4 and D-5 models are sufficiently similar that there are no differences between the Evaluation Models for CPSES-1 and CPSES-2, although separate analyses are performed for each unit.) Thus, the material presented in this topical report is intended to supplement rather than to replace the methodologies already in

place. Large and small LOCA analyses for CPSES-2 will continue to be performed with the already approved Evaluation Models (References 1 and 5, supplemented by Reference 3).

The methodologies and this supplement will be used to perform large and small break LOCA-ECCS licensing analyses that comply with NRC regulations contained in 10 CFR 50.46 and 10 CFR 50, Appendix K. The small break methodology and its supplement also satisfies the requirements of NUREG-0737, TMI Action Item II.K.3.30.

Chapter 2 of this report presents an overview of the Δ-76 RSGs primarily focusing on the differences pertinent to the LOCA methodologies with respect to the D-4 and D-5 steam generators. Chapters 3 and 4 deal with the small break LOCA. Chapter 3 discusses the model changes and Chapter 4 presents demonstration analyses, including a base case and the applicable sensitivities, as in Reference 5. Chapters 5 and 6 address the large break LOCA analysis similarly.

In order to comply with a 10 CFR 50, Appendix K requirement, a spectrum of large and small breaks, were examined in each respective section, Chapter 4 for the small break and Chapter 6 for the large break .

As in the original topical report (Reference 1), two additional types of sensitivity studies were performed for the large break LOCA. The first was a single failure study to confirm that the most limiting single failure is used. The second was a convergence criterion study, demonstrating that the value used for this parameter is adequate to produce converged results. These are presented in Chapter 6.

Similarly for the small break model, as in Reference 5 , two additional types of sensitivity studies were performed and are presented in Chapter 4. The first was a time step study demonstrating that

all break spectrum results are converged. The second was the cross-flow study required by the methodology (Reference 6).

This supplement to the TXU Power LOCA methodologies presented herein — including all results, input decks, inferences and conclusions presented within this report — will be incorporated into TXU Power's LOCA methodologies used to perform large and small break LOCA analyses for CPSES-1. The large and small break LOCA analyses for CPSES-2 will continue being performed with the existing NRC-approved methodologies (References 1, 5, supplemented by Reference 3).

CHAPTER 2

OVERVIEW OF THE STEAM GENERATOR DESIGNS

This section provides a brief summary of both the existing D-4 and of the replacement Δ-76 steam generators, focusing primarily (but not exclusively) on those features whose differences are significant to the LOCA progression.

Original Steam Generator Design Overview:

The original steam generator design used in CPSES-1 is a Westinghouse D-4 design (see Figure 2.1). This steam generator design includes an integral pre-heater, where approximately 90% of the total main feedwater flow is injected directly into the cold leg side of the tube bundle. This area is physically separated from the bulk of the recirulating fluid within the steam generator. Baffles direct the main feedwater across the tube bundle five times before it exits the pre-heater region and is allowed to mix with the recirculating fluid and continue to flow through the tube bundle. The remainder of the main feedwater flow is injected through the auxiliary feedwater nozzle where it mixes with the recirculating fluid and flows down to the tube bundle entrance. The use of the auxiliary feedwater nozzle for the main feedwater flow necessitates a connection between the Auxiliary Feedwater System and the Main Feedwater System. A significant portion of the auxiliary feedwater line is filled with relatively hot fluid from the Main Feedwater System that must be purged before the colder auxiliary feedwater fluid can enter the steam generator.

The D-4 steam generator design incorporates integral flow restrictors in the main feedwater nozzle and the steam nozzle. There are 4578 U-tubes, with a total heat transfer area of 48,300 ft^2. The tubes are fabricated from Alloy 600 Inconel. The outer diameter is 0.75" and they are arranged in a square lattice with a pitch of 1.0625". The volume of the shell side of the steam generator is approximately 5954 ft^3.

The steam generator water level instrumentation has a nominal span of 233". The lower tap is located in the annular downcomer region near the top of the U-tubes. The upper tap is located above the mid-deck plate (above the outlet of the primary separators). The nominal water level is 66.5% span.

Replacement Steam Generator Design Overview:

The replacement steam generator to be used in CPSES-1 is a Westinghouse Δ-76 design (see Figure 2.2). This steam generator design includes a feed ring through which 100% of the main feedwater is distributed into the recirculating fluid. Thirty-six spray nozzles, each comprised of 156 holes, one quarter of an inch in diameter, distribute the main feedwater into the upper downcomer region of the steam generator. The Auxiliary Feedwater System and the Main Feedwater System are completely separate. As a result, only relatively cold auxiliary feedwater is injected through the auxiliary feedwater nozzle, there is no purging of hot auxiliary feedwater, as described above for the D-4 CPSES-1 implementation.

The Δ-76 steam generator design incorporates integral flow restrictors in the steam nozzle only. There are 5532 U-tubes, with a total heat transfer area of 76,000 ft^2. The tubes are fabricated from Alloy 690 Inconel. The outer diameter of the U-tube is 0.75", and they are arranged in a triangular lattice with a pitch of 1.03". The volume of the shell side of the steam generator is approximately 5329 ft^3.

The steam generator water level instrumentation has a nominal span of 251". The lower tap is located above the annular downcomer region well below the top of the U-tubes. The upper tap is located above the mid-deck plate (above the outlet of the primary separators). The nominal water level is 67% span.

The primary and secondary steam separators, as well as the steam nozzles with their integral flow restrictors, are similar to the original steam generator design.

Comparison of Steam Generator Designs:

Comparisons of design and operating characteristics are presented in Table 2.1. The items of interest for the LOCA analyses, using the D-4 as the base case are:

(1) The larger tube-side volume results in an 11% increase in the overall Reactor Coolant System (RCS) volume. This affects the large break LOCA blowdown and post blowdown residual core mass. It also affects the loop seal clearing characteristics, which is relevant to the SBLOCA.

(2) The slightly larger shell-side mass, together with the 60% larger heat transfer area results in an overall higher heat transfer. This affects the loop seal clearing characteristics by condensing more water on the primary side of the tubes. This is relevant to the SBLOCA. The impact of the LBLOCA is negligible.

(3) The lack of a purge flow prior to cold auxiliary feed water reaching the steam generator downcomer may also affect the loop seal clearing characteristics by affect heat transfer. This is relevant to the SBLOCA.

TABLE 2.1

COMPARISON OF CURRENT AND REPLACEMENT STEAM GENERATORS

PARAMETER	Current Steam Generator (D-4)	Replacement Steam Generator (Δ-76)
Number of Tubes	4578	5532
Tube Outer Diameter, in	0.75	0.75
Tube Wall Thickness, in	0.043	0.043
Pitch, in	1.0625, square	1.03, triangular
Tube Material	Inconel 600	Alloy 690 Inconel
Secondary Side Heat Transfer Area, ft^2	48,300	76,000
Secondary Side Volume, ft3	5954	5329
Primary Side Volume, ft3	967	1303
Nominal Circulation Ratio at full power	2.44	4.10
Narrow Range Instrument Span, in	233	251
Nominal Water Level at power	66.5	67.0
Nominal Secondary Side Mass at power	105,000	112,000

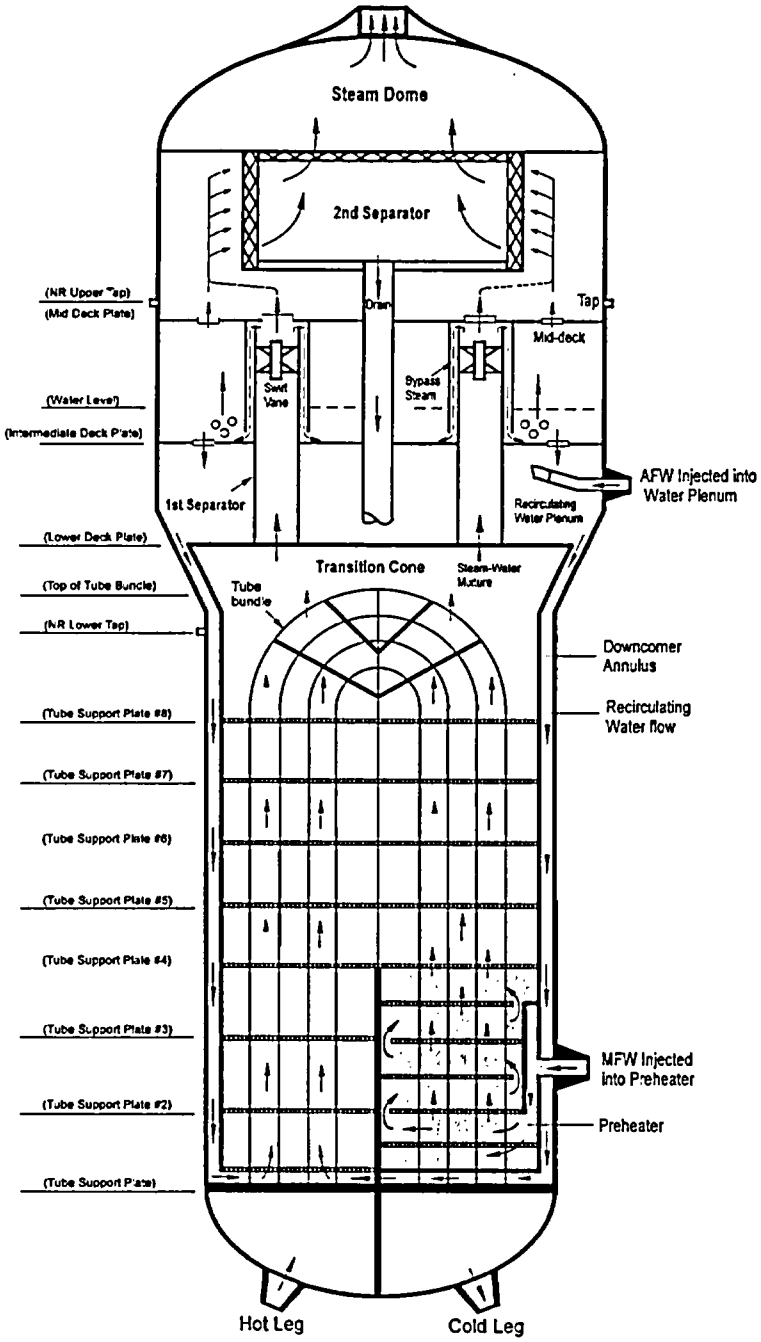

Figure 2.1 - D-4 Steam Generator Overview

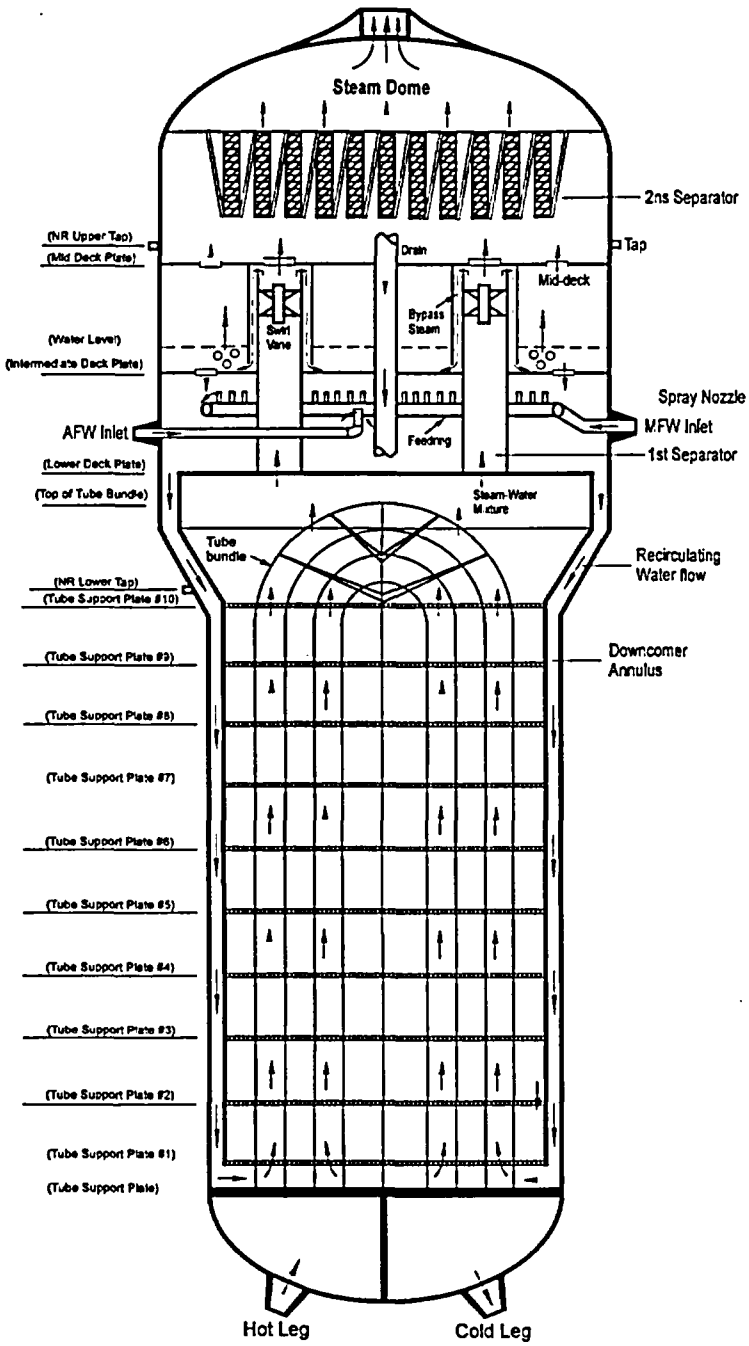

Figure 2.3 - Δ-76 Steam Generator Overview

CHAPTER 3

SMALL BREAK LOCA MODEL CHANGES

As discussed in Chapter 2 of Reference 5, TXU Power's Small Break LOCA (SBLOCA) methodology can be said, for presentation purposes, to embody three basic types of calculations: (1) Determination of Initial Fuel Conditions (RODEX2), (2) System Thermal-Hydraulic Response (ANF-RELAP), and (3) Hot-Rod Thermal Response and Cladding Heatup (TOODEE2). The only portion of the SBLOCA methodology affected by replacing the D-4 with the Δ-76 steam generators is the System Thermal-Hydraulic Response, namely, item (2) above. Consequently, the ANF-RELAP input model was the only portion of the methodology requiring modification. All other codes and inputs remain as described in Reference 5 and as supplemented in Reference 3.

Only the differences between the current ANF-RELAP input model, which is applicable to the D-4 and D-5 steam generators and the proposed ANF-RELAP input model, which is applicable to the Δ-76 RSGs are presented in this chapter. This approach makes sense because this topical report is a supplement to the existing model of Reference 5, as supplemented by Reference 3, which remains active and applicable to CPSES-2. In addition, restricting the ANF-RELAP input model discussion to these differences shortens this report, the material to be reviewed, and avoids use of proprietary information that is already available in previous topical reports approved by the NRC.

Δ-76 ANF-RELAP Input Model:

As discussed in Chapter 2 of Reference 5, the system thermal-hydraulic response during the SBLOCA is analyzed using ANF-RELAP, a modified version of RELAP5/MOD2.

Both, the existing CPSES ANF-RELAP NSSS model described in Reference 5, supplemented in Reference 3 and the model changes described in this report reflect a considerable amount of engineering insight and experience and incorporate:

a. Information from the most recent plant drawings, design basis documents, vendor documents and Technical Specifications.

b. Careful consideration of the guidelines set forth by Framatome for the application of their methodology (Reference 7).

The ANF-RELAP input model with the Δ-76 steam generators is quite similar to the existing input model with the D-4 and D-5 steam generators. In addition to the main difference, i.e. the steam generator model, there are two other minor differences:

a. The upper downcomer is modeled as four nodes (volumes 100, 102, 104 and 106) in the D-4/D-5 model (Figure 2.3 of Reference 5). These are collapsed into two nodes (104+106 and 100+102) in the new model. This change makes the model more robust numerically for Δ-76 applications, but does not significantly improve the numerics of the Unit 2 model. Thus, the Unit 2 model remains unchanged from the existing approved model described in Reference 5.

b. The flow area between the upper downcomer and the upper head "spray holes" was updated to reflect more accurate, recently developed design information. This is the junction connecting the new volume (100+102) to volume 181 (Figure 2.3 of Reference 5). This change wasn't implemented into the Unit 2 model either because the current value is more conservative, and, as stated above, it was desired to leave the Unit 2 model unchanged from the existing approved model.

The proposed Δ-76 ANF-RELAP steam generator model has the same nodalization structure of the D-4 model, depicted in Figure 2.3 of Reference 5. The Δ-76 model is essentially the same as the D-4 model. Only differences in the generators themselves, which are summarized in Chapter 2, were used to change the steam generator model: nodalization philosophy was unchanged. The Δ-76 geometrical information is based on TXU Power's RETRAN model (Reference 8), which is shown in Figure 3.1.

The ANF-RELAP model for the Δ-76 steam generators is developed from the RETRAN model by making the following two changes:

a. RETRAN volume X76 is split into three volumes: one a SEPARATOR component, another becomes an element of a PIPE component and a third becomes a SINGLE VOLUME component. The SINGLE VOLUME represents the upper most portion of the steam dome and is approximated by the corresponding volume in the D-4 model (Figure 2.3 of Reference 5). The SEPARATOR volume is simply the volume of X-76 in the RETRAN model minus the top sliver shown in Figure 3.1 and minus the volume on the outside of the primary separators between the lower and main decks. This latter volume (V*) is added to RETRAN volume X77 to form the uppermost downcomer node, which is represented by a PIPE component.

b. RETRAN volume X78 is divided in three to match the ANF-RELAP nodalization of Figure 2.3 of Reference 5.

The remaining RETRAN volumes and junctions, essentially correspond one to one to ANF-RELAP volumes and junctions. Again, the ANF-RELAP nodalization diagram of the Δ-76 steam generator is shown in Figure 2.3 of Reference 5 and is identical to the nodalization diagram of the D-4.

The nomenclature of the ANF-RELAP nodalization diagram corresponds to that in the RETRAN diagram of Figure 3.1 as shown in Table 3.1 for one loop. The ANF-RELAP model numbering scheme for the corresponding components in other loops is presented in Table 3. 2.

A final difference to be noted relates to the delivery of auxiliary feedwater. In the Δ-76, the motor driven auxiliary feedwater flow is delivered to the steam generators 60 seconds after the "S" signal (which bounds all delays). This flow consists of cold (120^{0}F) water. In the D-4, at the corresponding time, the motor driven auxiliary feedwater flow consists of hot (440^{0}F) water because this flow is delivered through a portion of the main feed water lines, which must first be purged of residual main feedwater, which is at 440^{0}F. This does not occur in the Δ-76, because the piping for main and auxiliary feedwater are completely separate in its CPSES-1 installation. The duration of this purge flow is approximately 150 seconds. Thus, between 60 seconds and 210 seconds after the "S" signal, the D-4 auxiliary feed water is at 440^{0}F, dropping to 120^{0}F after that, whereas in the Δ-76 it comes in at 120^{0}F 60 seconds after the "S" signal.

TABLE 3.1

CORRESPONDENCE BETWEEN THE RETRAN NODE NUMBERS OF FIGURE 3.1 AND THE ANF-RELAP NODE NUMBERS OF FIGURE 2.3 OF REFERENCE 5.

DESCRIPTION	RETRAN Node (Volume) Number	ANF-RELAP Node (Volume) Number
Inlet Plenum	X 20	VOL 422
Tubes	X 21 – X 28	VOLS 424 -1 thru - 8
Upper Downcomer	X 77 + V* (see text)	VOL 510 - 1
Lower Downcomer	X 78	VOL 510 – 2 thru 4
Shell Side in the Tube Bundle Region	X 71 – X 74	VOLS 540 - 1 thru - 4
Top of Tuble Bundle and Separators	X 75	VOL 540 - 5
Most of Steam Dome and Dryers	X 76 – top of steam dome – V* (see text)	VOL 560
Top of Steam Dome	Included in X 76	VOL 570
Outlet Plenum	X 40	VOL 426

TABLE 3.2

CORRESPONDENCE BETWEEN ANF-RELAP NODE NUMBERS FOR THE 4 IDENTICAL LOOPS

ANF-RELAP Loop 1 (Volume) Number	ANF-RELAP Loop 2 (Volume) Number	ANF-RELAP Loop 3 (Volume) Number	ANF-RELAP Loop 4 (Volume) Number
VOL 422	VOL 429	VOL 434	VOL 446
VOLS 424 -1 thru - 8	VOLS 431 -1 thru - 8	VOLS 437 -1 thru - 8	VOLS 444 -1 thru - 8
VOL 510 - 1	VOL 511 - 1	VOL 512 - 1	VOL 513 - 1
VOL 510 – 2 thru 4	VOL 511 – 2 thru 4	VOL 512 – 2 thru 4	VOL 513 – 2 thru 4
VOLS 540 - 1 thru - 4	VOLS 541 - 1 thru - 4	VOLS 542 - 1 thru - 4	VOLS 543 - 1 thru - 4
VOL 540 - 5	VOL 541 - 5	VOL 542 - 5	VOL 543 - 5
VOL 560	VOL 561	VOL 562	VOL 563
VOL 570	VOL 571	VOL 572	VOL 573
VOL 426	VOL 433	VOL 439	VOL 442

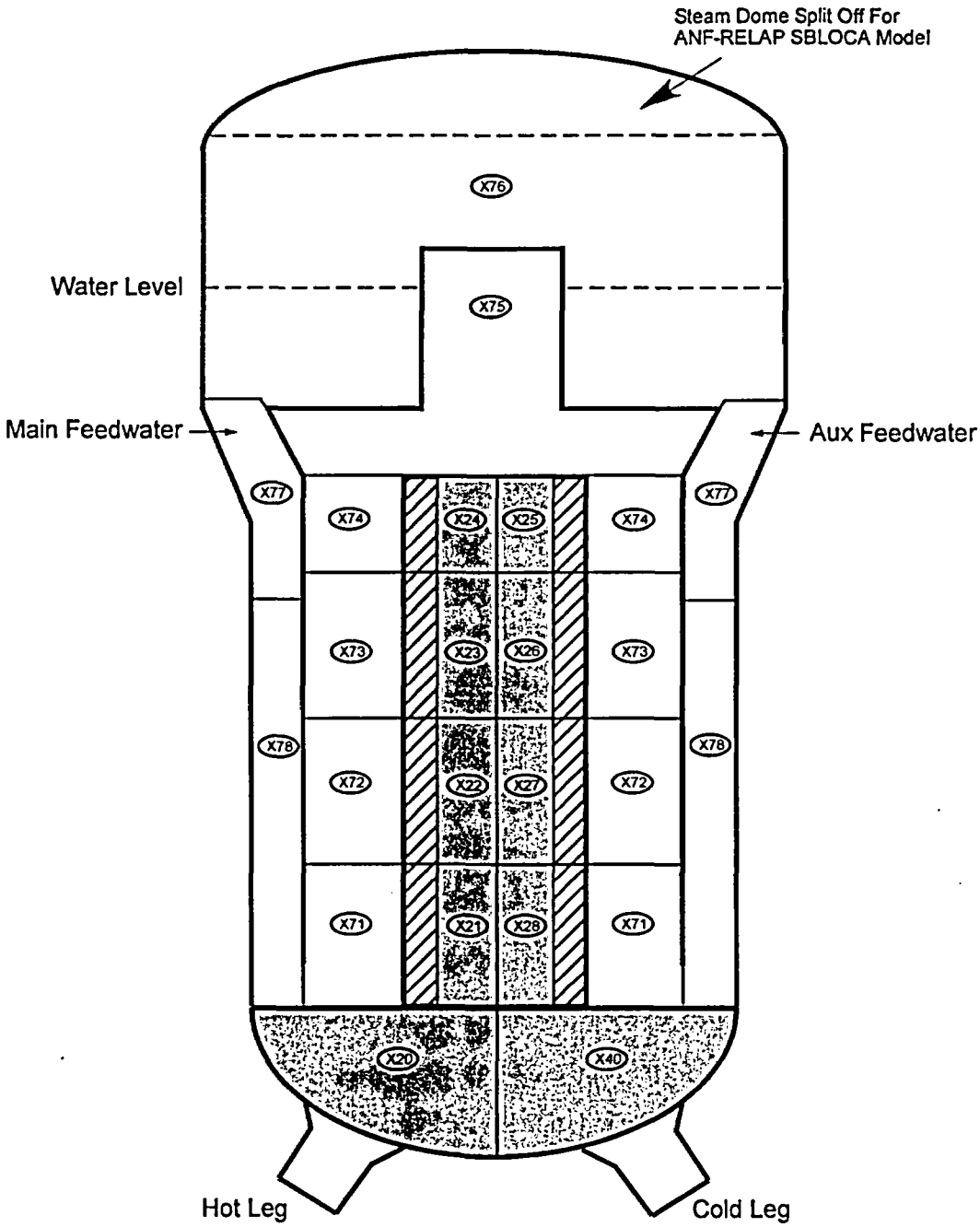

Figure 3.1 - RETRAN (Reference 8) RSG Nodalization Diagram

CHAPTER 4

SMALL BREAK LOCA DEMONSTRATION ANALYSES

As in Reference 5, method- and plant-specific issues were systematically considered in order to determine a base case and to thoroughly evaluate the impact of the ANF-RELAP model changes presented in Chapter 3.

Method-specific issues are suggested throughout 10 CFR 50.46, Appendix K thereto, and in NUREG-0737 II.K.3.30, and were addressed in Reference 6. The present work is simply a supplement to TXU Power's approved Evaluation Methodology (Reference 5), using method-specific parameters as prescribed by the method developers (Reference 7). Hence, the effect of variations in method-specific parameters within the bounds of methodology recommendations were already ascertained in Reference 6 and sensitivity studies for these variables need not be repeated here, with two notable exceptions: (1) a cross-flow sensitivity study and, (2) a time step study. The former was performed for the model incorporating the ANF-RELAP model changes, as mandated by Reference 9 for new implementations of that EM. The latter was conducted as well, even though the threshold for this requirement also per Reference 9, was not reached. Both these studies are presented in this chapter.

The plant-specific issues which warrant investigation are given in the following passages from 10 CFR 50.46, Appendix K thereto and NUREG-0611, along with the approach taken in addressing each one.

10 CFR 50.46 (a)(1)(i), requires that "a number of postulated loss-of-coolant accidents of different sizes, locations and other properties" be calculated in sufficient amount "to provide assurances that

the most severe postulated loss-of-coolant accidents are calculated." In compliance with this requirement, a break spectrum study was conducted and is also presented in this report.

10 CFR 50, Appendix K, Part I, A, (1) states: "A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime shall be studied and the one selected should be that which results in the most severe calculated consequences for the spectrum of postulated breaks and single failures analyzed."

The existing methodology is not being changed with respect to the approach to power shape selection. In any case, the methodology development itself does not require power shape sensitivity studies, although it requires that possible power shapes be considered in each cycle-specific analysis, which they are and will continue to be, in the manner presented in Chapter 3 of Reference 5 and also in Reference 1.

The following discussion on the single failure selection was presented in Reference 5. It remains applicable to the model incorporating the ANF-RELAP model changes and is repeated here only for the completeness: 10 CFR 50, Appendix K, Part I, D, (1) states: "an analysis of possible failure modes of ECCS equipment and their effects on ECCS performance must be made. In carrying out the accident evaluation, the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place." The limiting single failure for the small break loss-of-coolant accident analyses in the CPSES-1 and CPSES-2 FSAR is the loss of one ECCS injection train. Unless a common cause is established, the loss of one ECCS injection train involves multiple failures of ECCS equipment and therefore is not a single failure. The required common cause is the loss of power to the train. In order to arrive at this condition consistently, it must also be assumed that both the preferred 345 KV and the alternate 138 KV offsite power sources are lost and that one emergency diesel generator fails to start. Hence, the most damaging single failure of ECCS equipment postulated for the present study is the failure of an

emergency diesel generator to start. Offsite power (which is not ECCS equipment) unavailability is postulated in order to make the single failure meaningful, i.e. the diesel generator is not needed if either the preferred 345 KV or the alternate 148 KV offsite power sources are available. Thus, one motor driven and one turbine auxiliary feedwater pump, one high head centrifugal charging pump, one intermediate head safety injection pump and one low head residual heat removal (RHR) pump (which is not challenged in these analyses) as well as all four accumulators are available to mitigate the accident and are credited in all the calculations.

Finally, the model incorporating the ANF-RELAP changes for the Δ-76 steam generators assumes that ten percent of the steam generator tubes are plugged. This assumption is made to support the potential need for operation under such circumstances and is a conservative assumption when fewer tubes are actually obstructed. This same assumption is made in the existing methodology (Reference 5, supplemented by Reference 3).

4.1 BASE CASE ANALYSIS

This section presents licensing analysis results for a 4.0 inch diameter break in the discharge line of the Reactor Coolant Pump. The axial power shape, the fuel rod exposure and the remaining fuel parameters used in this base case were taken from the reload analysis for CPSES-1 cycle 11.

The accident assumptions are summarized in Table 4.1 and the initial conditions are summarized in Table 4.2. Table 4.3 summarizes the timing of significant events for this base case. The variable names (legends) in the figures follow traditional RELAP5 nomenclature. The locations of the variables correspond to the node numbers of Figures 2.2 and 2.3 of Reference 5.

Figure 4.1 shows the primary and the secondary pressures and is used as a road map in the following discussion of system performance during this accident. The four accident periods (marked I through IV) in this figure have the following characteristics:

Period I - Depressurization:

The accident period marked I in Figure 4.1 corresponds to the early rapid depressurization which follows break opening. From the secondary side standpoint, period I includes: (1) the early pressure rise due to steam production in the steam generators while the main steam lines are isolated and the steam dump and bypass system is assumed to be inoperable and (2) part of the period where the steam generators are discharging through the safety valves.

Period II - Voiding:

Period I ends and period II begins when a substantial production of steam begins in the core and slows down the depressurization rate. This substantial steam production begins when the bottom of the core starts to boil. This indicates that the whole core is boiling. Thus, the onset of period II occurs when the lowest core nodes begin to develop a significant void fraction. This occurs at the same time, around 150 seconds, for all three core "channels": the hot assembly (Figure 4.2), the central (Figure 4.3) and the average core (Figure 4.4) regions. At this time then, the entire core is boiling, resulting in a large production of steam. The effect of this steam production is to reduce the net depressurization rate of the primary system. That in turn leads to the nearly flat primary system pressure trace, which characterizes the first half of period II, as seen in Figure 4.1. During the first half of period II, water is held up in the upper plenum (Figure 4.5) by the steam generated in the core. This is due to the counter current flow limitation (CCFL) condition which occurs in the vicinity of the upper tie plate. At the end of the first half of period II, as steam production decreases, because less water is available, due to liquid boil off in the core, all but the broken loop seal clear (Figure 4.8). This allows the depressurization rate to increase for the second half of period II, by clearing vent paths from the upper plenum to the break. The loop seal clearing also temporarily allows some liquid to flow back into the core as evidenced by the increase in the colapsed level in the core (Figure 4.6) and the drop in the top core node void fractions (Figures 4.2, 4.3, 4.4). Eventually the upper plenum and then the upper core begin to dry-out, just prior to the onset of period III (Figure 4.5). There is also in period II an intermediate heat up caused by the colapsed

level dropping below the mid point of the hot assembly (6 ft). This is followed by an intermediate quenching of the clad, driven by the previously mentioned re-entry of fluid in the core from the loops, induced by the loop seal clearing (Figure 4.7).

Period II from the secondary side point of view has all secondary pressures remaining stable near the safety valves' set points. This is because in period I the steam generators' safety valves have opened due to the steam dump and bypass system unavailability, in order to discharge the steam produced. The atmospheric relief valves (ARVs) are not credited. Period II continues this behavior.

Period III - Heatup:

The end of period II and beginning of period III starts with the end of significant steam production in the core caused by shortage of liquid, i.e. the onset of dryout. The end of period II and beginning of period III can be determined from the time at which the core collapsed level reaches the mid core height of 6 ft, indicating the top part of the core has dried out. This can be seen in Figure 4.6. Another indicator is when the top of the core void fractions jumps to 1.0, also indicating dry out conditions there, as shown in Figures 4.2, 4.3 and 4.4. Thus, period III or the heat up period begins just before the hot rods enter into critical heat flux (CHF, Figure 4.7). The dropping of the collapsed core level to mid height (6 ft, Figure 4.6) and the rate at which it is dropping are indications that the core is drying out quickly and that steam production has become very low.

Period III is characterized by a continuation of the increased depressurization rate of the second half of period II. This comes from the compounded effects of: (1) the previously cleared loop seals (Figure 4.8) and, (2) the reduction in steam generation which had been compensating for the energy discharge through the break and keeping the pressure fairly constant in the early part of period II.

From the secondary side point of view, period III is characterized by a constant pressure, for the base case 4 inch break. In the D-4 analyses of Reference 5, loops 2 and 3 depressurized because

they received motor driven auxiliary feedwater, whereas loops 1 and 4 didn't for the same reason. In this Δ-76 analysis, broken loop 1, the only loop seal blocked for the duration of the transient, and loop 4 with the pressurizer received motor driven auxiliary feedwater 60 seconds after the "S" signal. Loops 2 and 3 did not receive motor driven auxiliary feedwater. This follows from the single failure described at the beginning of the chapter and is reversed from the original D-4 analysis of Reference 5 after sensitivities studies discovered this to be the more conservative assumption. In addition, the turbine driven auxiliary feedwater begins flowing to all four steam generators 10 minutes after the "S" signal or approximately at 600 seconds in the Δ-76 analysis. Eventually, loops 1 and 4 will begin to depressurize first as secondary inventory builds up from this feed flow, but this does not happen prior to the end of the transient in the 4-inch break Δ-76 analysis and therefore is not shown in Figure 4.1

It is during period III that the fuel experiences its main temperature excursion as shown in Figure 4.7. The clad temperatures of Figure 4.7 start to rise right at the beginning of period III, because that is by definition when these axial locations dry out.

Period IV - Recovery:

Period III ends when the system pressure reaches the accumulator injection pressure. At that time, shown in Figure 4.9, the injection of accumulator water marks the onset of period IV. Accumulator injection causes the core collapsed water level to rise (Figure 4.6) and clad temperatures to begin turning around. Figure 4.7 shows the clad temperature histories above, below and at the PCT node as calculated by ANF-RELAP. The rods are quenched from the bottom up with [node 11 quenching first, 12 next, followed by nodes 13 and 14]. Finally, Figures 4.9, 4.10 and 4.11 show the baseline break flow being overtaken by the combined pumped injection and accumulator flows, indicating the transient is over and stable recovery is underway.

4.2 SENSITIVITY STUDIES

4.2.1 BREAK SPECTRUM

The most limiting break location has been determined (Reference 10) to be in the cold leg at the reactor coolant pump discharge. Therefore, this cold leg break location remains most limiting for the present evaluation and a worst break location search need not be repeated. This most limiting break location is the one considered in all cases discussed throughout this work.

According to the TXU Power's approved Small Break LOCA methodology (Reference 5), the break size is the first sensitivity issue addressed. The rationale for addressing break size first is that system thermal-hydraulic behavior is largely affected by break size and less dependent on other issues. Consequently, the break size is a first order effect, while the others are second order.

The break spectrum study is conducted using the same power shape used for the base case.

Three break sizes were analyzed in detail, namely: 4 inch (base case), 3 inch, and 5 inch.

The accident assumptions for this and other studies are summarized in Table 4.1 and the initial conditions are summarized in Table 4.2. The sequence of events for the break spectrum study is summarized in Table 4.4.

The result of this study is that the most limiting break is the 4 inch break located in the reactor coolant pump discharge. The 3 inch and the 5 inch breaks result in lower peak clad temperatures than the base case. The other sensitivity studies use the limiting 4 inch break.

4 inch Break:

This is the base case calculation described in Section 4.1. The ANF-RELAP PCT is calculated to be 1829°F in node 12. The clad temperature history as calculated by the TOODEE2 code at the node where the PCT occurs is shown in Figure 4.12. The TOODEE2 PCT is 1830°F, 10.125 ft above the bottom of the core.

3 inch Break:

The calculated system behavior for this case is similar to the base case, although event durations are somewhat longer due to the smaller break size. The PCT is also lower. The ANF-RELAP PCT is calculated to be 1197°F also in node 12. The clad temperature history as calculated by the TOODEE2 code at the node where the PCT occurs is shown in Figure 4.24. The TOODEE2 PCT for the 3 inch case is 1226°F, also10.125 ft above the bottom of the core.

Figure 4.13 shows the primary and the secondary pressures. The same four accident periods (also marked I through IV in this figure) are used in the following discussion of the 3 inch break.

Period I - Depressurization:

As in the base case the accident period marked I in Figure 4.13 corresponds to the depressurization of the primary system due to the break while the secondary pressure rises to and remains at the safety valves' set point. There are no major distinctions between system behavior during this period between the 3 inch break and the 4 inch base case except that the depressurization rate is higher for the larger break.

Period II - Voiding:

As in the base case, period I ends and period II begins when a substantial production of steam starts in the core and slows down the depressurization rate. This substantial steam production also begins with the formation of void at the bottom core elevation. This indicates the entire core is boiling. It

occurs at the same time, around 300 seconds, for all three core "channels": the hot assembly (Figure 4.14), the central (Figure 4.15) and average (Figure 4.16) core regions. The 4 inch base case discussion for this period applies to the 3 inch break as well. In this case too three loop seals clear, the broken loop 1 remains plugged but loop 4 re-plugs again near the end of the period. Loop seal clearing, as in the 4 inch case, also leads to an increase in the primary system depressurization rate. The core uncovery in this period is deeper and longer lasting than in the 4 inch case as evidenced by the time the colapsed level spends below the mid-core elevation as seen in Figure 4.18. The high void fractions in the upper elevations of all core regions in this time frame (Figures 4.14, 4.15 and 4.16) also provide further evidence of dry out. As in the base case, the quenching of the 3 inch break intermediate heatup of the first half of period II is driven by redistribution of fluid from the loops to the core, induced by the loop seal clearing. Note the collapsed level increases due to this (Figure 4.18) as does the upper plenum liquid fraction (Figure 4.17). It is important to note that after the loop seals clear, the water level in the core drops more rapidly in the 4 inch break case than in the 3 inch break case. The main reason for this is that in the 4 inch case the vapor velocity seems to have prevented liquid in the hot legs and in the steam generator inlet plenum from flowing back into the core, so the water level reached the core mid point about 200 seconds after the loop seal cleared (Figure 4.6). In the 3 inch case vapor velocities were low enough that this water was able to find its way back into the core and keep the level above the mid point for 600 seconds (Figure 4.18). This additional water that flows back into the core is the main reason why the 4 inch break is more limiting than the 3 inch break. Secondary side behavior is similar to the 4 inch case, except that since the duration of the transient is longer there is time for the pressure to come down in steam generators 1 and 4 which received both motor-driven and turbine-driven auxiliary feed water, while staying at the safety valve set point in steam generators 2 and 3, which receive only the late-starting turbine driven auxiliary feedwater.

Period III - Heatup:

As in the 4 inch discussion, the end of period II and beginning of period III occurs when the core collapsed level drops below the mid core elevation of 6 ft (Figure 4.18) for the second time. (Recall that the first time is a temporary uncovery recovered by water from the loops finding its way back into the core after loop seal clearing.) The dropping of this level to about 6 ft means the top half of the core is dry, and steam production has been substantially reduced. The jump in top core elevations' void fractions to 1.0 also signals the onset of dryout in this case. The primary system pressure continues to drop significantly as two loop seals remain clear and steam production is low. It is also in period III that the fuel experiences its main temperature excursion as shown in Figure 4.19. For the 3 inch break the loop seals 2, 3 and 4 are also clear before the beginning of period III, however in this case loop 4 plugs up again near the end of period II, reducing the depressurization rate, as shown in Figure 4.13. Secondary side pressure follows primary side pressure in loops 1 and 4, which receive more auxiliary feedwater and that is expected as both systems are saturated and at similar temperature, due to heat transfer in those steam generators.

Period IV - Recovery:

As in the base case, period III ends when the system pressure reaches the accumulator injection pressure. At that time, shown in Figure 4.21, the injection of accumulator water marks the onset of period IV. Accumulator injection causes the core collapsed water level to rise (Figure 4.18) and clad temperatures to begin to turn around. Figure 4.19 shows the clad temperature histories above, below and at the PCT (node 12) location as calculated by ANF-RELAP. The rods are quenched from the bottom up with node 11 quenching first, 12 next, then 13 and node 14 last. Finally, Figures 4.22 and 4.23 show break flow and pumped injection flow, respectively. Pumped injection flow together with accumulator flow (Figure 4.21) is well on its way to overcome break flow also in the middle of period IV, indicating stable recovery is underway.

The same conclusion drawn for the base case applies to the 3 inch calculation. The pumped injection flows (Figure 4.23) cannot keep up with the break flow (Figure 4.22) during periods I, II and III. Still, the accumulator injection pressure is reached well before the clad temperatures are too high and the temperatures are effectively turned around. Although the 3 and 4 inch breaks show the same phenomena on slightly different time scale, the 4 inch break is more limiting because higher vapor velocities after the loop seals clear prevent water from the steam generator plenum and the hot legs from draining into the core, while this does not occur in the 3 inch break.

5 inch Break:

Again, the calculated system behavior for this case is similar to the base case, although for the 5 inch break event durations are somewhat shorter due to the larger break size. Still, as with 3 inch break, the PCT is also lower than the base 4 inch break case. The ANF-RELAP PCT is calculated to be 1203°F also in node 12. The clad temperature history as calculated by the TOODEE2 code at the node where the PCT occurs is shown in Figure 4.36. The TOODEE2 PCT for the 5 inch case is 1236°F, also 10.125 ft above the bottom of the core. The phenomenology of the 5 inch break is sufficiently similar to that of the 4 inch break (and of the 3 inch for that matter), as illustrated in Figures 4.25 through 4.35, that the detailed discussions for the various accident periods presented for those cases needn't be repeated for this size break.

D-4 versus Δ-76 SBLOCA Response for 3 Inch and 4 Inch Breaks:

Table 4.9 presents the SBLOCA 3 inch and 4 inch PCTs for CPSES-1 Cycle 11, as calculated with the current NRC-approved methodology of Reference 5 (as supplemented by Reference 3) for the D-4 steam generators. The table also includes, for comparison purposes, the 3 inch and 4 inch PCTs calculated as described in this report for the Δ-76 steam generators.

While comparing the results for the two steam generator types, it is appropriate to keep in mind that a key phenomenon driving differences between SBLOCA calculations is the loop seal clearing. The

time, the number and which loop seals clear, all significantly affect the analysis progression, because the depressurization rate is affected by these parameters. The depressurization rate in turn directly affects the PCT because the PCT occurs just after the RCS pressure reaches the accumulator set point. In addition, loop seal clearing drives liquid from the loops back into the core and/or allows liquid to drain back into the core from the hot side of the steam generators and the hot legs. This can also be a significant amount of liquid flow into the core and thereby also substantially affect the accident progression. The driving force for loop seal clearing is the pressure differential between the hot leg and the cold leg. The resisting force preventing the clearing is the amount of liquid in the loop seal, the water level, etc... This resistance to clearing is not the same in the four loop seals because: Loop 1 has the break, motor-driven auxiliary feedwater (MDAFW) and turbine-driven auxiliary feedwater (TDAFW); Loops 2 and 3 have TDAFW and Loop 4 has MDAFW, TDAFW and the pressurizer. Thus, different pressure differentials might be required to clear each of the loops, although Loops 2 and 3 should have a similar requirement. Given these considerations, it is clear why small variations in the driving pressure differential, which may be near pressure thresholds that will clear different loop seal configurations, can result it multiple loop seal clearing scenarios. Another factor that contributes to the number of loop seal clearing scenarios is the fact that once a loop seal is cleared the pressure differential for loop seal clearing drops significantly and may or may not build up to the level needed to clear other loop seals. It may even drop to a low enough level that partial or full plugging of previously cleared loop seals could occur. As a result of this threshold effect, apparently minor differences in initial and/or boundary conditions may result in different loop seal clearing histories and thereby have an impact on the PCT that appears disproportional to that scenario difference.

The most obvious difference is that the limiting D-4 PCT occurs for the 3 inch break, while the Δ-76 limiting PCT occurs for the 4 inch break. This shift is explained by two factors: (1) the (approximately 10%) larger primary side volume of the Δ-76 steam generators, which is due to the larger number of steam generator tubes and (2) the fact that the tube bundle is about 8ft taller in the Δ-76.

In order to explain how these factors result in PCT hierarchy reported above, the 4 inch break cases in the Δ-76 and D-4 are compared and then the 3 inch breaks in two generators are also compared.

Comparison of the D-4 and Δ-76 4 inch break cases shows that the RCS mass is always greater in the Δ-76 than in the D-4, which is consistent with the larger SG tube bundle volume. So why is the Δ-76 4 inch PCT higher? There are two reasons: One, the Δ-76 cleared only 3 loop seals whereas the D-4 cleared 4. This could be because the larger volume and the taller tube bundle make it comparatively more difficult (i.e. requiring a higher pressure differential) to clear the loops in the Δ-76. When more loop seals are cleared the depressurization rate is faster so that the accumulator injection occurs earlier. Since accumulator injection terminates the heat up, the higher inventory, if it is plugging up a loop seal and comparatively slowing down the depressurization rate, results in a higher rather than lower PCT. The second reason why the Δ-76 PCT is higher, in spite of having a higher RCS inventory, is due to the location of the inventory. Here too the number of loop seals cleared plays a role. Of the approximately 15,000 lbs more mass the in the Δ-76 case, approximately 10,000 lbs are held up in the loop seal that didn't clear. Obviously, this mass does not contribute to prevent core heat up. This still leaves the Δ-76 with about 5000 lbs more in the RCS than the D-4. Further examination of the runs indicates that the Δ-76 is storing water in excess of these 5000 lbs in the hot legs and steam generator inlet plena. Again this could be due to the larger tube bundle volume and higher tube bundle height in the Δ-76 making it harder to blow this mass around the tubes and back into the reactor vessel downcomer during the loop seal clearing. Also, after loop seal clearing, the vapor velocities in the Δ-76 (6 ft/s) seem to be high enough to prevent this liquid from draining back into the core. It is noted that these vapor velocities are higher than they are in the D-4 (4.5 ft/s). Thus, the 4 inch break is more limiting in the Δ-76 than in the D-4 because it clears one less loop seal and it stores more water in the hot leg and steam generator inlet plenum, away from the core, after loop seal clearing, ultimately because of the larger volume and taller tube bundle geometry.

Comparison of the Δ-76 and D-4 for the 3 inch break results shows that both cases cleared two loop seals. The Δ-76 case showed a strong level depression between 400 and 800 seconds whereas the D-4 case showed none. The cause of the level depression was traced to storage of fluid in the upside of the tubes of the Δ-76. The D-4 case showed no such storage. After the loop seals cleared this liquid stored in the upside of the tubes of the Δ -76 flowed back into the core. This resulted in the relatively lower heat up for the Δ -76, since the D-4 had no such liquid storage.

It should be noted that in the Δ-76, the motor driven auxiliary feedwater flow is delivered to the steam generators 60 seconds (which bounds all delays) after the "S" signal. This flow consists of cold (120^{0}F) water. In the D-4, at the corresponding time, the motor driven auxiliary feedwater flow consists of hot (440^{0}F) water because this flow is delivered through a portion of the main feed water lines which must be purged of the hotter main feedwater temperature. Because of separate piping for main and auxiliary feedwater, this does not occur in the Δ-76. The duration of this purge flow is approximately 150 seconds. Thus, between approximately 60 seconds and 210 seconds after the "S" signal, the D-4 auxiliary feedwater is at 440^{0}F, dropping to 120^{0}F after that, whereas in the Δ-76 it comes is at 120^{0}F 60 seconds after the "S" signal. This difference is notable and modeled. Nevertheless, since it occurs so early in the transient, its impact on the comparative results for the two steam generator designs cannot be singled out from the many other differences. Even so, it is consistent with more water in the vicinity of the tubes in the Δ-76, as discussed above, because colder AFW has the potential to condense more water on the primary side.

4.2.2 CROSS-FLOW SENSITIVITY STUDY

The cross-flow sensitivity study is required for new implementations of this small break LOCA methodology (Reference 9). This study is performed for the most limiting break determined in the break spectrum study (4 inch, Section 4.2.1). The study is implemented as discussed in Reference 5. Framatome considered some of the material in the description of this study to be proprietary. Thus, additional information is only available in Reference 5.

The sequence of events for the two sensitivity cases are summarized and compared to the nominal case in Table 4.5. Figure 4.37 overlays the calculated ANF-RELAP clad temperatures for all three cases.

The conclusion, as seen in Figure 4.37, is that there is little difference in clad temperature history associated with these parameters for the CPSES model. In any case, nominal values for these parameters used in the base case calculation are the most limiting, as indicated in the PCT summary of Table 4.7. The same conclusion was reached in the D-4 application of Reference 5.

4.2.3 TIME STEP SENSITIVITY STUDY

A time step sensitivity study is not required (Reference 9) unless the PCT exceeds 2050^{0}F, or the clad temperature is increasing more than 0.75^{0}F in the maximum time step of 0.05 seconds or, the calculated small break LOCA PCT is within 25^{0}F of the large break LOCA PCT. None of these conditions apply to the present small break LOCA analyses.

Nevertheless, this study was performed for all three breaks in the break spectrum study. The main convergence criterion was a visual inspection of the behavior throughout the transient, of the most sensitive variable: the clad temperature. In addition to this visual criterion, in order to be deemed "converged", a run must also exhibit the same sequence of events of a smaller time step run. For example, if accumulator injection precedes the PCT in the smaller time step, this must also be the case for a larger time step to be acceptable and the loop seal clearing sequence should be the same. Thus, similar clad temperature histories are considered necessary but not sufficient conditions. Finally, although not a requirement, a maximum time step that was consistent throughout the break spectrum study was felt to be desirable if reasonably achievable.

Figure 4.38 as well as Table 4.8 show three time step runs (0.005, 0.0025 and 0.00125 seconds) for the base case 4 inch break. All three show similar results (any given PCT within approximately

30^{0}F of the next time step's PCT) and identical event sequences. In addition the PCTs are decreasing with time step. Based on these results, it is concluded that a maximum time step of 0.005 seconds is adequate.

Figure 4.39 and Table 4.8 show three time steps (0.005, 0.0025 and 0.00125 seconds) for the 3 inch break. Here too, all three show similar results although not as close as the base 4 inch break case. (The 0.005 sec PCT was within approximately 35^{0}F of the 0.0025 sec PCT but the next time step's was further apart 65^{0}F to 85^{0}F depending on the code result, i.e. TOODEE2 versus ANF-RELAP). However, the event sequences were identical for all cases. In addition, here too the PCTs are decreasing with time step. It is felt the smaller time step variation is not important because this break size is not limiting, the difference is amplified by the slow progression of the transient and the optimum time step and the next smallest one are very close. Based on these results, it is concluded that a maximum time step of 0.005 seconds is adequate.

Figure 4.40 as well as Table 4.8 show three time step runs (0.005, 0.0025 and 0.00125 seconds) for the base case 5 inch break. All three show similar results (any given PCT within 10 - 20^{0}F of the next time step's PCT depending on the code result, i.e. TOODEE2 versus ANF-RELAP) and identical event sequences. Based on these results, it is concluded that a maximum time step of 0.005 seconds is adequate.

In summary, all cases were certainly converged at 0.005 seconds and that value was chosen as the optimum time step. That is the same optimum time step value approved for use in the D-4 and D-5 version of this methodology, documented in Reference 5 .

Although converged at 0.005 seconds, the actual numerical value of the PCT can vary somewhat as the time step is reduced further. This can be interpreted as a convergence band. If the band is to the right of (i.e. higher than) the PCT, there could be a concern that the "actual" PCT would be

larger. This issue was examined and it is evident that the PCT is decreasing with time step for the limiting break. For the limiting break, this variation is concluded to be around 30^0F. If the PCT approaches the SER 2050^0F limit or the large break LOCA PCT, a time step study is required by the methodology (Reference 9).

Time step studies will only be conducted in future applications if:

(1) the conditions under which the SER requires a time step study apply or,

(2) if time steps larger than 0.005 second are utilized.

TABLE 4.1

SUMMARY OF CPSES-1 SMALL BREAK LOCA ANALYSIS

ASSUMPTIONS FOR BASE CASE AND SENSITIVITY STUDIES

1. The initial power is 3479 MWt, which is 0.6% above the licensed power level of 3458 MWt, to account for calorimetric measurement uncertainty.

2. 10% of the steam generator tubes are plugged.

3. Break in reactor coolant pump discharge occurs at 0.0 s.

4. Reactor trips due to a Lo-Pressurizer pressure signal.

5. Loss of offsite power coincides with reactor trip.

6. The reactor coolant pumps (RCP) are tripped at reactor trip since RCP cannot operate without offsite power after a reactor trip.

7. Steam flow isolation is initiated at the time of reactor trip. The steam dump and bypass system is not credited.

8. Main feedwater isolation is initiated 7 seconds after "S" signal.

9. Failure of one diesel generator to start takes out one high head centrifugal charging pump, one intermediate head safety injection pump, one RHR pump and one motor-driven AFW pump. This is the single failure assumed for compliance with 10 CFR 50, Appendix K, Part D.

10. One high head centrifugal charging pump, one intermediate head safety injection pump inject on demand after the appropriate delays, at conservative flow rates.

11. One of the two motor-driven AFW pumps is credited, but injection is conservatively delayed in order to account for motor start time. The turbine-driven AFW pump is also credited after 10 minutes.

12. All accumulators inject on demand.

TABLE 4.2

SUMMARY OF INITIAL CONDITIONS FOR CPSES-1 SMALL BREAK LOCA BASE CASE AND SENSITIVITY STUDIES

DESCRIPTION	VALUE
o Core Power Analyzed	3479 MWt
o Power Shape Analyzed	CPSES-1 Cycle 11
o Accumulator Water Volume per Tank	6119 gals
o Accumulator Cover Gas Pressure	603 psia
o Accumulator Water Temperature	150 °F
o Refueling Water Storage Tank Temperature	120 °F
o Initial Loop Flow	9522 lbm/sec
o Vessel Inlet Temperature	558 °F
o Vessel Outlet Temperature	620 °F
o Reactor Coolant Pressure	2280 psia
o Steam Pressure	956 psia
o Motor Driven Auxiliary Feedwater Flow to each of SGs 2 & 3	0.00 lb/sec
o Motor Driven Auxiliary Feedwater Flow to each of SGs 1 & 4	27.5 lb/sec (60 sec after "S" signal)
o Turbine Driven Auxiliary Feedwater Flow to each of SGs1, 2, 3 & 4	27.5 lb/sec (10 min after "S" signal)
o Steam Generator Tube Plugging Level	10%
o Fuel Parameters	CPSES-1 Cycle 11

TABLE 4.3

SEQUENCE OF EVENTS FOR BASE CASE[1] SMALL BREAK LOCA

EVENT	TIME (SECONDS)
1. Break opens (period I begins)	0.0
2. Reactor Trip Signal	6.0
3. RCP tripped	8.0
4. MSIV closed	10.0
5. "S" Signal	14.4
6. MFW isolated	21.4
7. Centrifugal charging pumps inject	31.4
8. Safety injection pumps inject	36.4
9. Motor-Driven Auxiliary Feedwater reaches SGs 1 & 4	76.0
10. Entire core boils (period II begins)	~150
11. Loop seals clear	~470
12. Turbine-Driven Auxiliary Feedwater reaches all SGs	~600
13. Critical Heat Flux at PCT node (period III begins)	~650
14. Accumulator injection (period IV begins)	~1040
15. Peak clad temperature reached	~1050
16. Calculation ends	1200.0

1 4 inch break, nominal cross flow parameters and 0.005 second maximum time step.

TABLE 4.4

SEQUENCE OF EVENTS FOR BREAK SPECTRUM[2] STUDY

EVENT	TIME (SECONDS)		
	3 inch	4 inch	5 inch
1. Break opens (period I begins)	0.0	0.0	0.0
2. Reactor Trip Signal	10.7	6.0	4.0
3. RCP tripped	12.7	8.8	6.0
4. MSIV closed	14.0	10.0	8.0
5. "S" Signal	20.2	14.4	11.5
6. MFW isolated	27.2	21.4	18.5
7. Centrifugal charging pumps inject	37.2	31.4	28.5
8. Safety injection pumps inject	42.7	36.4	33.5
9. Motor-Driven Auxiliary Feedwater reaches SGs 1 & 4	82.0	76.0	72.0
10. Entire core boils (period II begins)	~ 300	~ 150	~ 100
11. Loop seals clear	~ 1470	~ 470	~ 300
12. Turbine-Driven Auxiliary Feedwater reaches all SGs	~ 600	~ 600	~ 600
13. Critical Heat Flux at PCT node (period III begins)	~ 1650	~ 650	~ 475
14. Accumulator injection (period IV begins)	1798	1044	626
15. Peak clad temperature reached	1860	1050	630
16. Calculation ends	2000.0	1200.0	650.0

[2] All cases: nominal cross-flow parameters and maximum ANF-RELAP time step of 0.005 seconds.

TABLE 4.5

SEQUENCE OF EVENTS FOR CROSS-FLOW STUDY[3]

EVENT	TIME (SECONDS)		
	[Nominal]	[Times 10]	[Times 0.1]
1. Break opens (period I begins)	0.0	0.0	0.0
2. Reactor Trip Signal	6.0	6.0	6.0
3. RCP tripped	8.0	8.0	8.0
4. MSIV closed	10.0	10.0	10.0
5. "S" Signal	14.4	14.4	14.4
6. MFW isolated	21.4	21.4	21.4
7. Centrifugal charging pumps inject	31.4	31.4	31.4
8. Safety injection pumps inject	36.4	36.4	36.4
9. Motor-Driven Auxiliary Feedwater reaches SGs 1 & 4	76.0	76.0	76.0
10. Entire core boils (period II begins)	~ 300	~ 300	~ 300
11. Loop seals clear	~ 466	~ 494	~ 444
12. Turbine-Driven Auxiliary Feedwater reaches all SGs	~ 600	~ 600	~ 600
13. Critical Heat Flux at PCT node (period III begins)	~ 650	~ 650	~ 650
14. Accumulator injection (period IV begins)	1044	1034	1032
15. Peak clad temperature reached	1050	1048	1042
16. Calculation ends	1200.0	1200.0	1200.0

[3] All cases: 4 inch break and maximum ANF-RELAP time step of 0.005 seconds. TOODEE2 runs were not needed.

TABLE 4.6

PCT SUMMARY FOR BREAK SPECTRUM STUDY[4]

BREAK SIZE (INCHES)	ANF-RELAP PCT (^{0}F)	TOODEE2 PCT (^{0}F)
3.0	1197	1226
4.0	1829	1830
5.0	1203	1236

TABLE 4.7

PCT SUMMARY FOR CROSS-FLOW STUDY[5]

CROSS-FLOW PARAMETER	ANF-RELAP PCT (^{0}F)	TOODEE2 PCT (^{0}F)
NOMINAL	1829	1830
10 TIMES NOMINAL	1768	1772
NOMINAL DIVIDED BY 10	1766	1766

---

4 All cases: nominal cross-flow parameters and maximum ANF-RELAP time step of 0.005 seconds.

5 All cases: 4 inch break and maximum ANF-RELAP time step of 0.005 seconds.

TABLE 4.8

PCT SUMMARY FOR TIME STEP STUDY[6]

4 INCH BREAK		
MAX ANF-RELAP Δt (Sec)	ANF-RELAP PCT (°F)	TOODEE2 PCT (°F)
0.005	1829	1830
0.0025	1800	1804
0.00125	1762	1776

3 INCH BREAK		
MAX ANF-RELAP Δt (Sec)	ANF-RELAP PCT (°F)	TOODEE2 PCT (°F)
0.0050	1197	1226
0.0025	1163	1193
0.00125	1081	1128

5 INCH BREAK		
MAX ANF-RELAP Δt (Sec)	ANF-RELAP PCT (°F)	TOODEE2 PCT (°F)
0.0050	1203	1236
0.0025	1205	1245
0.00125	1197	1228

6 All cases: nominal cross-flow parameters.

TABLE 4.9

PCT COMPARISON: D-4 (CPSES-1 CYCLE 11) VERSUS Δ-76 (THIS STUDY)

BREAK SIZE (INCHES)	ANF-RELAP PCT (^{0}F)		TOODEE2 PCT (^{0}F)	
	D-4	Δ-76	D-4	Δ-76
3.0	1828	1197	1843	1226
4.0	1680[7]	1829	1687[7]	1830

[7] Calculated with CPSES-1 Cycle 11 model with the D-4 SG.

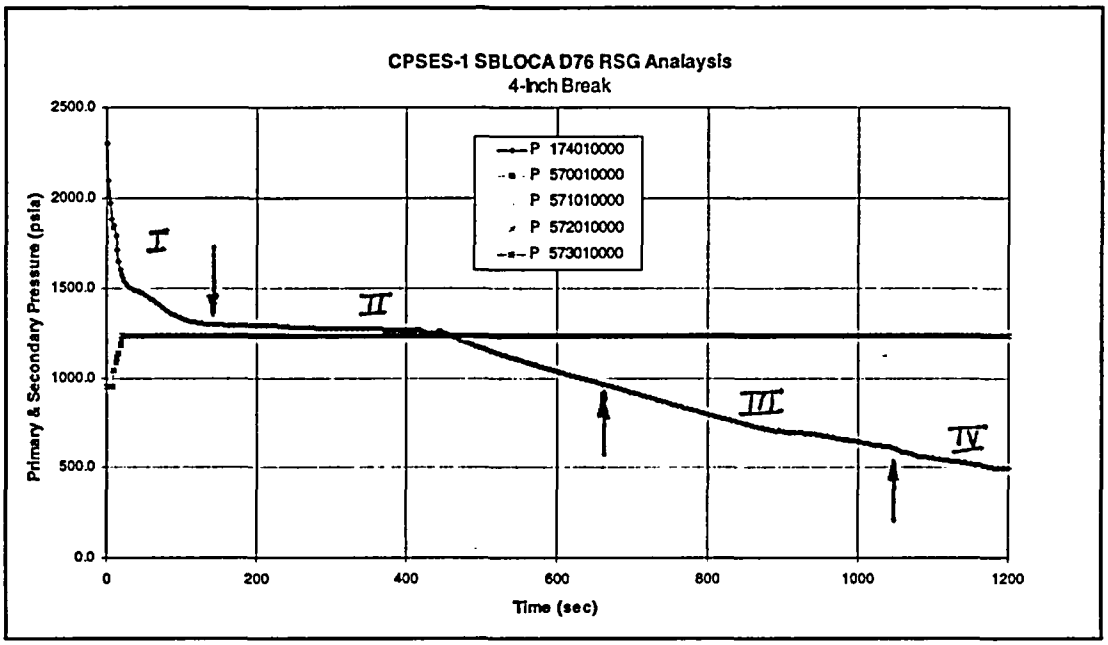

Figure 4.1 - Primary and Secondary Pressures

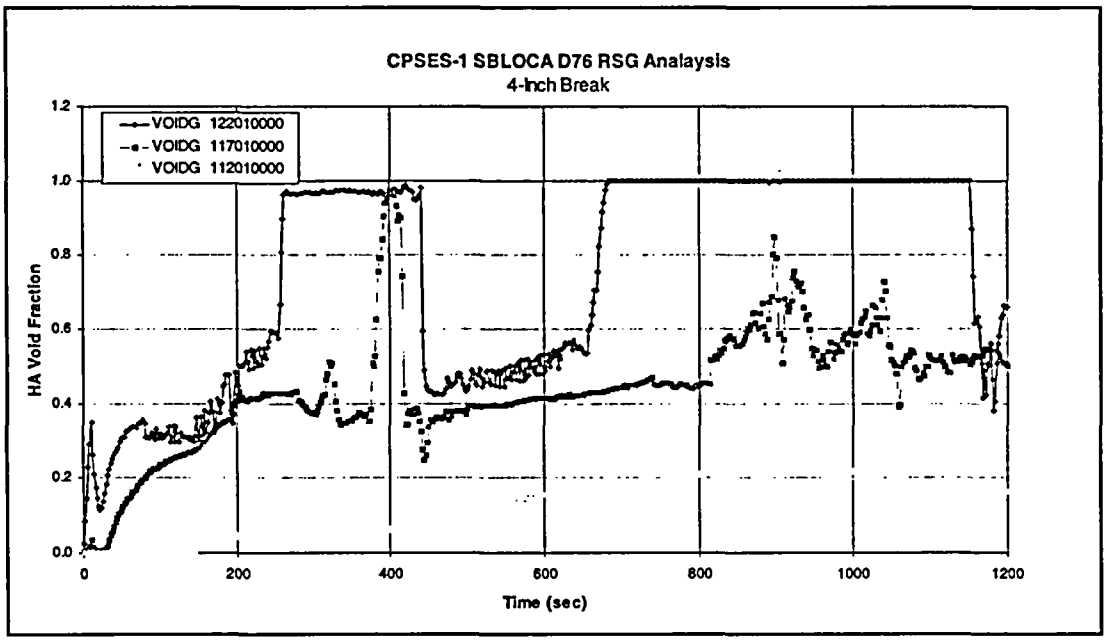

Figure 4.2- Hot Assembly Void Fractions

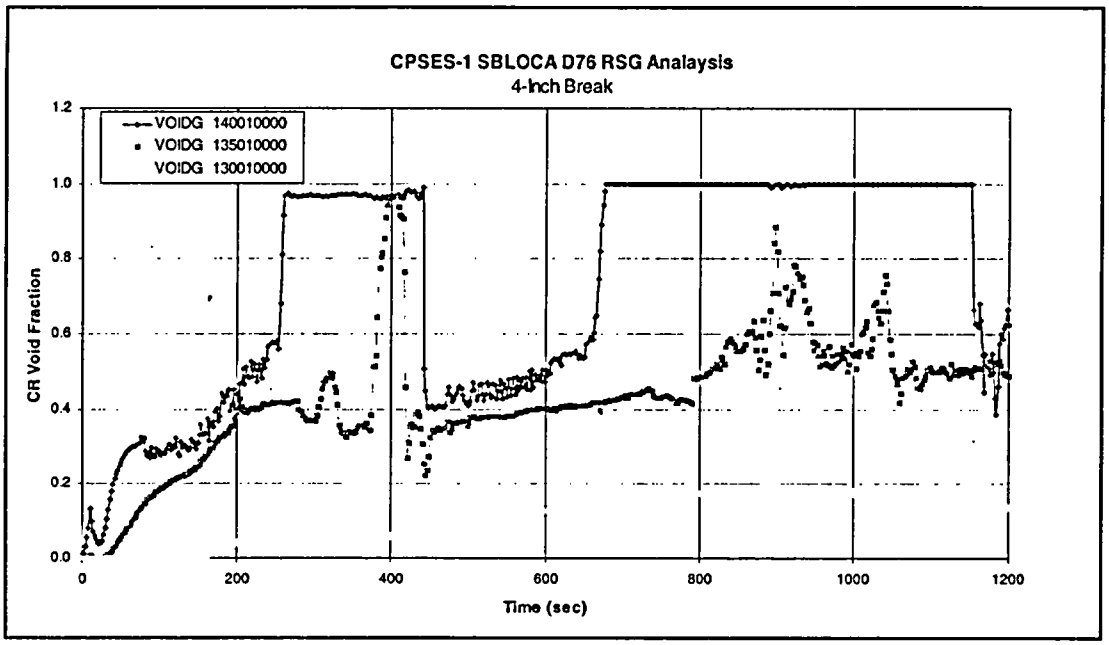

Figure 4.3 - Core Central Region Void Fractions

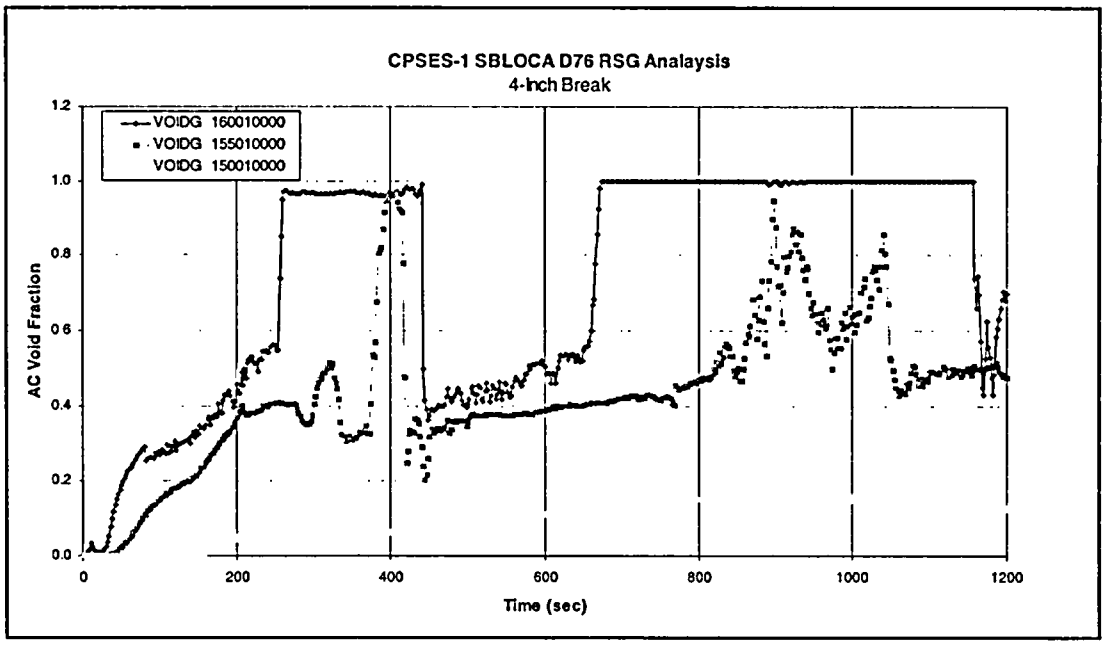

Figure 4.4 - Average Core Void Fractions

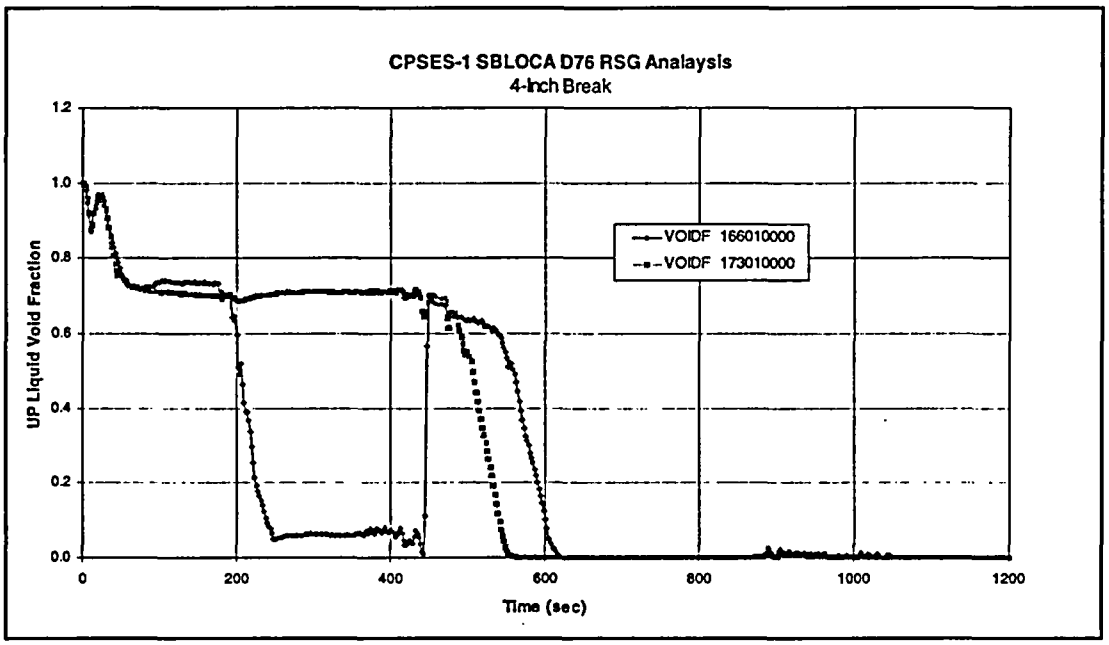

Figure 4.5 - Upper Plenum Liquid Fractions

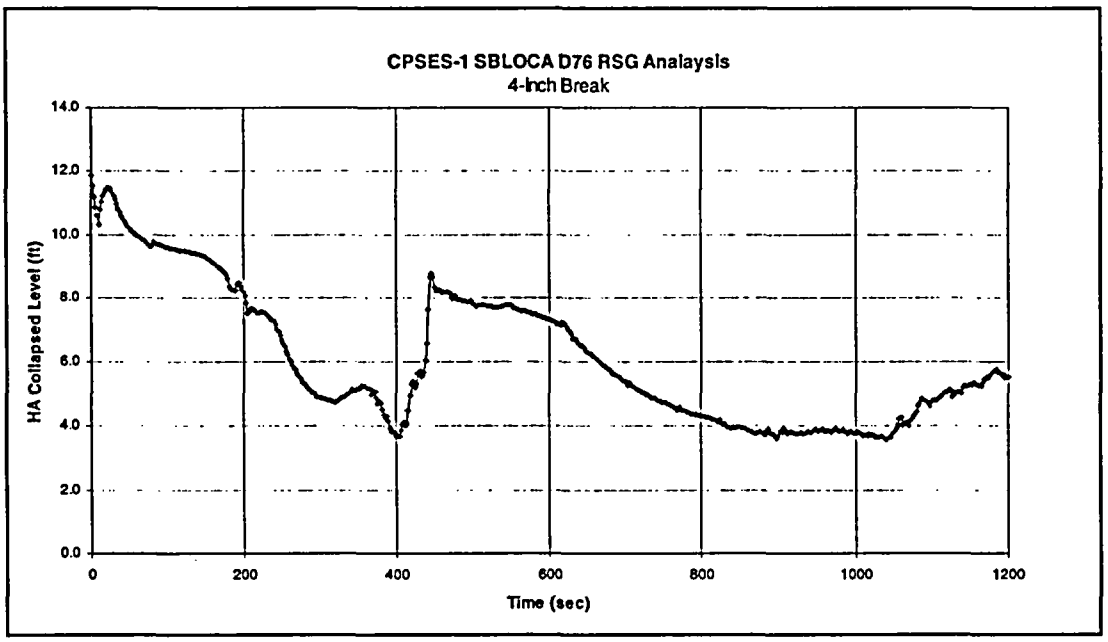

Figure 4.6 - Hot Assembly colapsed Water Level

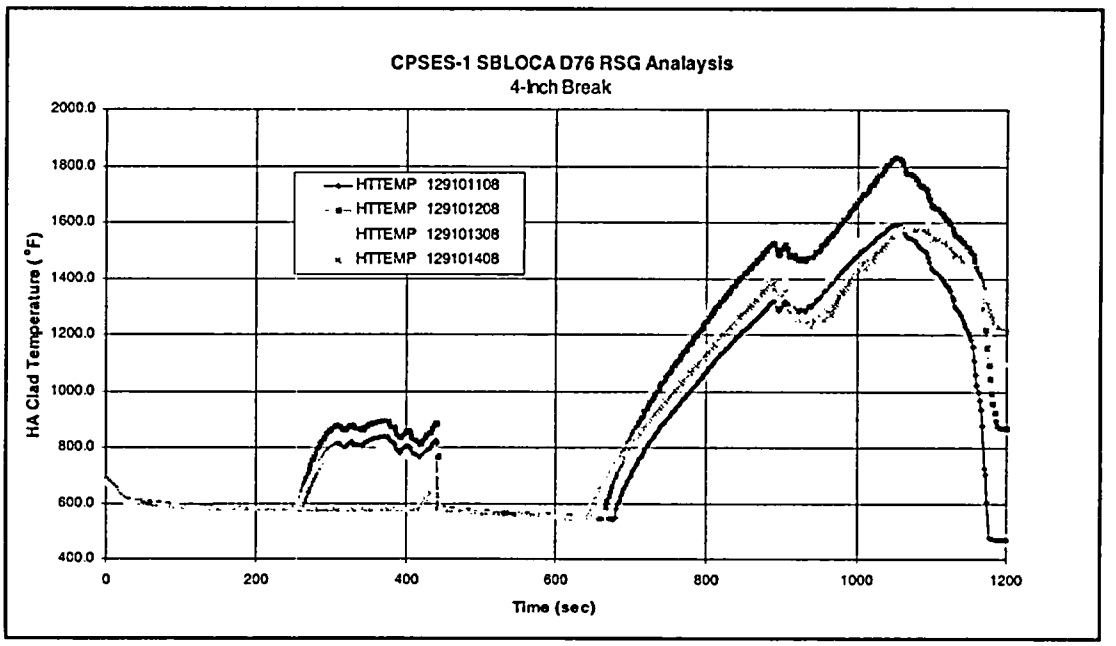

Figure 4.7 - Hot Assembly Clad Temperatures

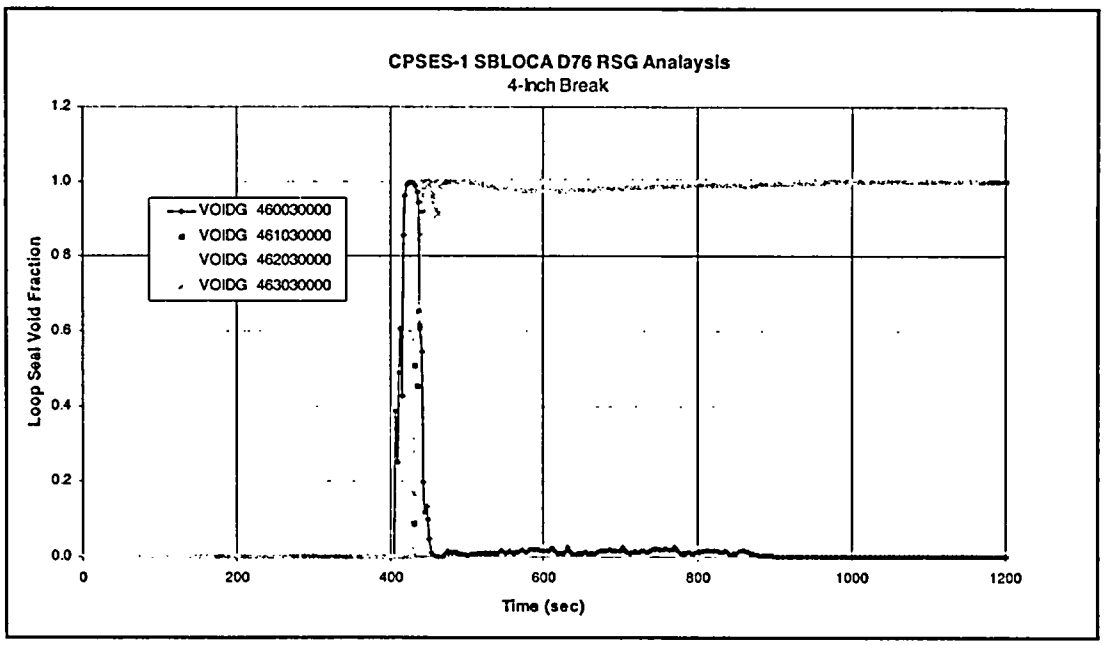

Figure 4.8 - Loop Seal Void Fractions

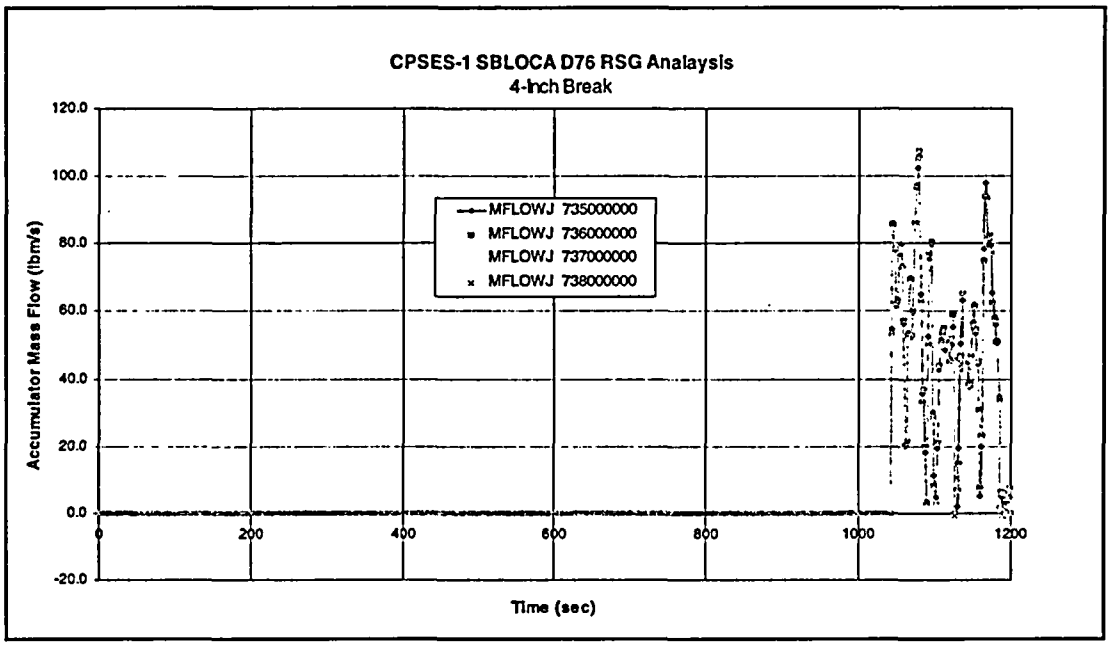

Figure 4.9 - Accumulator Flow Rates

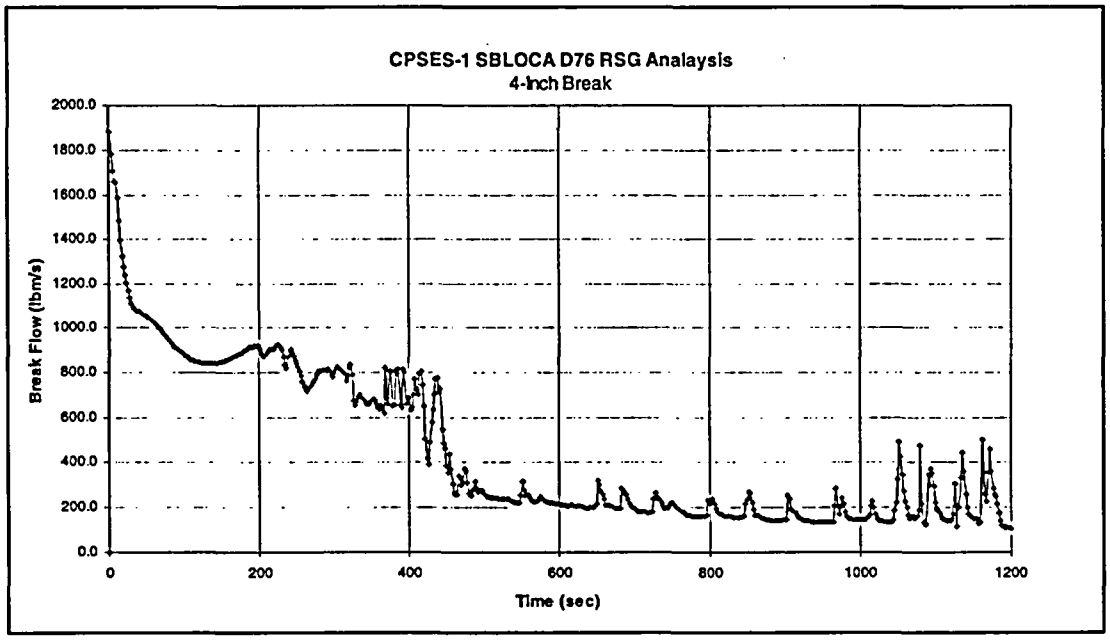

Figure 4.10 - Break Flow

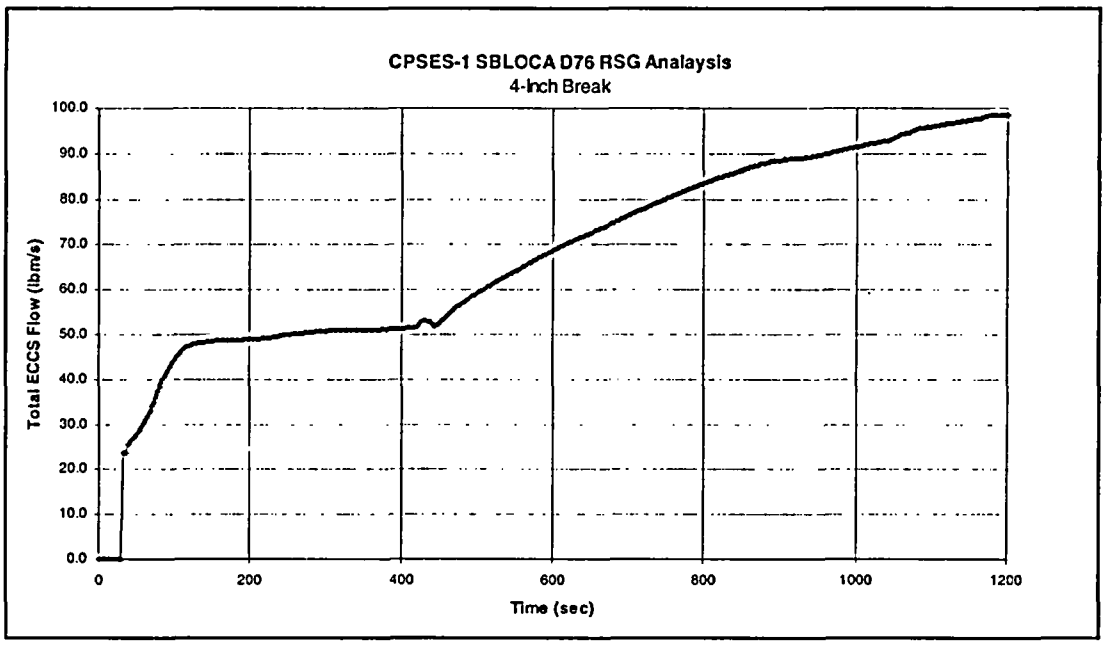

Figure 4.11 - Total Pumped ECCS Flow

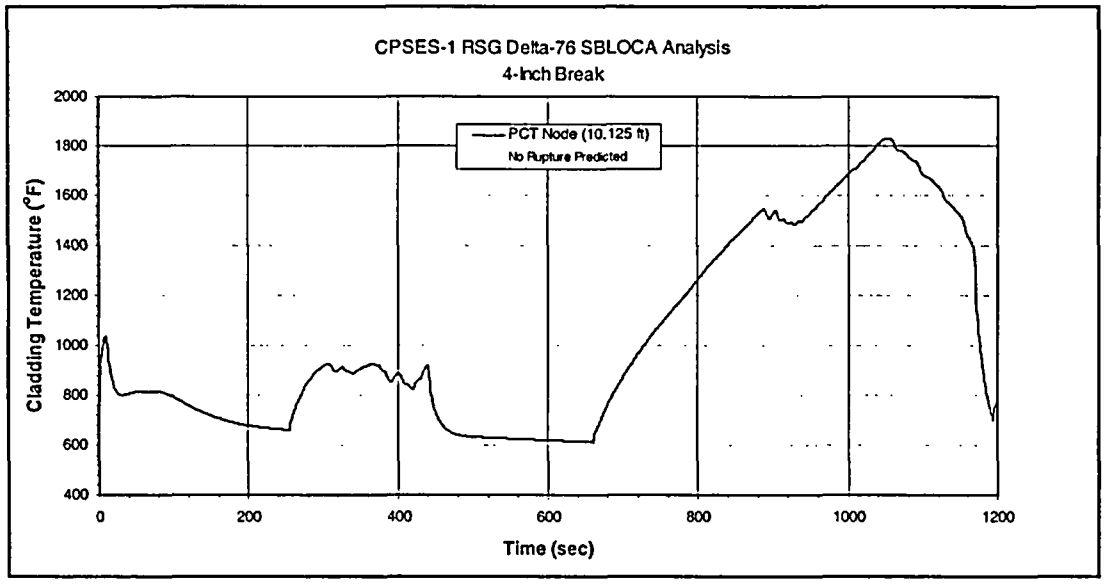

Figure 4.12 - TOODEE2 PCT Node Temperature

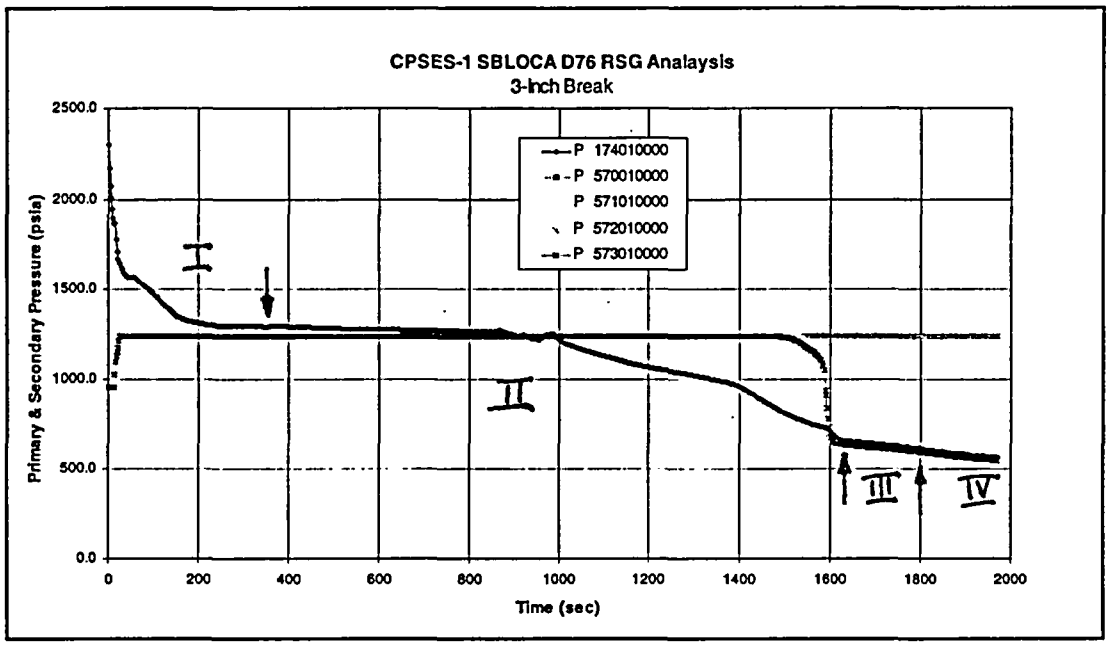

Figure 4.13 - Primary and Secondary Pressures

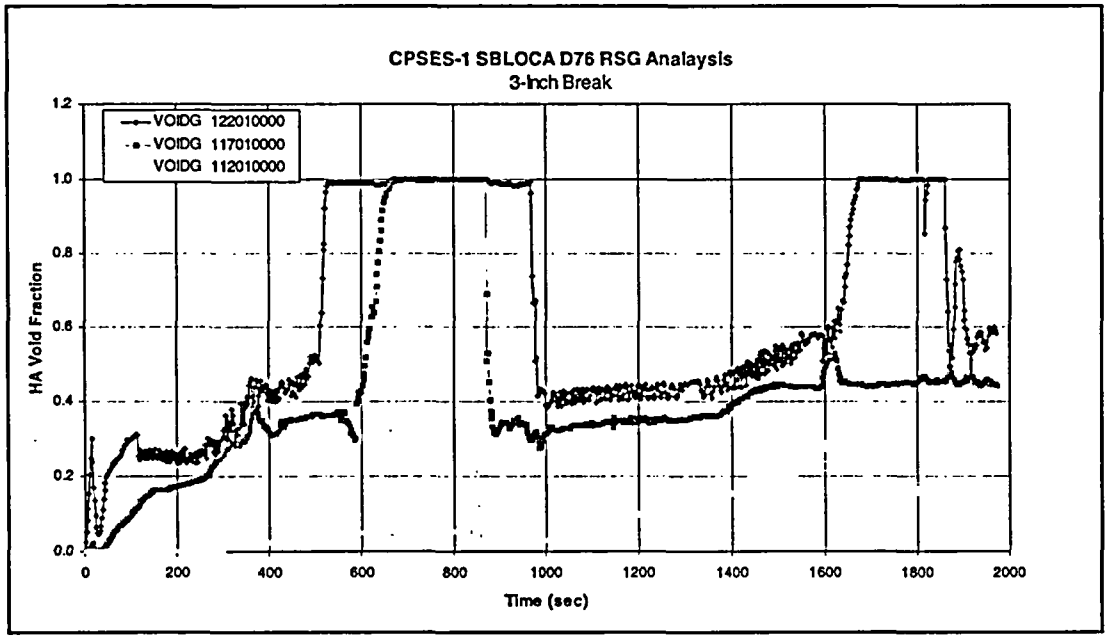

Figure 4.14 - Hot Assembly Void Fractions

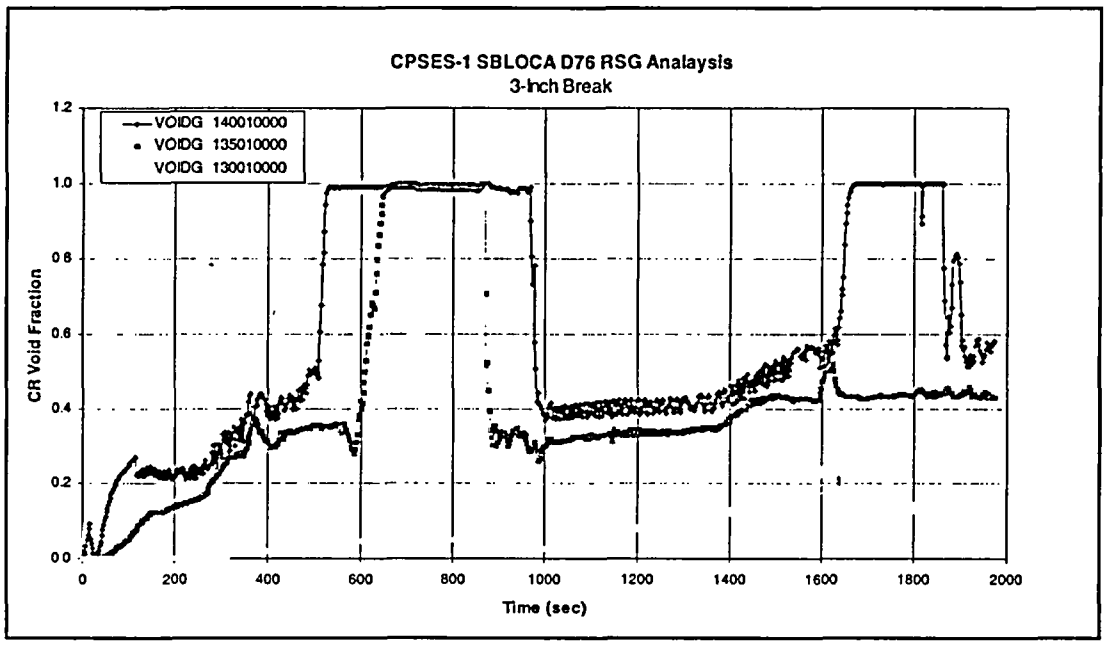

Figure 4.15 - Core Central Region Void Fractions

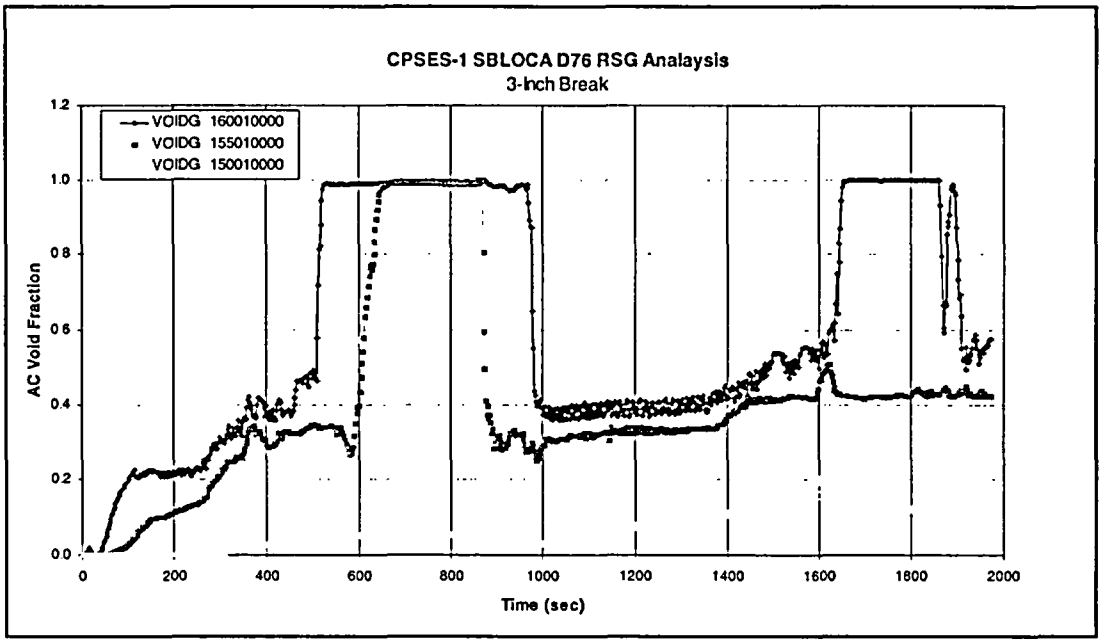

Figure 4.16 - Average Core Void Fractions

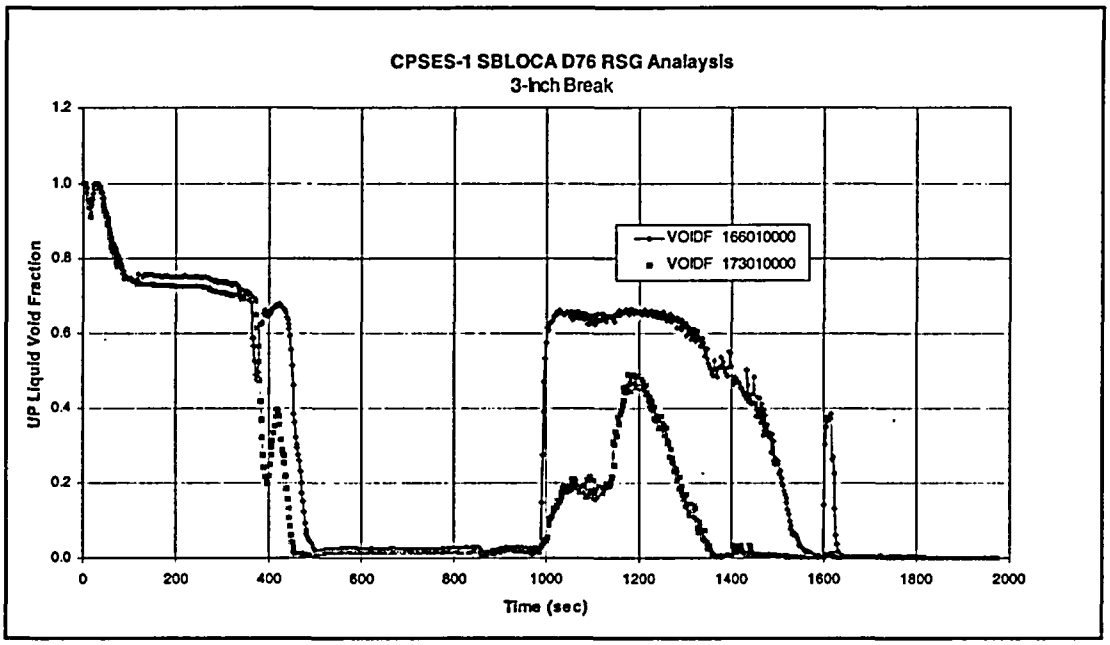

Figure 4.17 - Upper Plenum Liquid Fractions

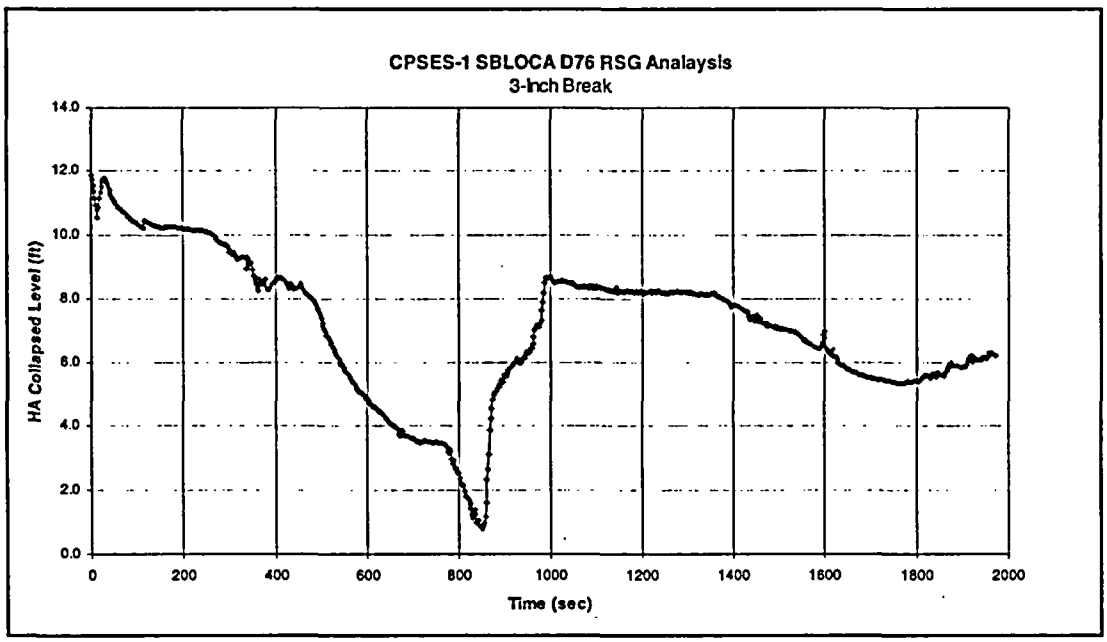

Figure 4.18 - Hot Assembly Collapsed Liquid Level

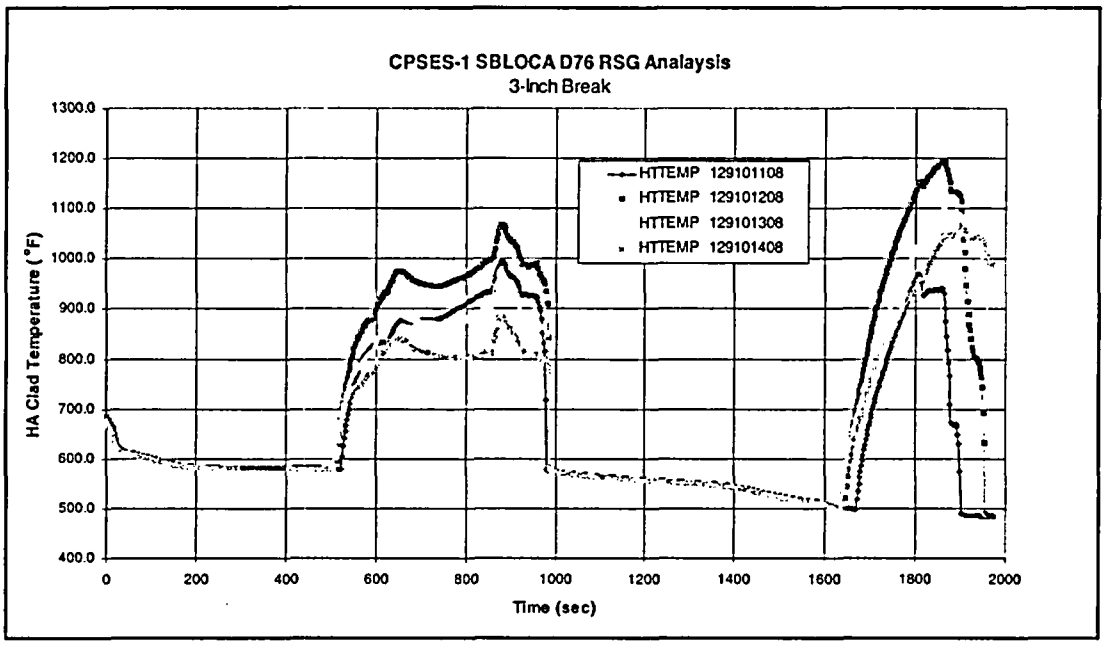

Figure 4.19 - Hot Assembly Clad Temperatures

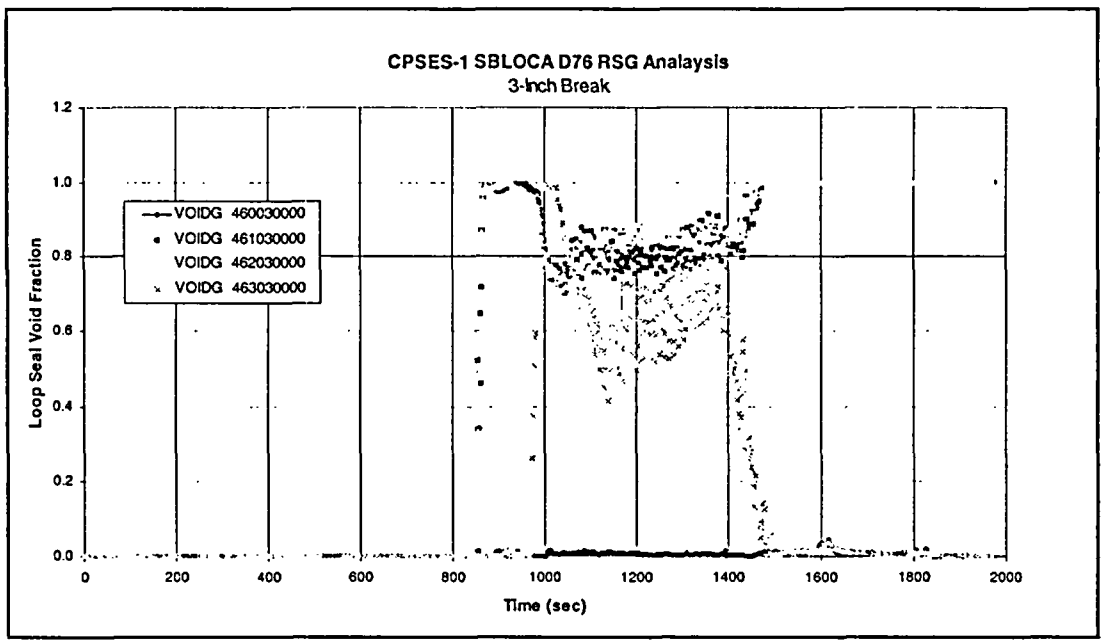

Figure 4.20 - Loop Seal void Fractions

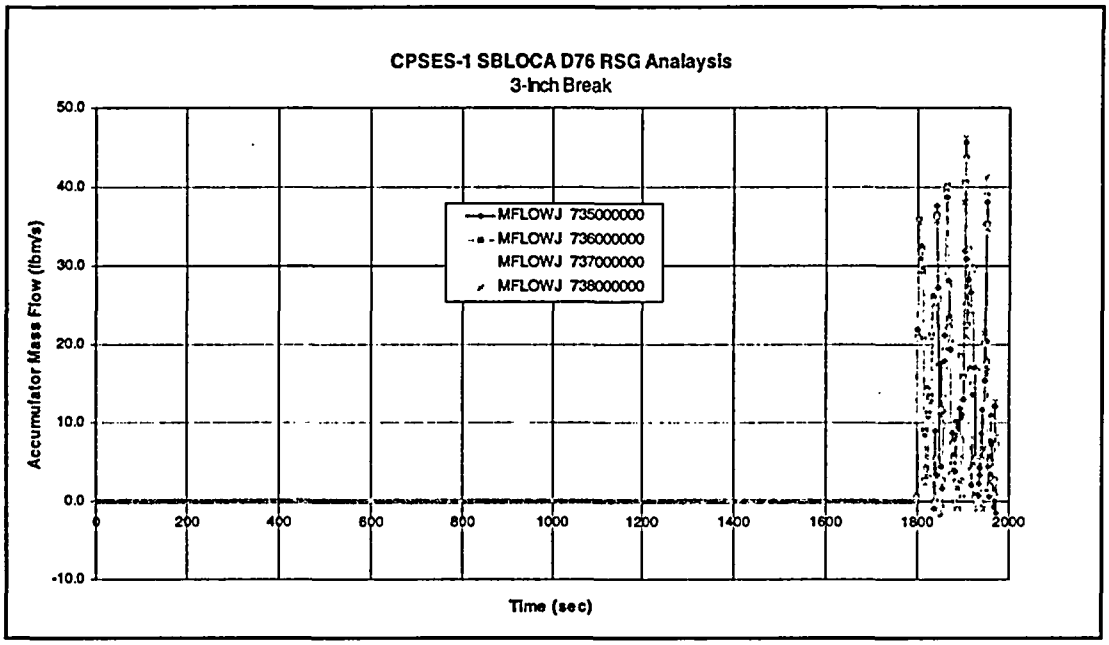

Figure 4.21 - Accumulator Flow

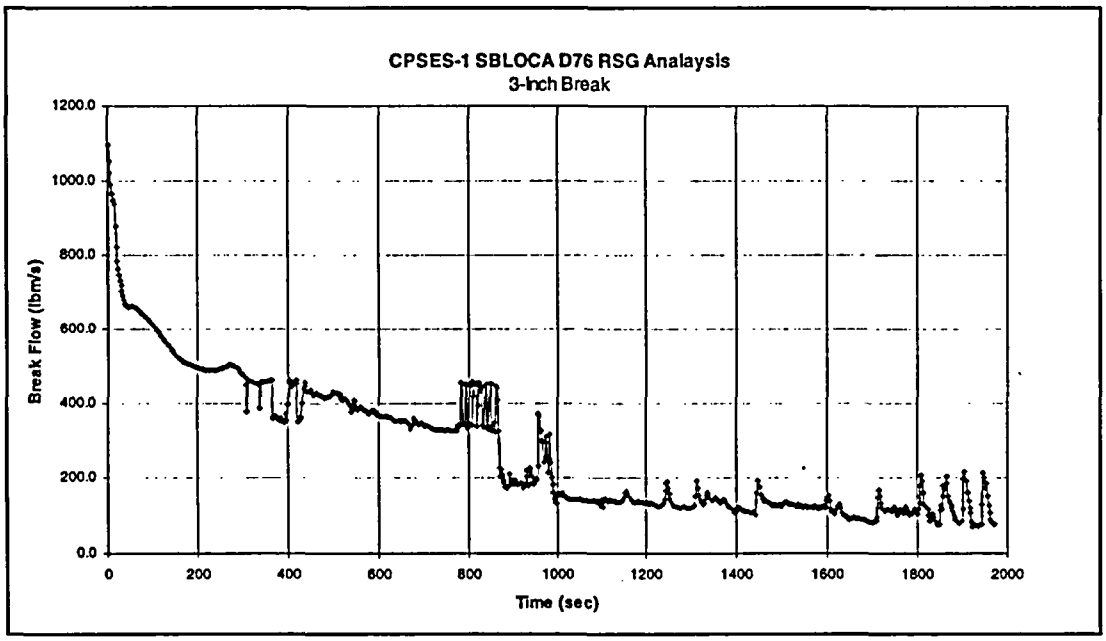

Figure 4.22 - Break Flow

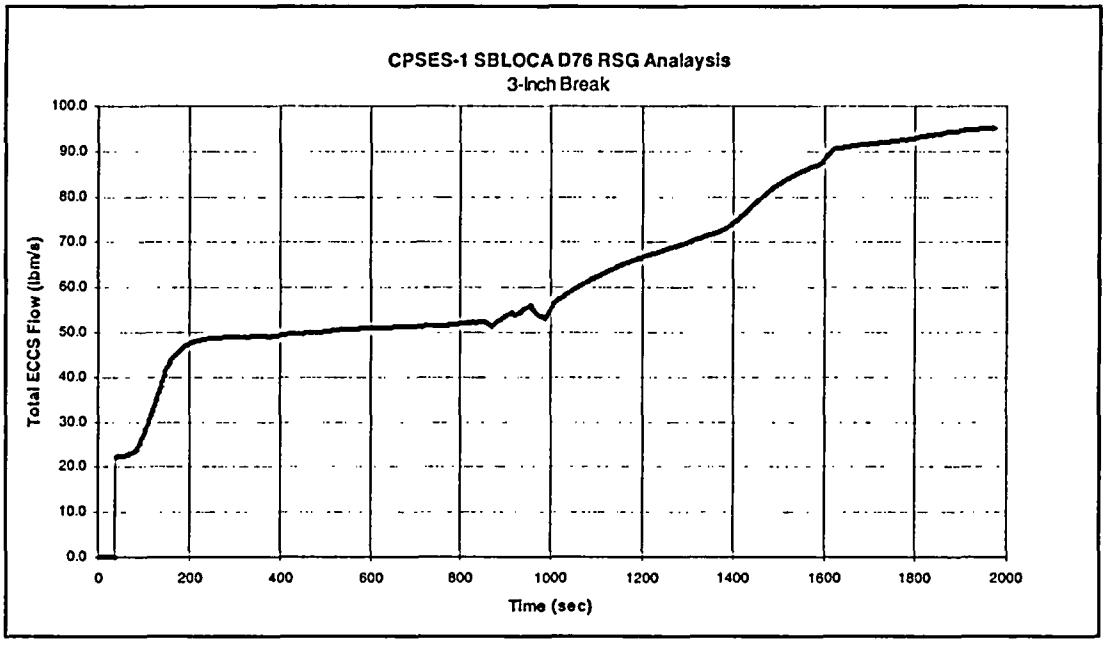

Figure 4.23 - Total Pumped ECCS Flow

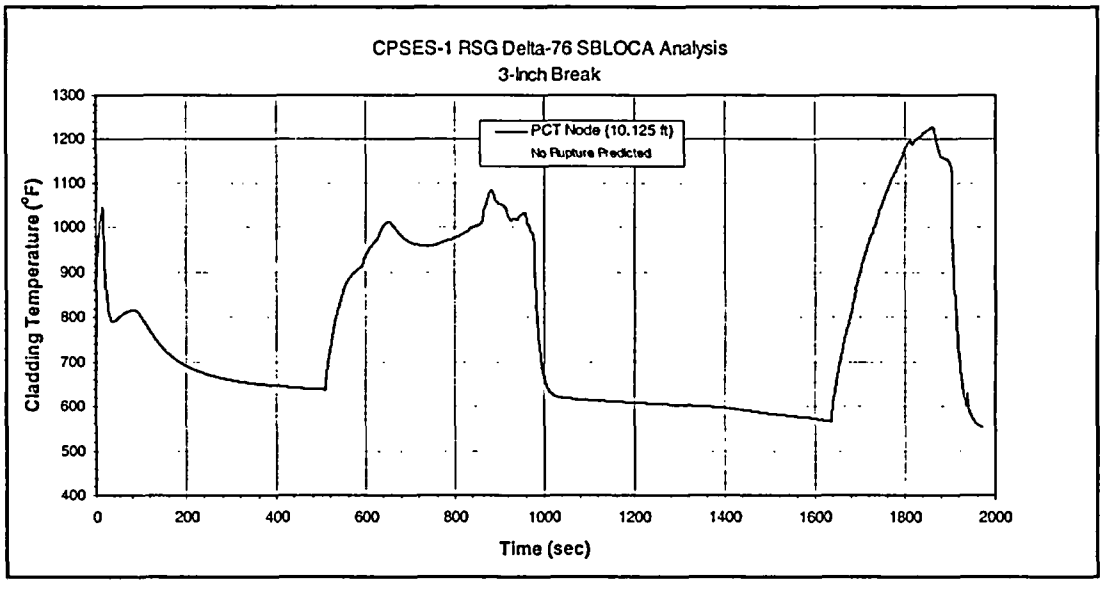

Figure 4.24 - TOODEE2 PCT Node Clad Temperature

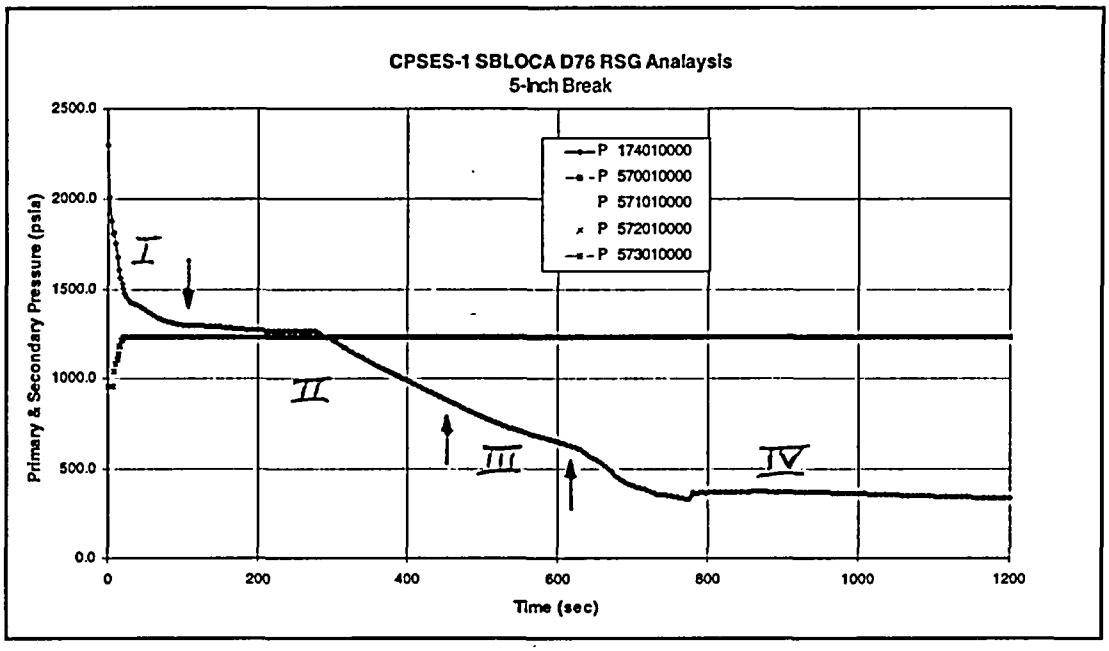

Figure 4.25 - Primary and Secondary Pressures

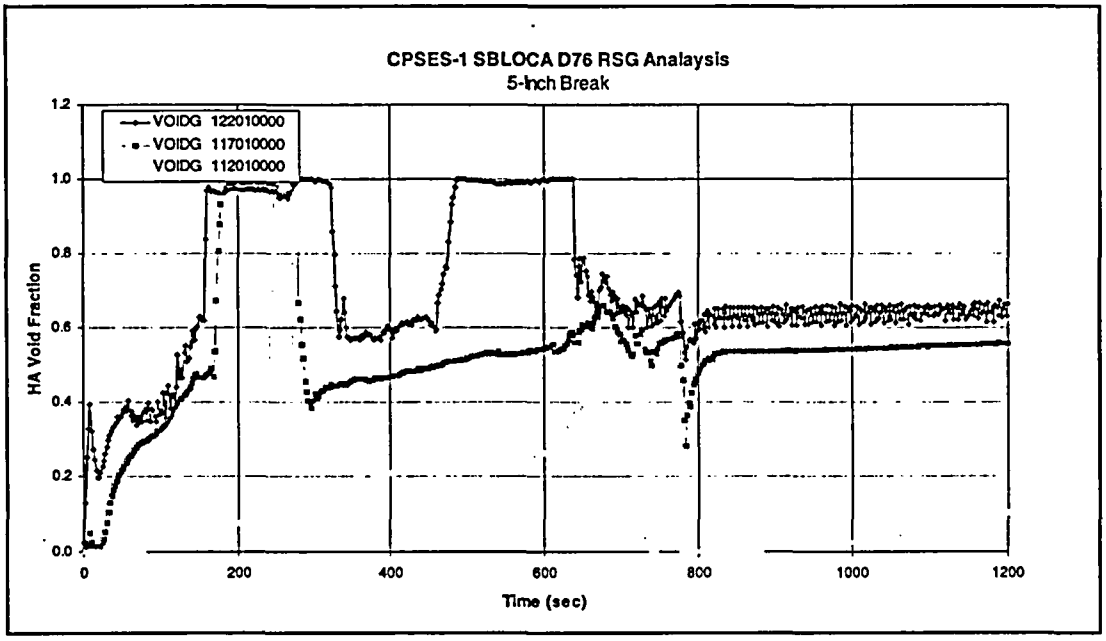

Figure 4.26 - Hot Assembly Void Fractions

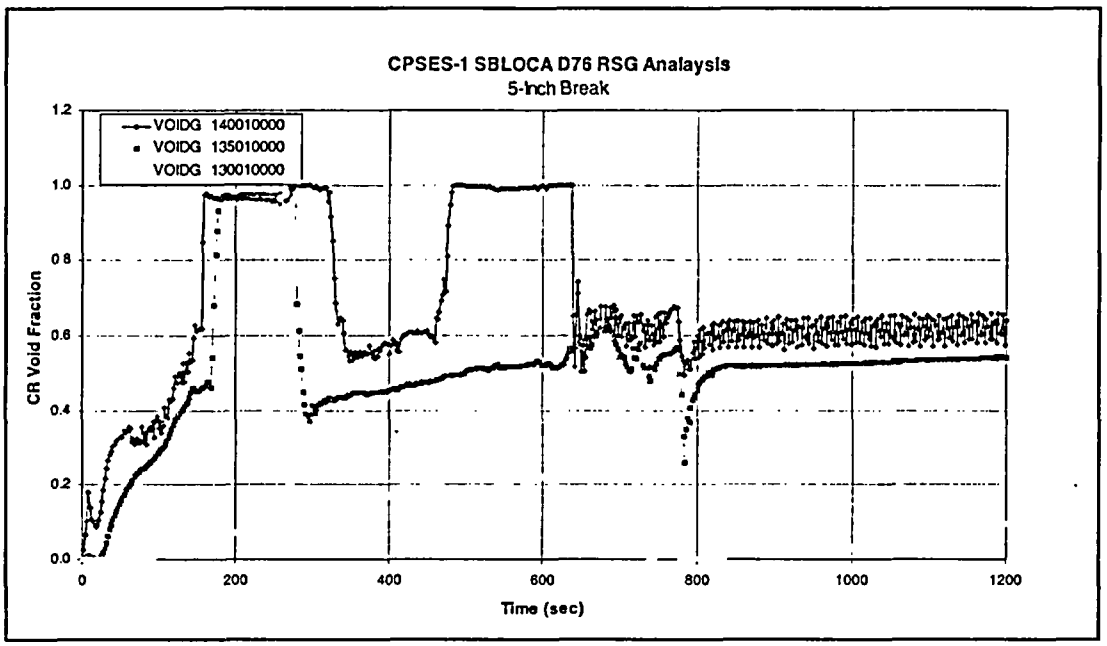

Figure 4.27 - Central Core Region Void Fractions

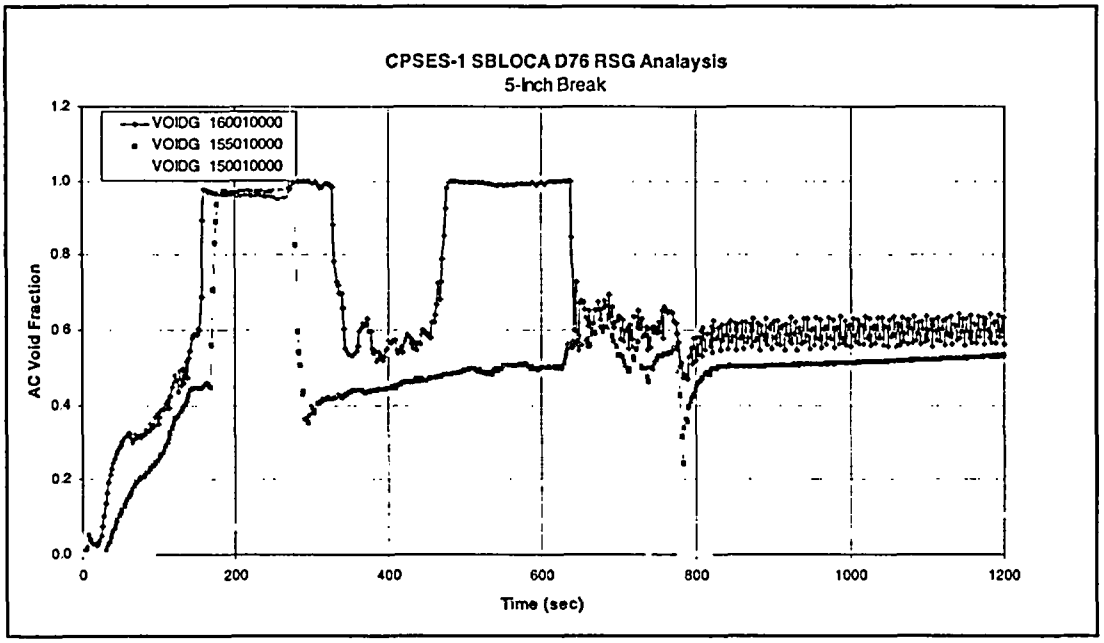

Figure 4.28 - Average Core Void Fractions

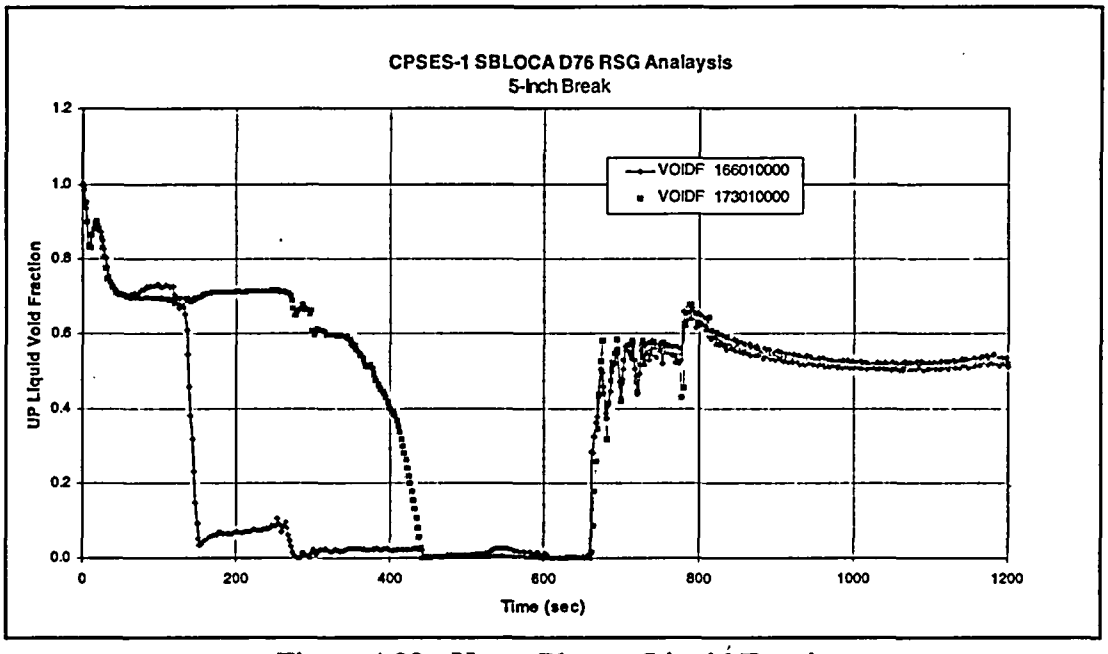

Figure 4.29 - Upper Plenum Liquid Fractions

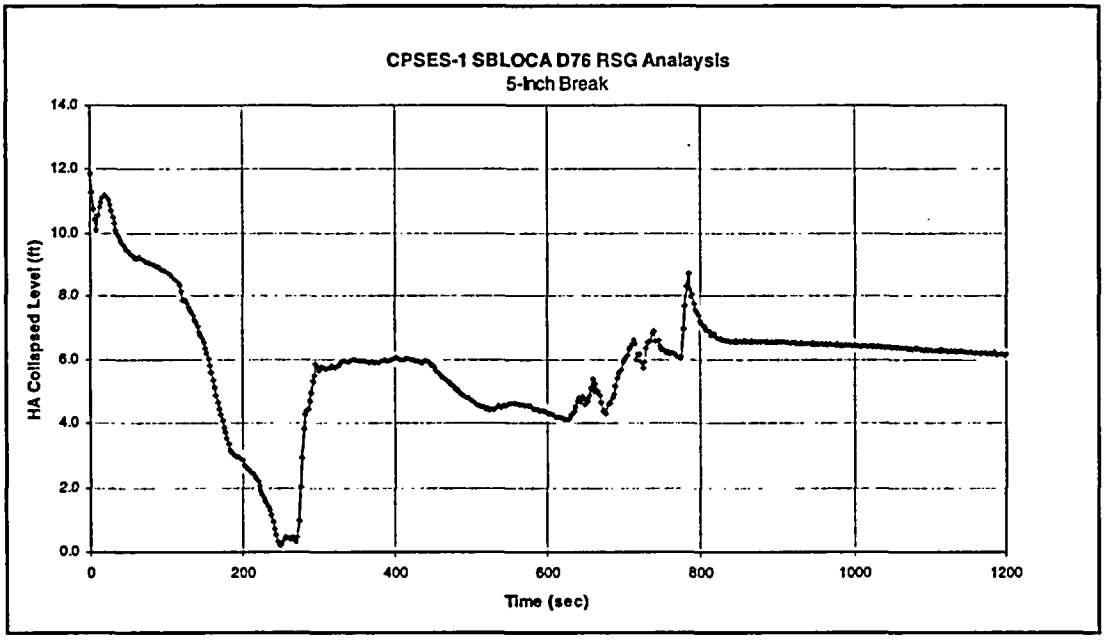

Figure 4.30 - Hot Assembly Collapsed Liquid Level

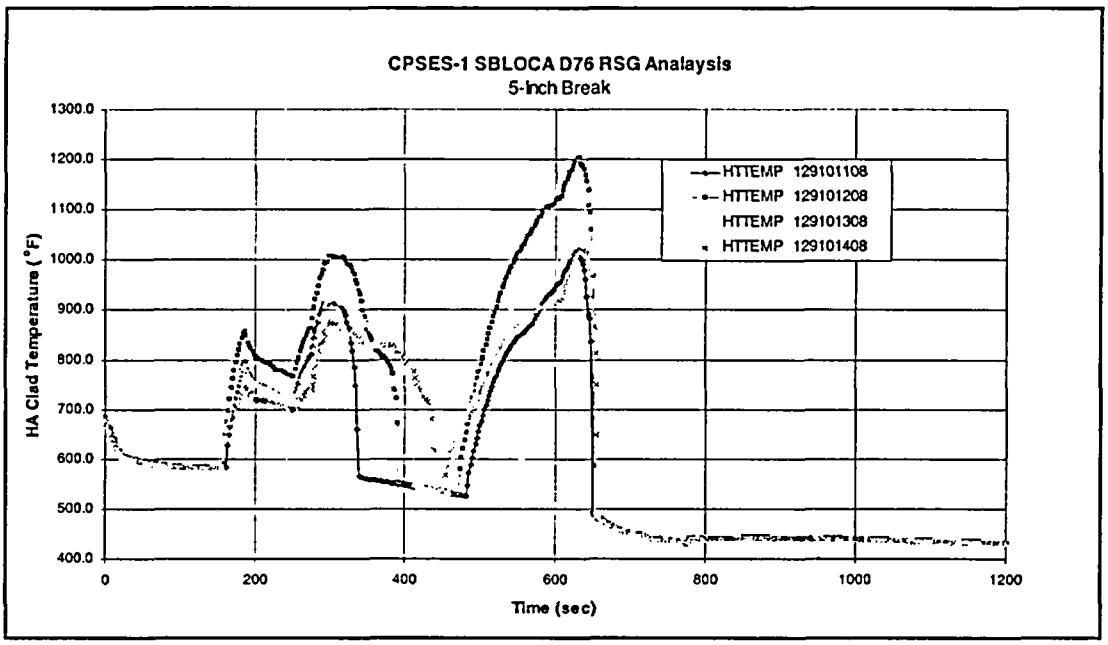

Figure 4.31 - Hot Assembly Clad Temperature

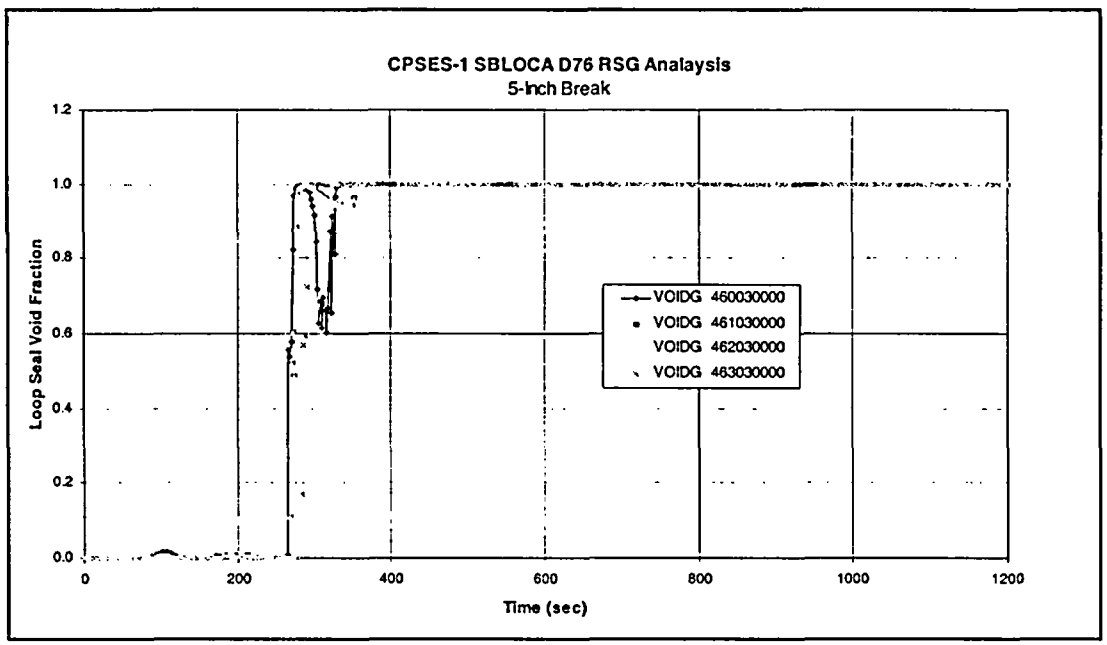

Figure 4.32 - Loop Seal Void Fractions

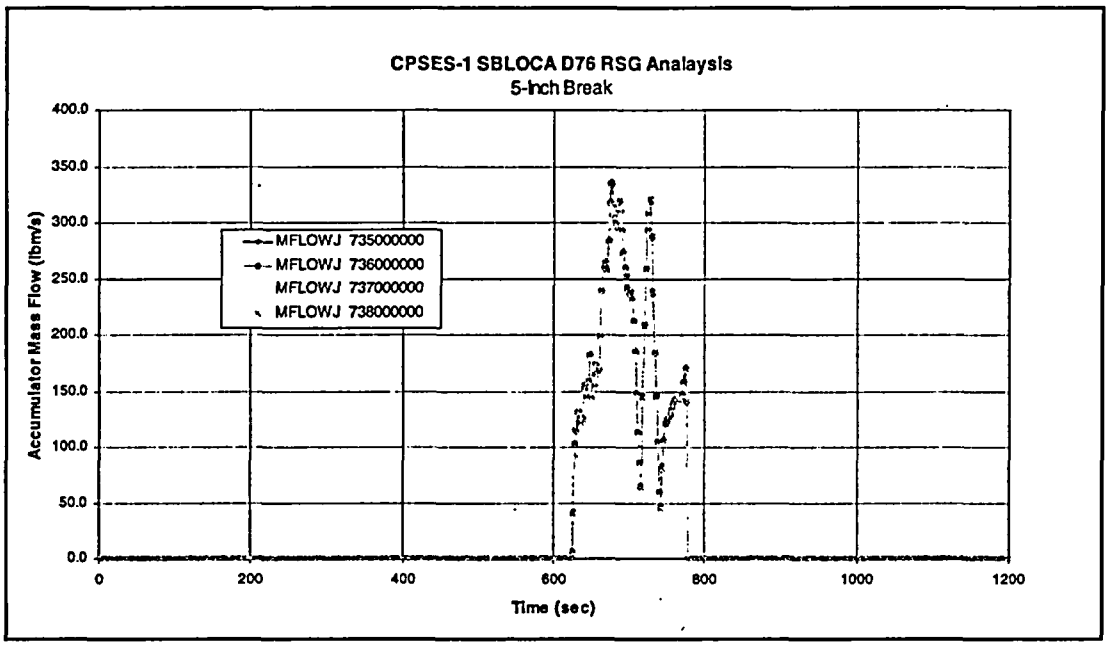

Figure 4.33 - Accumulator Flow

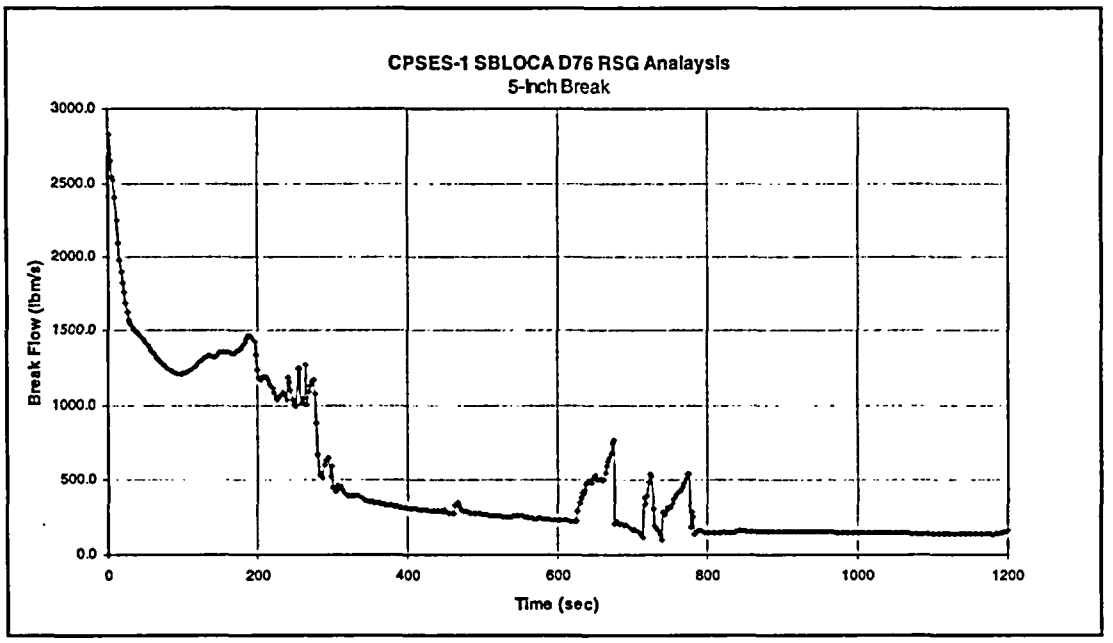

Figure 4.34 - Break Flow

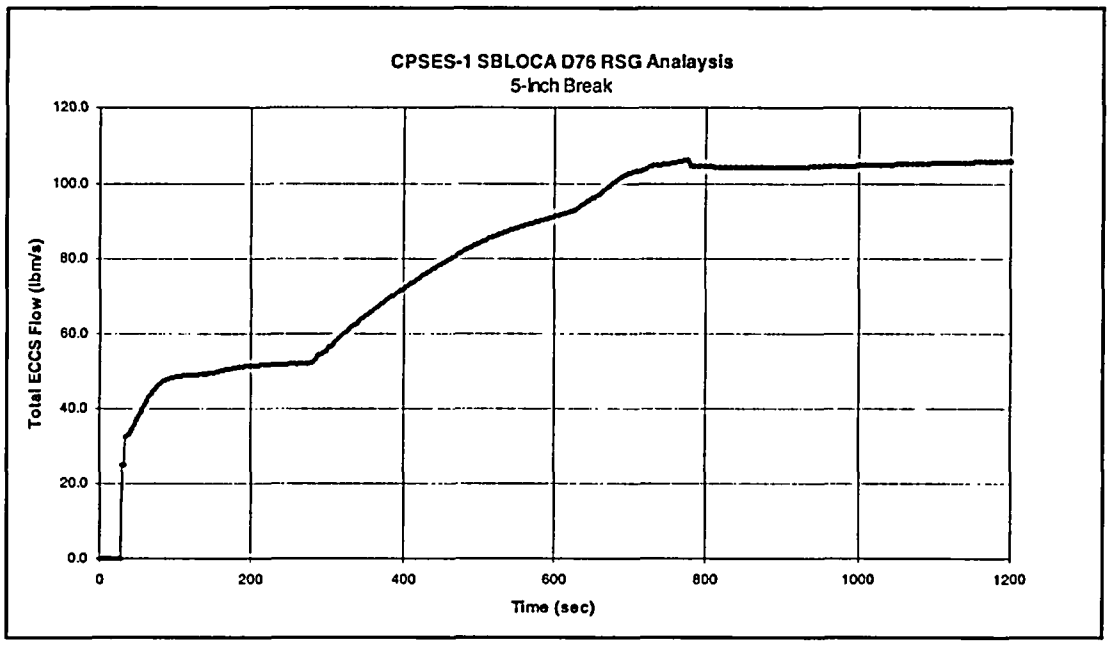

Figure 4.35 - Total Pumped ECCS Flow

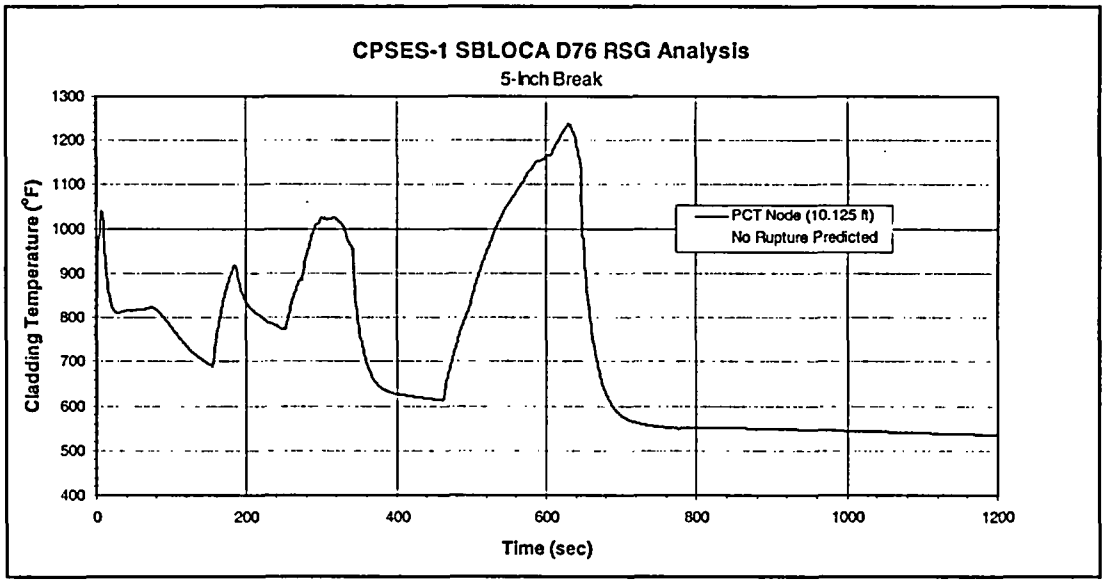

Figure 4.36 - TOODEE2 PCT Node Temperature

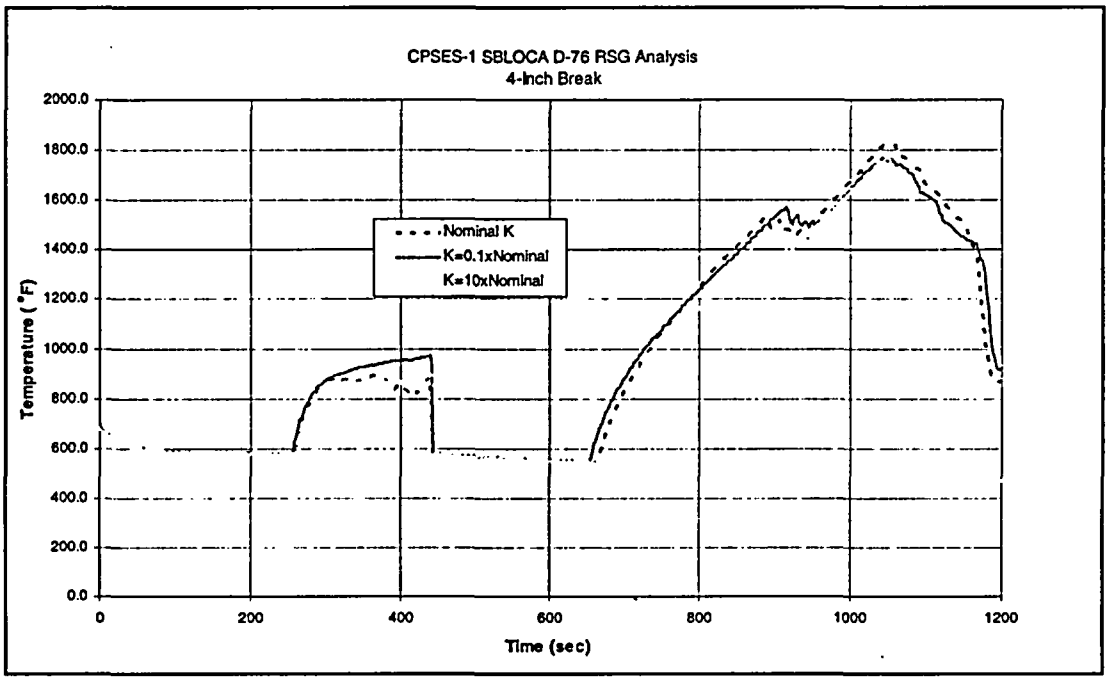

Figure 4.37 - ANF-RELAP Clad Temperatures for Crossflow Study

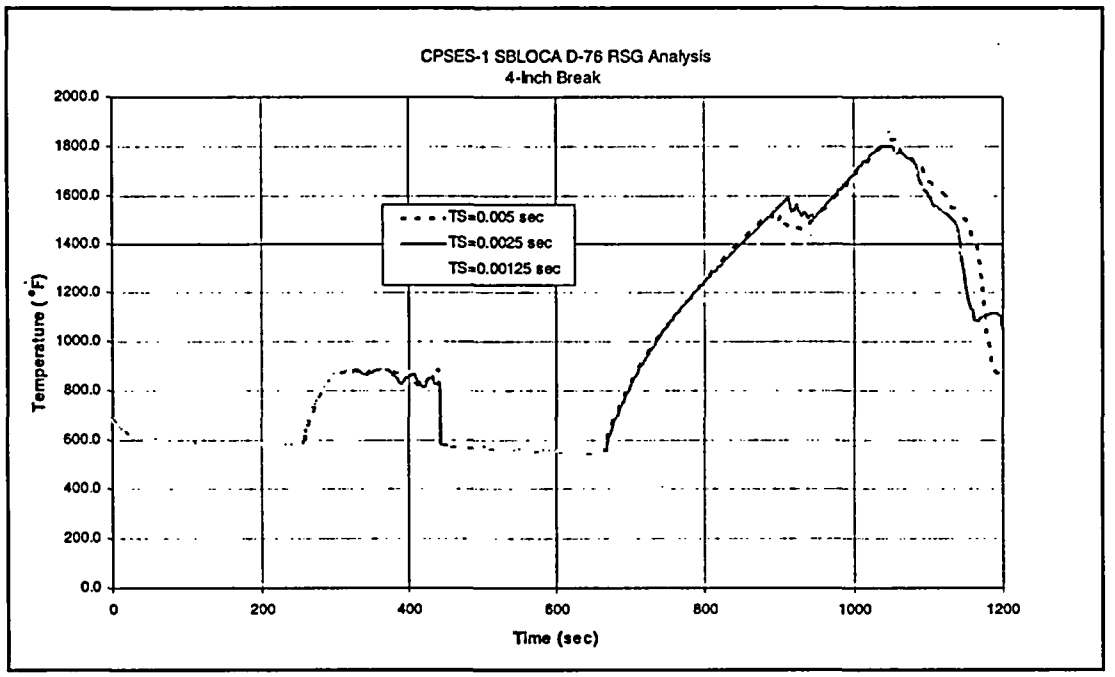

Figure 4.38 - ANF-RELAP Clad Temps. for 4-inch Break Time Step Sensitivity Study

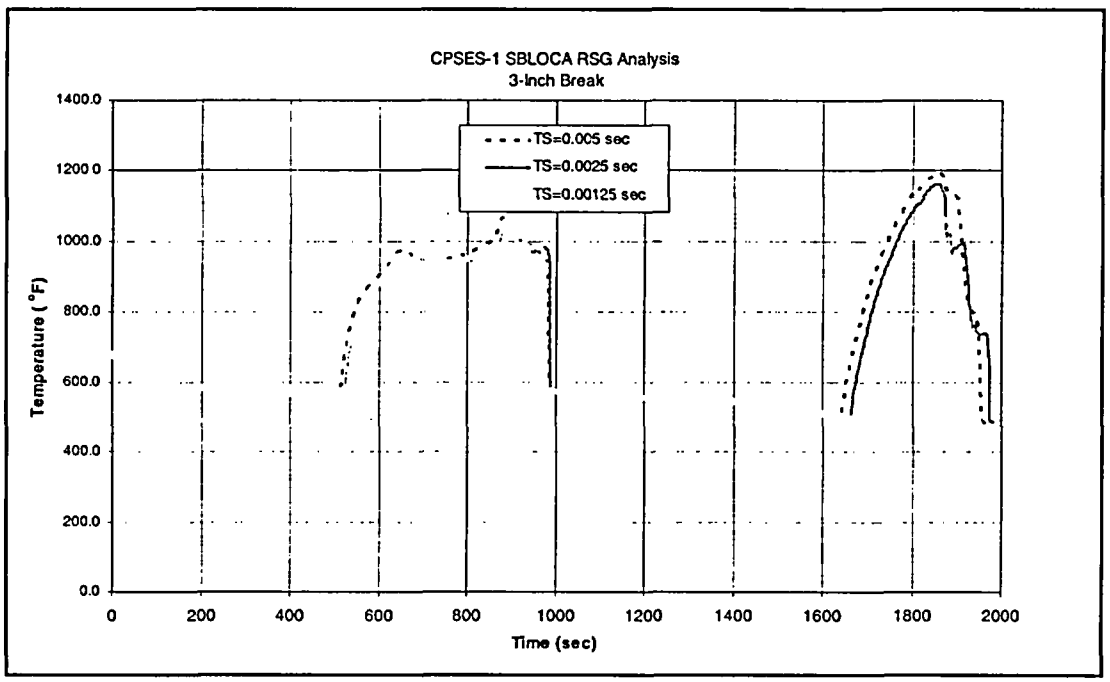

Figure 4.39 - ANF-RELAP Clad Temps. for 3-inch Break Time Step Sensitivity Study

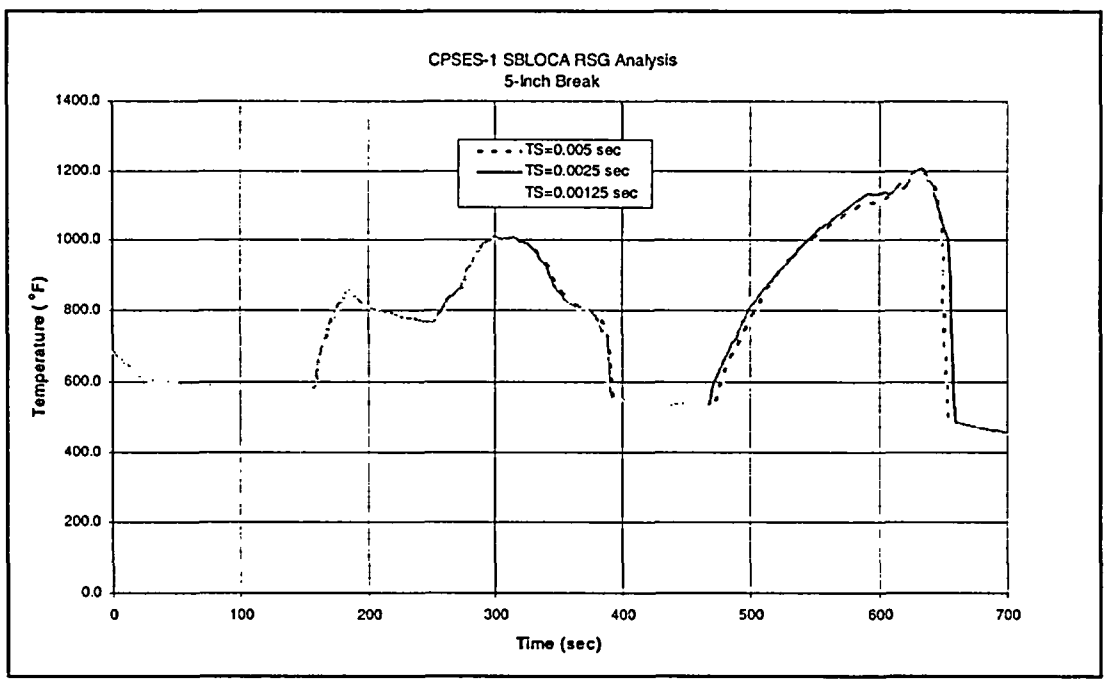

Figure 4.40 - ANF-RELAP Clad Temps. for 5-inch Break Time Step Sensitivity Study

CHAPTER 5

LARGE BREAK LOCA MODEL CHANGES

TXU Power's large break LOCA (LBLOCA) methodology (Reference 1) consists of a series of computer codes which are linked together to perform the large break loss-of-coolant analysis to demonstrate plant and fuel design conformance to 10CFR50.46 criteria and Appendix K requirements. The SEM/PWR-98 computer codes and the information transfers are illustrated schematically in Figure 2.1 of Reference 1. The two computer codes requiring input changes due to the introduction of the RSGs are RELAP4 and REFLEX.

As with the counterpart small break LOCA discussion, and for the reasons discussed in Chapter 3, only the differences between the current input models, which are applicable to the D-4 and D-5 steam generators and the proposed input models, which are applicable to the Δ-76 RSGs are addressed in this chapter. For the LBLOCA methodology, the only significant differences between these models are those associated with the steam generator geometrical inputs.

The proposed Δ-76 RELAP4 and REFLEX steam generator models have the same nodalization structure of the D-4 models, depicted in Figures 2.2 and 2.5 of Reference 1, respectively. The Δ-76 model is essentially the same as the D-4 model. Only differences in the generators themselves, which are described in Chapter 2, were used to change the steam generator model: the nodalization itself was unchanged. The Δ-76 geometrical information is based on TXU Power's RETRAN model (Reference 8), which is shown in Figure 3.1. This was the same approach taken in the development of the SBLOCA model.

As can be seen by comparing Figure 3.1 and Figures 2.2 and 2.5 of Reference 1, the mapping of information from the more detailed RETRAN model into RELAP4 and REFLEX is straight- forward.

Suffice it to re-iterate that Figures 2.2 and 2.5 remain unchanged, along with the entire nodalization of the LBLOCA methodology and simply the nodal geometrical information is changed to refect the Δ-76 rather than the D-4 steam generators.

CHAPTER 6

LARGE BREAK LOCA DEMONSTRATION ANALYSES

As in Reference 1, method- and plant-specific issues were systematically considered in order to determine a base case and to thoroughly evaluate the impact of the LBLOCA model changes presented in Chapter 5.

The considerations of the introductory discussion of Chapter 4 also apply to the LBLOCA except there is only one method specific issue to address: the convergence criterion. That issue is analogous to the time step study in the small break, namely, it tests the numerical robustness of the implementation. Thus, the convergence criterion constitutes one of the sensitivity studies addressed in this section. The plant specific issues examined are the same as those examined for the SBLOCA in Chapter 4: break spectrum and single failure. Reference1 also has a detailed discussion on the rationale for sensitivities performed for a previous implementation of this methodology, so it too provides the basis for the cases examined here.

6.1 BASE CASE ANALYSIS

This section presents licensing analysis results for a Double-Ended Guillotine break in the discharge line of the Reactor Coolant Pump. This break location has been generically shown to be the most limiting (e.g., Reference 10). The axial power shape, the fuel rod exposure and the remaining fuel parameters used in this base case were taken from the reload analysis for CPSES-1 cycle 11.

The accident assumptions are summarized in Table 6.1 and the initial conditions are summarized in Table 6.2.

The major assumptions are that a DEG break occurs at 0.03 seconds with coincident loss of offsite power. ECCS injection into the broken loop is lost, and is postulated to spill directly to the containment. Loss of one train of low pressure pumped injection (residual heat removal pumps, RHR) is the postulated single failure as required by 10 CFR 50, Appendix K. (In a sensitivity study, an alternative single failure, the loss of a diesel-generator resulting in the loss of one full train of ECCS, is examined.) Thus, for this base case, two high head centrifugal charging pumps, two intermediate head safety injection pumps and one low pressure high flow residual heat removal pump along with three accumulators are available to mitigate the accident. Containment pressure is minimized in accordance with Branch Technical Position CSB 6-1 (Reference 11), "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Minimization of containment pressure is done by minimizing initial pressure and temperature and maximizing free volume and heat sinks. Furthermore, containment safeguards are also assumed to function as designed while consistent with the single failure; i.e., two trains of containment sprays are available for the base case. (Only one train of spray is considered in the single failure sensitivity case, the other train of spray taken out by a postulated failure of the diesel-generator.) The fan coolers are disabled on the SI signal as per design.

Ten percent of the steam generator tubes are assumed plugged for this analysis. This assumption is made to support the potential need for operation under these circumstances and is a conservative assumption for fewer obstructed tubes.

All of the above assumptions are identical to those of previous implementations of this methodology (Reference 1).

The first data column in Table 6.3 summarizes the timing of significant events for this case. This table should assist in the review of the following figures, which present key results.

Figures 6.1 and 6.2 show reactor power and net reactivity following the accident during the system blowdown phase. The reactor power decreases rapidly due to negative reactivity from core voiding. Between 3 and 10 seconds the power goes through a local maximum because of an increase in reactivity, which in turn is caused by an increase in the liquid fraction in the center of the core (Figure 6.5). The increase in power results from a temporary coolant accumulation in that region, which is associated with a flow reversal (Figures 6.3 and 6.4). Beyond this time, core power follows the 1971 ANS Draft Standard decay heat values.

Figure 6.5 shows mid-core average quality. The figure indicates that core flashing takes place around 2.5 seconds. Again the quality falls between 3 and 10 seconds due to the flow reversal discussed above and evidenced in Figs. 6.3 and 6.4. Shortly after accumulator injection (at approximately 16 seconds, Figure 6.10) the mid-core quality again drops quickly, but begins to increase again right after the drop and is back to 1.0 at approximately 27 seconds.

Figure 6.6 shows the downcomer liquid inventory. The downcomer remains nearly full until almost 5 seconds. As shown in Figure 6.3, the drainage coincides with the decrease and subsequent flow reversal which is caused by the break and occurs starting around 5 seconds as well. After that the downcomer is quickly depleted reaching a minimum inventory at the time the accumulators begin to inject, when it once again begins to fill quickly.

Figure 6.7 shows the total break flow. The flow rapidly accelerates to two-phase critical flow (Moody model) in less than 0.1 second at the pump discharge. Rapid depressurization and flashing limit the initial break flow rates. The break flow rate gradually diminishes as volumes upstream of the break become void.

Figure 6.8 and 6.9 show system and containment pressures respectively. Superimposed on the primary pressure is the seconday pressure showing that the heat transfer direction is reversed at approximately 6.0 seconds. The containment pressure peaks to about 36 psia, approximately 17 seconds into the blowdown. The pressure turns around at this time due to steam condensation on equipment and concrete surfaces. Containment spray comes into play only at approximately 34 seconds, injecting at a constant rate thereafter.

ECCS flow rates are presented in Figures 6.10 through 6.12. The accumulators begin to inject at 16 seconds and are empty at 45 seconds. All pumped ECCS come on at about 28 seconds to account for all delays.

Figure 6.13 shows the heat transfer coefficient at the peak clad temperature (PCT) node. Heat transfer is abruptly degraded as the core flashes at approximately one second into the accident. The blowdown clad temperatures at the PCT node are presented in Fig 6.14.

The core flooding rates are shown in Figure 6.15. The flooding rate does not drop below one inch per second until approximately 90 seconds. The PCT time is approximately 206 seconds.

The metal reaction depth at the hot spot is shown in Figure 6.16.

The PCT node clad temperature history is shown in Figure 6.17. The PCT is calculated to be 1999 °F at 206 seconds, at 10.875 ft. The ruptured node was at elevation 10.125 ft and it occurred at 36.8 seconds. The maximum nodal oxidation was 3.8% with maximum total pin oxidation 0.39%.

D-4 versus Δ-76 LBLOCA Response for the 1.0 DECLG Break:

The effect of the steam generator type on the LBLOCA response is predictable and small. Table 6.8 compares the timing for the sequence of events. The most notable feature of the table is the similarity of the timing for the two cases. The most notable time difference is the slightly earlier occurrence of the PCT in the Δ-76 versus the D-4. This is due to the higher Δ-76 RCS water inventory, which retains more water at the end of the blowdown and therefore quenches sooner. As a result of this, the event is slightly less severe in the Δ-76, whose PCT information is given just at the bottom of the previous section. The D-4 PCT is calculated to be 2040 °F at 218 seconds, at 10.875 ft. The ruptured node was at elevation 10.125 ft and it occurred at 36.8 seconds. The maximum nodal oxidation was 4.3% with maximum total pin oxidation 0.43%.

6.2 SENSITIVITY STUDIES

6.2.1 BREAK SPECTRUM

The most limiting break location has been generically determined (e.g., see Reference 10) to be in the cold leg at the reactor coolant pump discharge. This determination results primarily from the loss of ECCS flow to the core associated with it. Therefore, this cold leg break location remains most limiting for the present evaluation and a worst break location search need not be repeated. This most limiting break location is the one considered in all cases discussed throughout this and all previous implementations of this methodology.

The break size is the first sensitivity issue addressed. The rationale for addressing break size first is that system thermal-hydraulic behavior is largely affected by break size and less dependent on other issues, i.e., the break size is a first order effect, while the others are second order.

Three DEG break sizes are examined by using the break discharge coefficient values of 1.0 (base case), 0.8 and 0.6, respectively.

Split type breaks are also analyzed. Three longitudinal split break sizes are examined: 2.0, 1.6 and 1.2 times the cold leg cross-section area, while maintaining the discharge coefficient at 1.0.

The accident assumptions for this and the other sensitivity studies are summarized in Table 6.1 and the initial conditions are summarized in Table 6.2.

The sequence of events for the break spectrum study is summarized in Table 6.3.

The results of the 0.8 DEG calculation are quite similar to those of the base case (1.0 DEG, Section 6.1), during the various stages of the thermal-hydraulic analysis. The PCT node clad temperature history is shown in Figure 6.18. The PCT is calculated to be 1991 °F at 201 seconds, at 10.875 ft. The ruptured node was at elevation 10.125 ft and it occurred at 36.8 seconds. The maximum nodal oxidation was 3.7% with maximum total pin oxidation 0.40%.

The 0.6 DEG calculation is nearly identical to the one discussed above (0.8 DEG). The PCT node and the ruptured node do not coincide for this calculation either, as shown in Figure 6.19. The PCT node clad temperature history is shown in Figure 6.19. The PCT is calculated to be 1923 °F at 194 seconds, at 10.875 ft. The ruptured node was at elevation 10.125 ft and it occurred at 40.1 seconds. The maximum nodal oxidation was 3.0% with maximum total pin oxidation 0.34%.

The longitudinal split break calculation shows results that are respectively similar to the DEG. For example the 2.0 split PCT is 1993°F (which is similar to the same break area/CD combination of the 1.0 DEG, with a PCT of 1999 °F). The 1.6 split PCT is 1979°F (which is similar to the same

the 1.0 DEG, with a PCT of 1999 °F). The 1.6 split PCT is 1979°F (which is similar to the same break area/CD combination of the 0.8 DEG PCT, with a PCT 1991 °F). The 1.2 split PCT is 1937°F (which is similar to the same break area/CD combination of the 0.6 DEG, with a PCT of 1923 °F).

Results of this sensitivity study are summarized in Table 6.5. The conclusion of this study is that the most limiting break is a Double-Ended Guillotine with a 1.0 discharge coefficient located in the main coolant pump discharge. Future studies will be performed using 1.0 as the limiting discharge coefficient and assuming a Double-Ended Guillotine break.

6.2.2 SINGLE FAILURE

The competing single failures for the large break loss-of-coolant accident analyses have been determined by experience (Reference 2). These are either: (a) the loss of one ECCS injection train or (b) the loss of 1 train of low pressure injection. A sensitivity study is performed to verify which of these scenarios is the most limiting. The base case analysis of Section 6.1 assumed the failure one train of low pressure pumped injection (1 residual heat removal pump, RHR) as the single failure required by 10 CFR 50, Appendix K. This sensitivity study examines an alternative single failure, namely a postulated failure of a diesel-generator. This postulated single failure will result in the loss of one full train of ECCS, assuming loss of offsite power. Thus, for this sensitivity case, one high head centrifugal charging pump, one intermediate head safety injection pump and one low pressure high flow residual heat removal pump along with three accumulators are available to mitigate the accident. Containment pressure is minimized in accordance with Branch Technical Position CSB 6-1 (Reference 11), "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Minimization of containment pressure is done by minimizing initial pressure and temperature and maximizing free volume and heat sinks. Furthermore, containment safeguards are also assumed to function as designed and to be consistent with the assumed single failure; i.e., for this sensitivity case

only one train of containment spray is available, the other is taken out by the postulated failure the diesel-generator. The fan coolers are also disabled in this case on the SI signal as per design. The rationale for selecting this case is to examine the trade-off between the deleterious effect on the peak clad temperature of: (a) a lower containment pressure as in the base case, where both trains of containment spray pumps work versus, (b) a lesser ECCS injection into the core as in this sensitivity case, but where containment back pressure can be higher due to the loss of one train of spray pumps.

The sequence of events for the single failure of 1 train of ECCS is summarized in Table 6.4. Results of this sensitivity study are summarized in Table 6.6. The PCT is calculated to be 1971 °F at 197 seconds, at 10.875 ft. The ruptured node was at elevation 10.125 ft and it occurred at 36.8 seconds. The maximum nodal oxidation was 3.5% with maximum total pin oxidation 0.38%.

The conclusion from the single failure study is that the single failure assumed in the base case is more limiting. Therefore the single failure of one low pressure injection pump (RHR pump) will be used in future analyses.

6.2.3 CONVERGENCE CRITERION

The base case analysis of Section 6.1 assumed a convergence criterion (Reference 2). This case simply varied the RELAP4 convergence criterion variable ESPW from 0.5 in the base case to 0.25 in this case in order to check the robustness of the methodology. The new PCT was less than 1 ^{0}F lower, so it is concluded that CPSES results are numerically robust and the recommended value for variable ESPW=0.5 is adequate. Results of this sensitivity study are summarized in Table 6.7.

Table 6.1

SUMMARY OF CPSES-1 LARGE BREAK LOCA ANALYSIS ASSUMPTIONS FOR BASE CASE AND SENSITIVITY STUDIES

1. The initial power is 3479.5 MWt, which is the current rated thermal power plus an allowance for the power calorimetric measurement uncertainty.

2. 10% of the steam generator tubes are plugged.

3. Break in reactor coolant pump discharge occurs at 0.03 s.

4. No Credit taken for Reactor trip (no scram reactivity insertion).

5. Three accumulators inject into intact loops on demand.

6. Two high head centrifugal charging pumps, two intermediate head safety injection pumps and one low head high flow residual heat removal pump inject on demand after the appropriate delays. Assumed single failure: 1 train of low pressure injection (RHR). (In a sensitivity study an alternative single failure, namely the loss of one full train of ECCS, taken out by a postulated failure its diesel-generator, is examined.)

7. Containment pressure is minimized in accordance with branch Technical Position CSB 6-1 (Reference 11), "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Minimization of containment pressure is done by minimizing initial pressure and temperature and maximizing free volume and heat sinks. Furthermore, containment safeguards are also assumed to function as designed and to be consistent with the single failure; i.e., two trains of containment sprays are available. The fan coolers are disabled on the SI signal per plant design. Passive heat structures are included. (Only one train of containment spray is considered in the single failure sensitivity case, the other spray train is taken out by a postulated failure of the diesel.)

8. No credit is given for Auxiliary Feedwater.

Table 6.2

SUMMARY OF INITIAL CONDITIONS FOR CPSES-1 LARGE BREAK LOCA BASE CASE AND SENSITIVITY STUDIES

DESCRIPTION	VALUE
o Core Power	3479.5 MWt
o Accumulator Water Volume per Tank	6119 gals
o Accumulator Cover Gas Pressure	623 psig
o Accumulator Water Temperature	88 °F
o Refueling Water Storage Tank Temperature	40 °F
o Initial Loop Flow	10,072 lbm/sec
o Vessel Inlet Temperature	560 °F
o Vessel Outlet Temperature	618 °F
o Reactor Coolant Pressure	2250 psia
o Pressurizer Water Volume	1123 ft^3
o Steam Pressure	1022 psia
o Containment conditions	Table 6.1, Item 7
o Steam Generator Tube Plugging Level	10%
o Single Failure	Loss of 1 RHR train
o Fuel Parameters and Power Shape	Unit 1 Cycle 11

Table 6.3

SEQUENCE OF EVENTS FOR BREAK SPECTRUM[8] STUDY

EVENT	TIME (Seconds)					
	1.0 DEG[9]	0.8 DEG	0.6 DEG	2.0 Split	1.6 Split	1.2 Split
1. Break Opens	0.03	0.03	0.03	0.03	0.03	0.03
2. Main Feedwater Isolated	0.03	0.03	0.03	0.03	0.03	0.03
3. Msiv Closed	0.03	0.03	0.03	0.03	0.03	0.03
4. High Containment Pressure Hi-1 Signal	1.15	1.23	1.40	1.19	1.22	1.27
5. Accumulator Injection, Intact Loop	16.58	16.76	18.43	16.91	17.05	17.51
6. End of Bypass	24.75	25.23	27.73	24.99	25.16	25.67
7. Safety Injection Pumps Inject	28.15	28.23	28.40	28.19	28.22	28.27
8. Bottom of Core Recovery (BOCREC)	38.90	39.46	42.17	39.14	39.33	39.84
12. Accumulator Empty	49.96	50.16	52.10	50.29	50.47	50.96
13. Rod Burst	36.8	36.8	40.1	37.3	37.5	37.9
14. Peak Clad Temperature Reached	206.2	201.3	193.6	204.9	202.2	198.8
15. Calculation Ends	250.0	250.0	250.0	250.0	250.0	250.0

8 All cases: Same power shape and fuel parameters from CPSES-1 Cycle 11, single failure of 1 RHR train.

9 Base case.

Table 6.4

SEQUENCE OF EVENTS FOR SINGLE FAILURE STUDY[10]

EVENT	TIME (Seconds)	
	1 TRAIN RHR[11]	1 TRAIN ECCS
1. Break Opens	0.03	0.03
2. Main Feedwater Isolated	0.03	0.03
3. Msiv Closed	0.03	0.03
4. High Containment Pressure Hi-1 Signal	1.15	1.15
5. Accumulator Injection, Intact Loop	16.58	16.58
6. End of Bypass	24.75	24.75
7. Safety Injection Pumps Inject	28.15	28.15
8. Bottom of Core Recovery (BOCREC)	38.90	38.91
9. Accumulator Empty	49.96	49.96
10. Rod Burst	36.8	36.8
11. Peak Clad Temperature Reached	206.2	196.6
12. Calculation Ends	250.0	250.0

10 All cases: Same power shape and fuel parameters from CPSES-1 Cycle 11, single failure of 1 RHR train, double-ended guillotine break (1.0 DEG).

11 Base case.

Table 6.5

RESULT SUMMARY FOR BREAK SPECTRUM STUDY[12]

BREAK DESCRIPTION	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
1.0 DEG	1999	3.8	0.39
0.8 DEG	1991	3.7	0.40
0.6 DEG	1923	3.0	0.34
2.0 Split	1993	3.7	0.39
1.6 Split	1979	3.6	0.38
1.2 Split	1937	3.1	0.34

12 All cases: Same power shape and fuel parameters from CPSES-1 Cycle 11, single failure of 1 RHR train.

Table 6.6

RESULT SUMMARY FOR SINGLE FAILURE STUDY[13]

SINGLE FAILURE	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
1 Train of RHR (Base Case)	1999	3.8	0.39
1 Train of ECCS	1971	3.5	0.38

Table 6.7

RESULT SUMMARY FOR CONVERGENCE CRITERION STUDY[14]

CONVERGENCE CRITRION	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
EPSW = 0.5	1999	3.8	0.39
EPSW = 0.25	1998	3.8	0.39

13 All cases: Same power shape and fuel parameters from CPSES-1 Cycle 11, double-ended guillotine break (1.0 DEG).

14 All cases: Same power shape and fuel parameters from CPSES-1 Cycle 11, single failure of 1 RHR train, double-ended guillotine break (1.0 DEG).

Table 6.8

SEQUENCE OF EVENTS AND RESULT COMPARISON: Δ-76 VERSUS D-4

EVENT	TIME (SECONDS)	
	Δ-76[15]	D-4
1. Break Opens	0.03	0.03
2. Main Feedwater Isolated	0.03	0.03
3. Msiv Closed	0.03	0.03
4. High Containment Pressure Hi-1 Signal	1.15	1.14
5. Accumulator Injection, Intact Loop	16.58	14.89
6. End of Bypass	24.75	22.74
7. Safety Injection Pumps Inject	28.15	28.14
8. Bottom of Core Recovery (BOCREC)	38.90	36.76
9. Accumulator Empty	49.96	47.99
10. Rod Burst	36.8	34.4
11. Peak Clad Temperature Reached	206.2	218.3
12. Calculation Ends	250.0	250.0

SG TYPE	PCT (°F)	% Oxidation (NODE)	% Oxidation (PIN)
Δ-76	1999	3.8	0.39
D-4	2040	4.3	0.43

15 Base case.

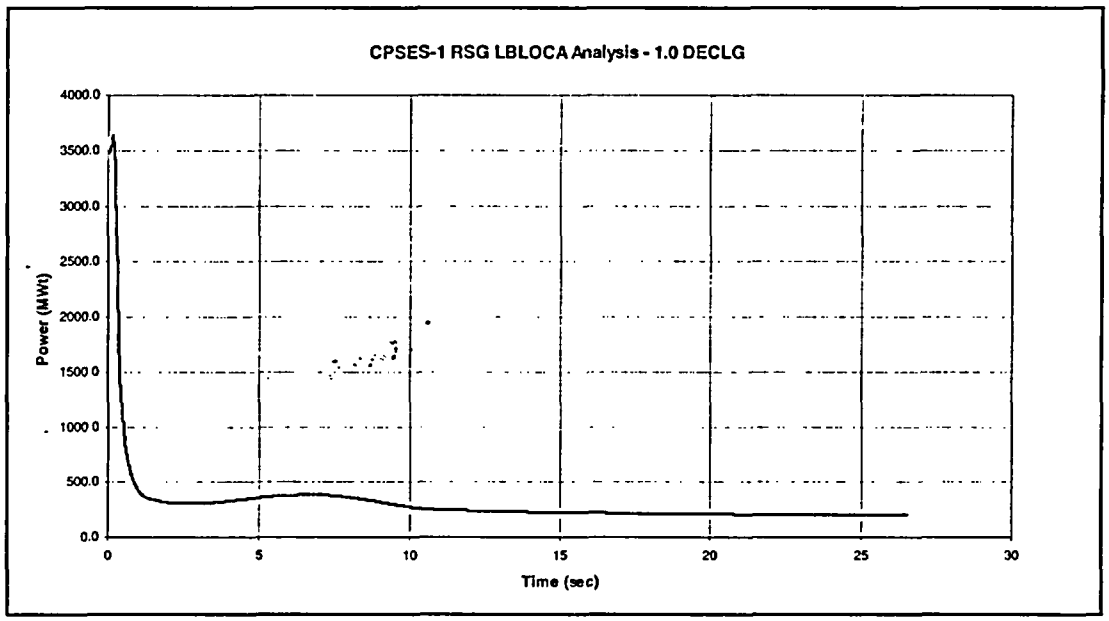

Figure 6.1 - Core Total Power

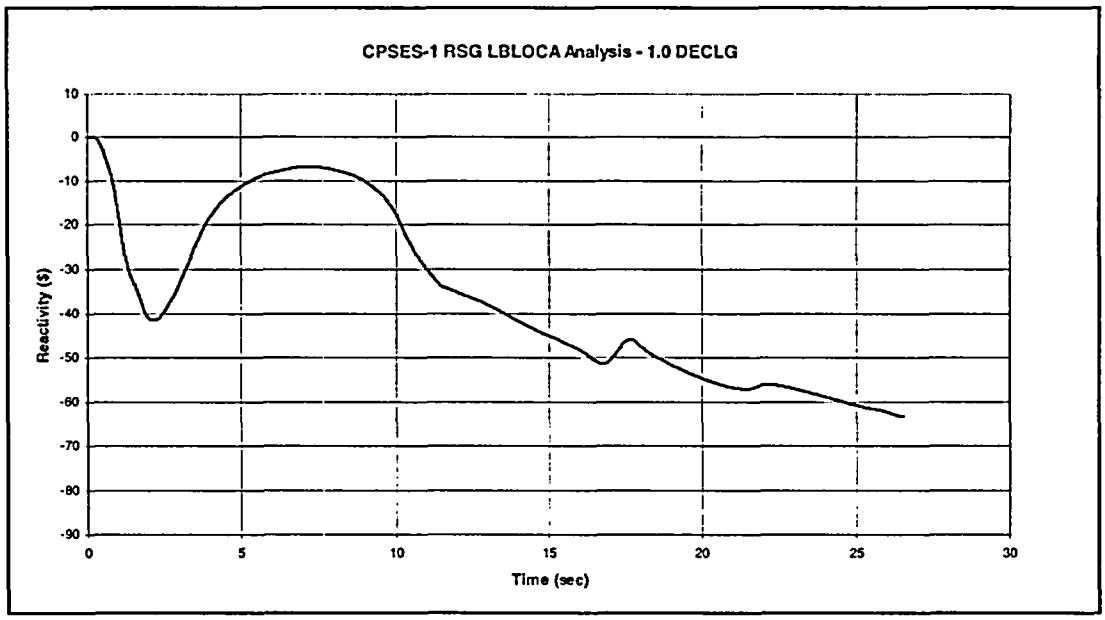

Figure 6.2 - Total Reactivity

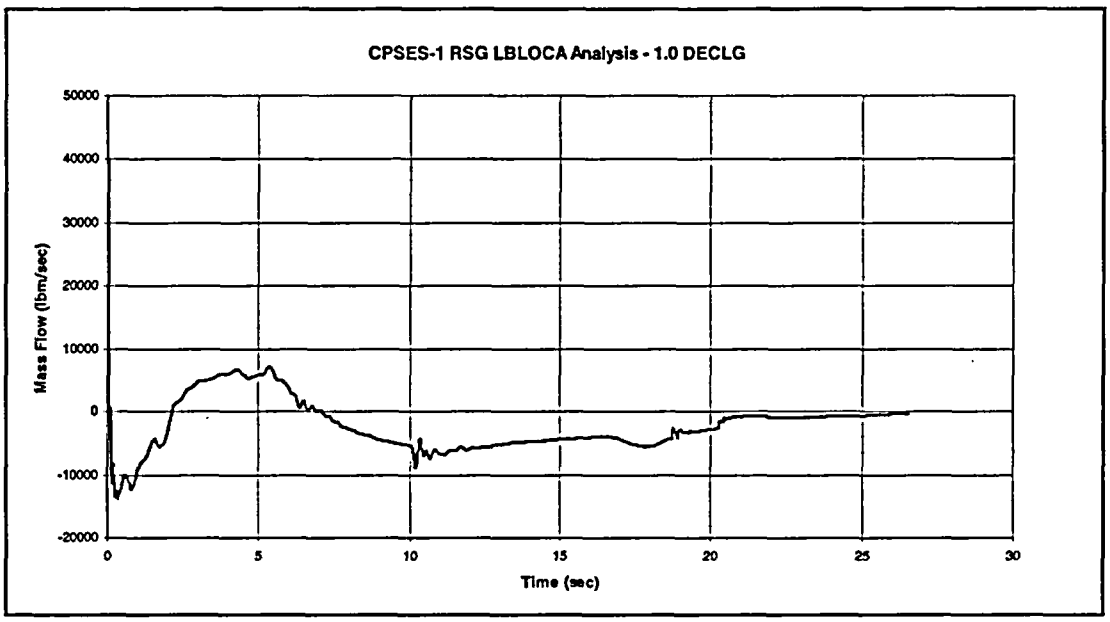

Figure 6.3 - Downcomer Flow Rate

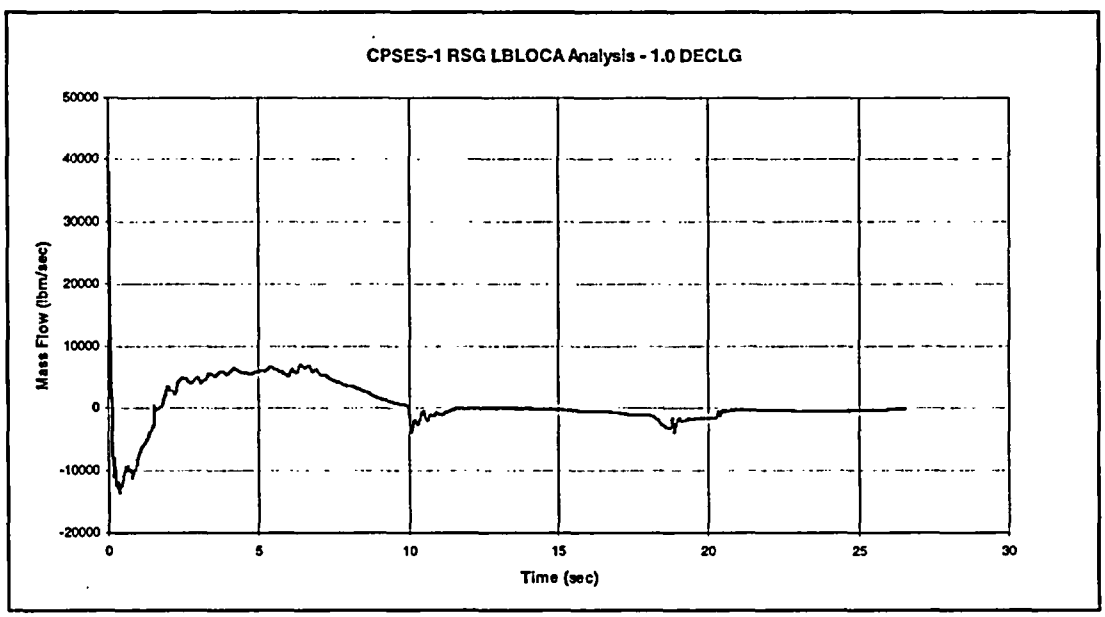

Figure 6.4 - Average Core Inlet Flow Rate

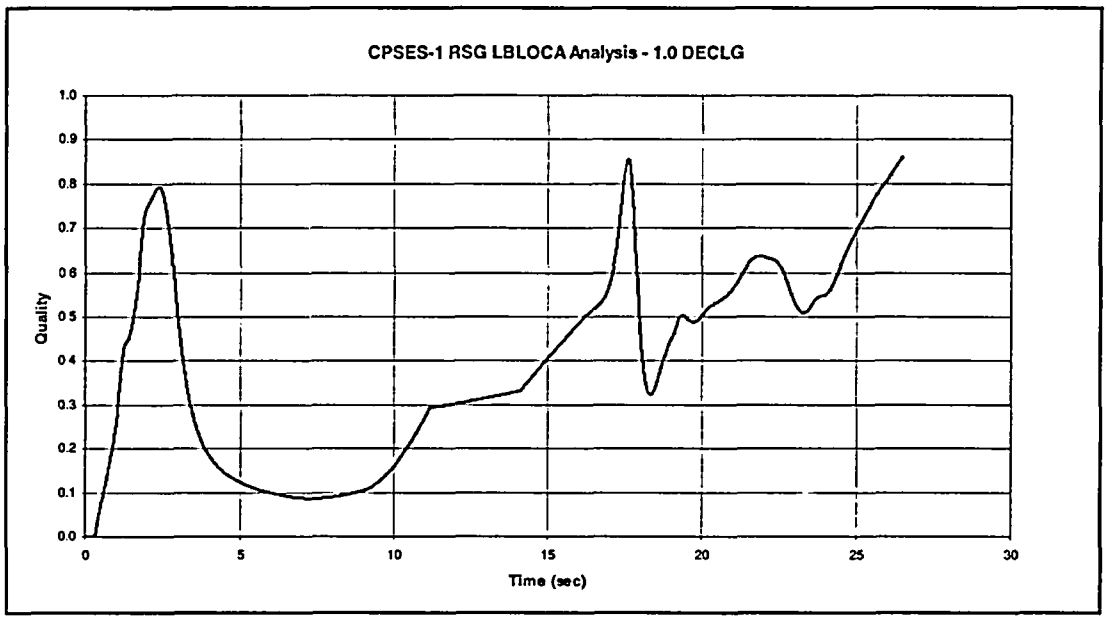

Figure 6.5 - Averagy Core Mid Plane Quality

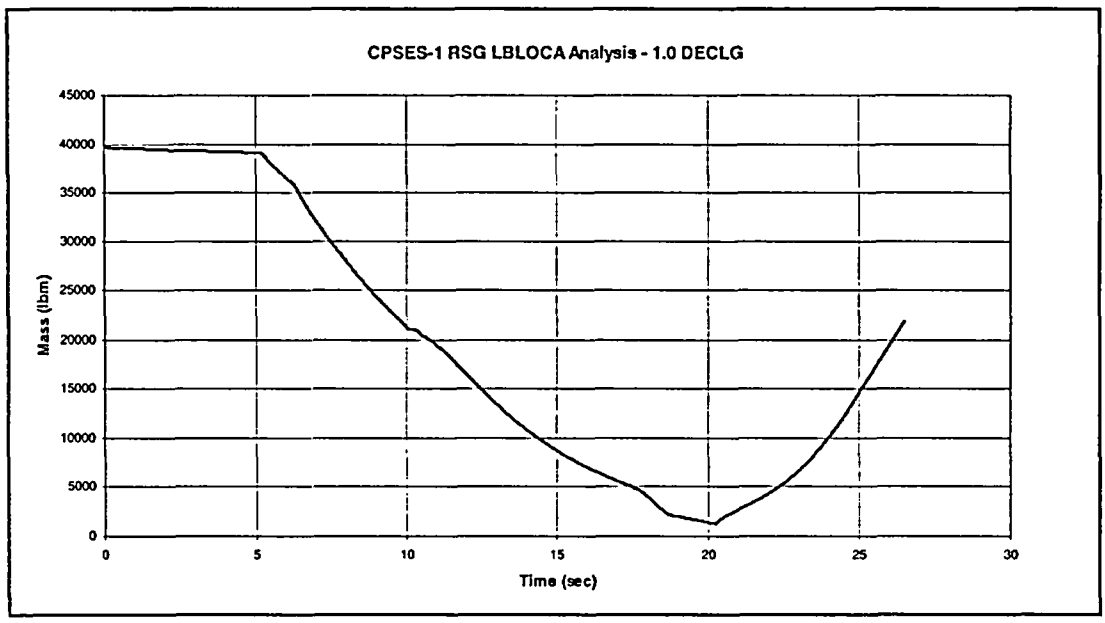

Figure 6.6 - Downcomer Mass Liquid Inventory

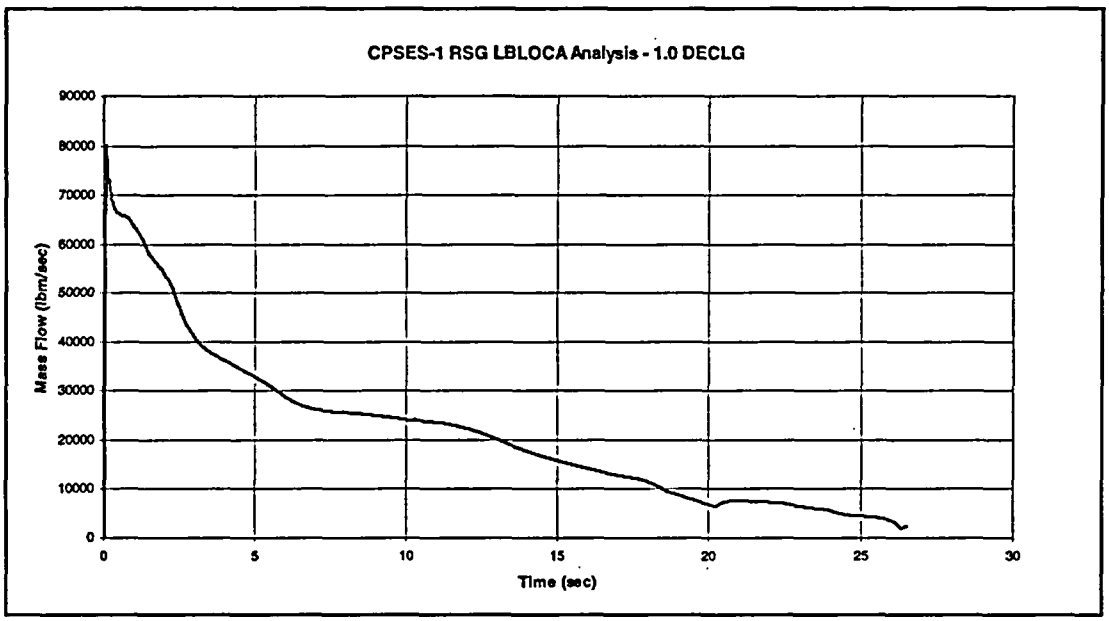

Figure 6.7 - Total Break Flow Rate

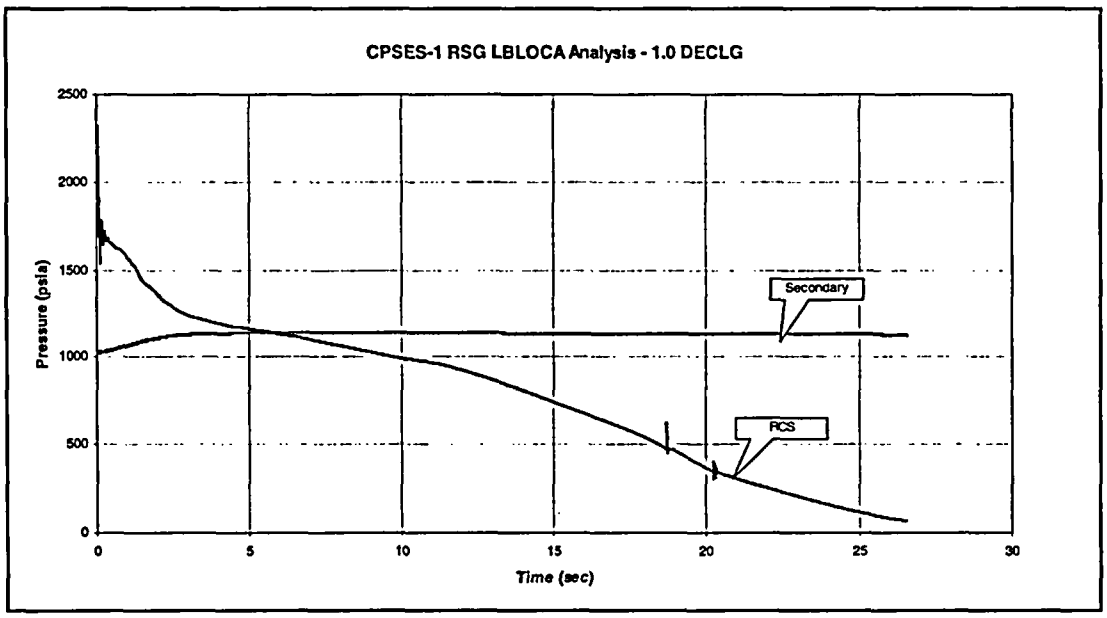

Figure 6.8 - RCS and Secondary Pressures

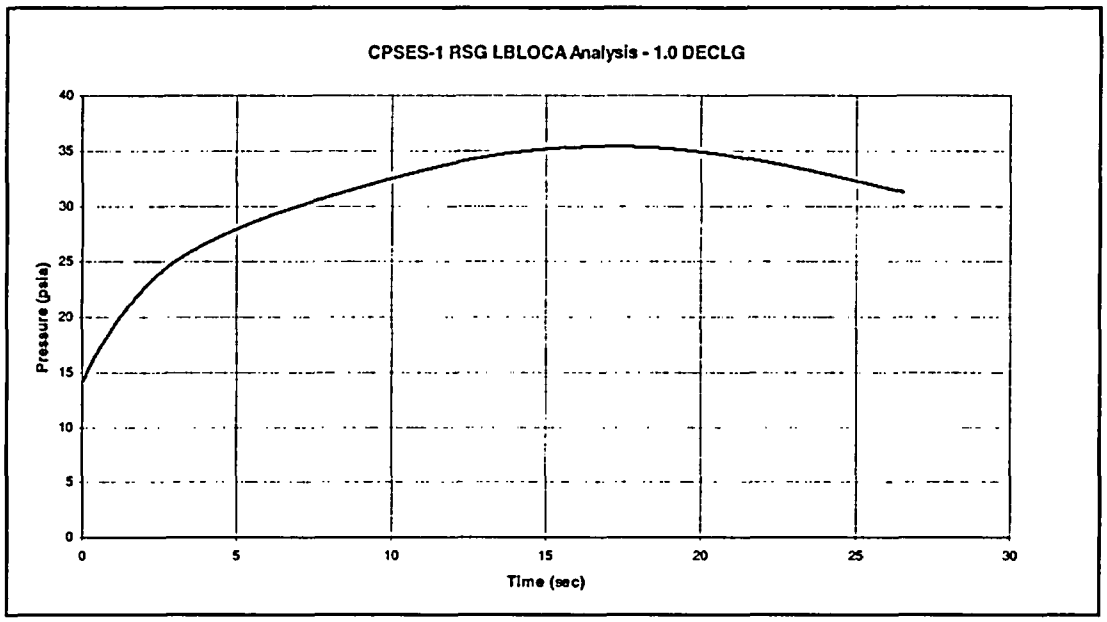

Figure 6.9 - Containment Pressure

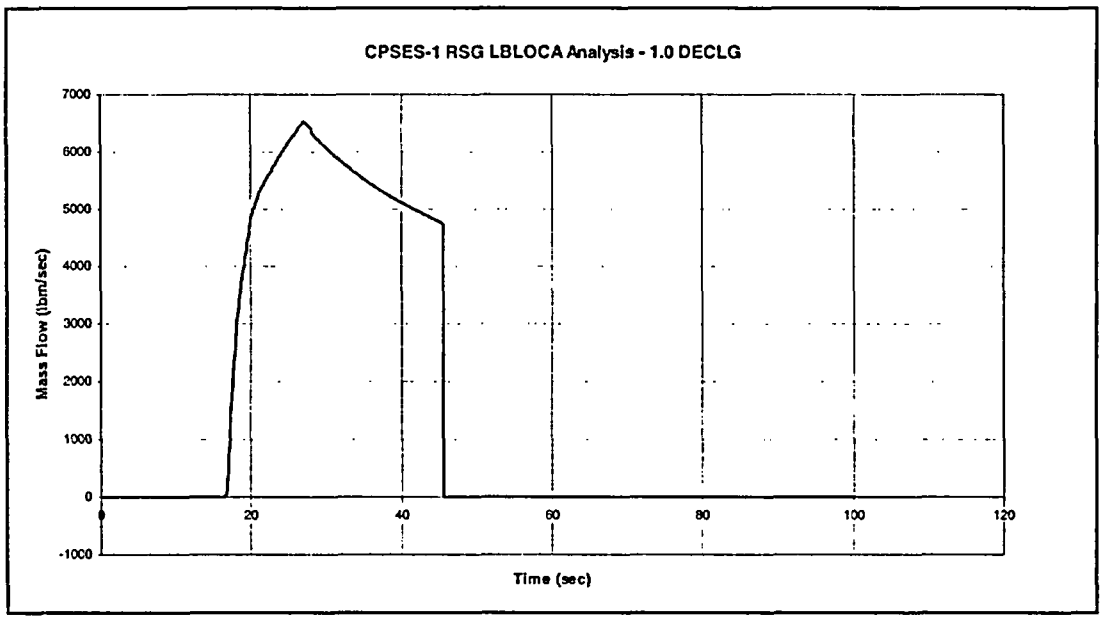

Figure 6.10 - Accumulator Flow Rate

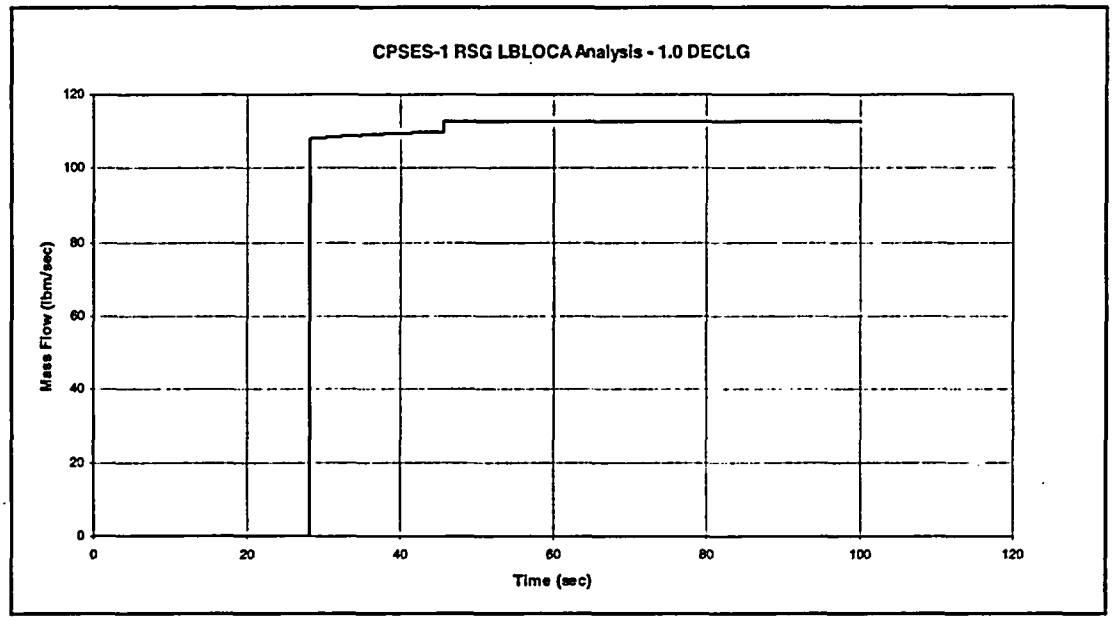

Figure 6.11 - CCP and HHSI Flow Rates

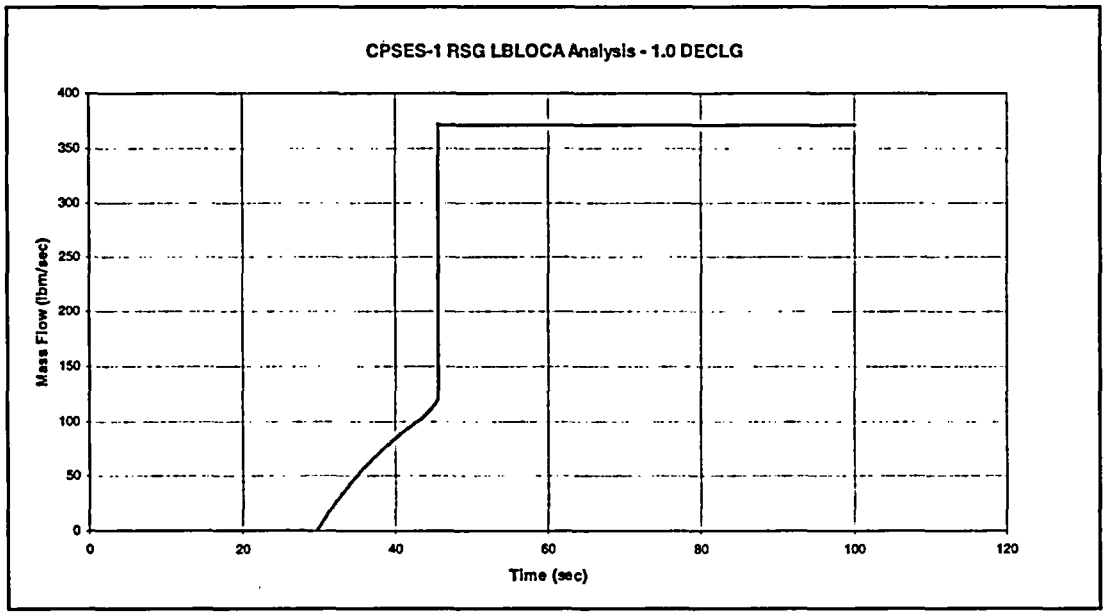

Figure 6.12 - RHR Flow Rate

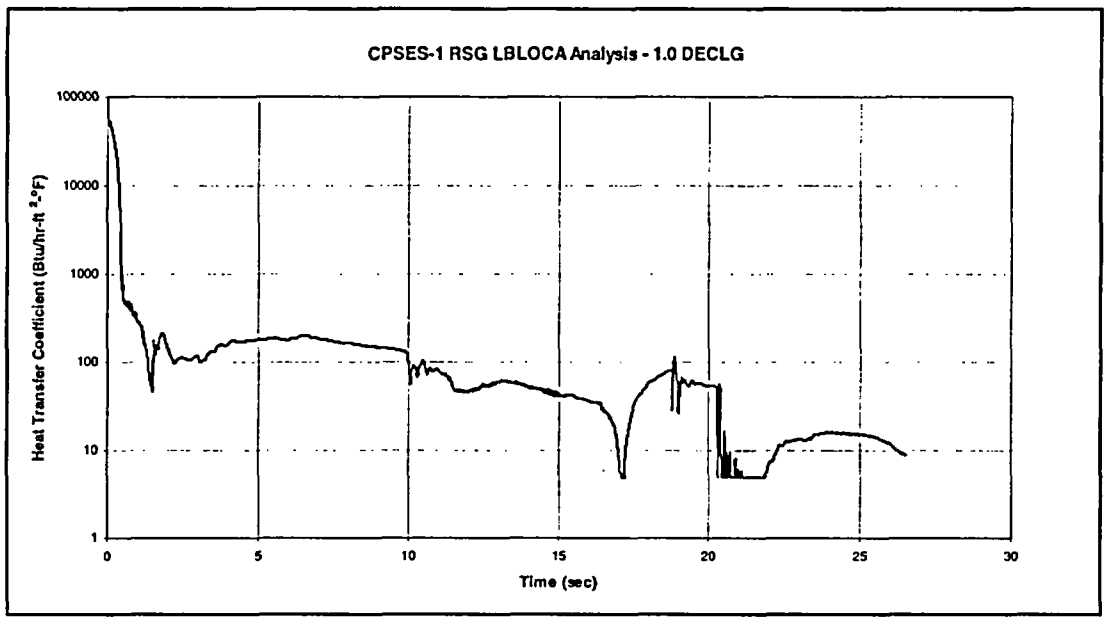

Figure 6.13 - Hot Assembly Peak Power Node Heat Transfer Coefficient

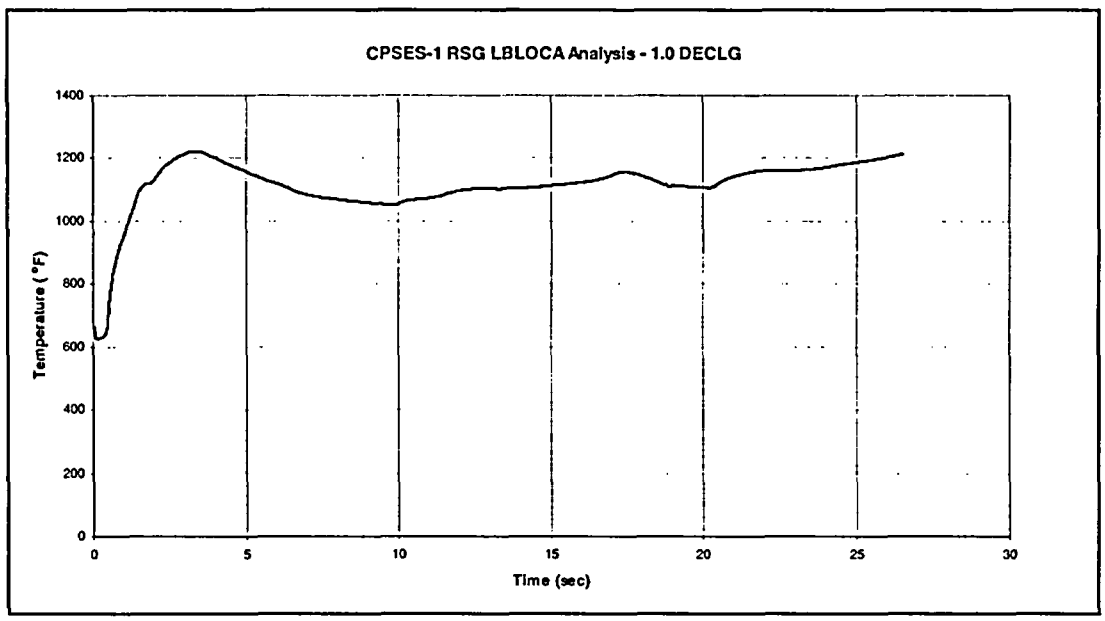

Figure 6.14 - Hot Rod Temperature at PCT Node Elevation

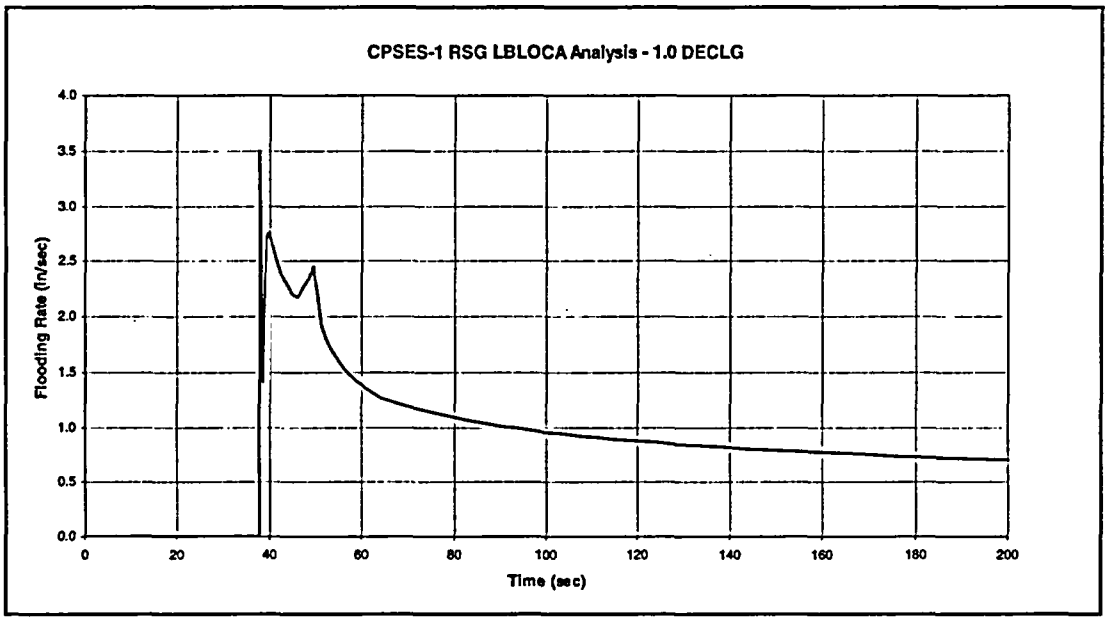

Figure 6.15 - Core Flooding Rate

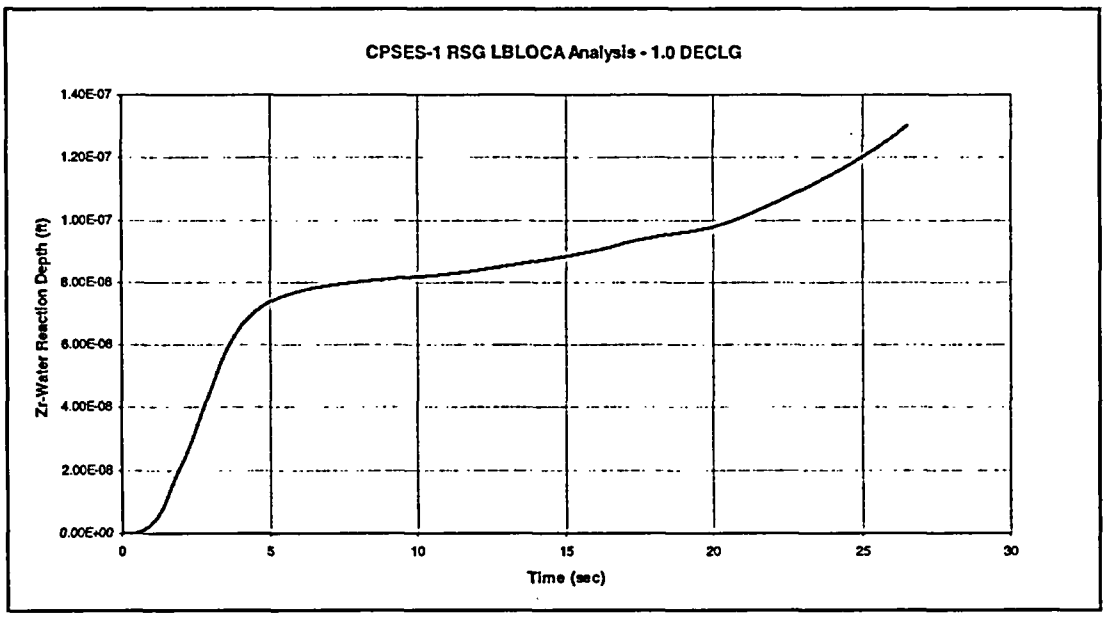

Figure 6.16 Hot Assembly Peak Power Node Zr/Water Reaction Depth

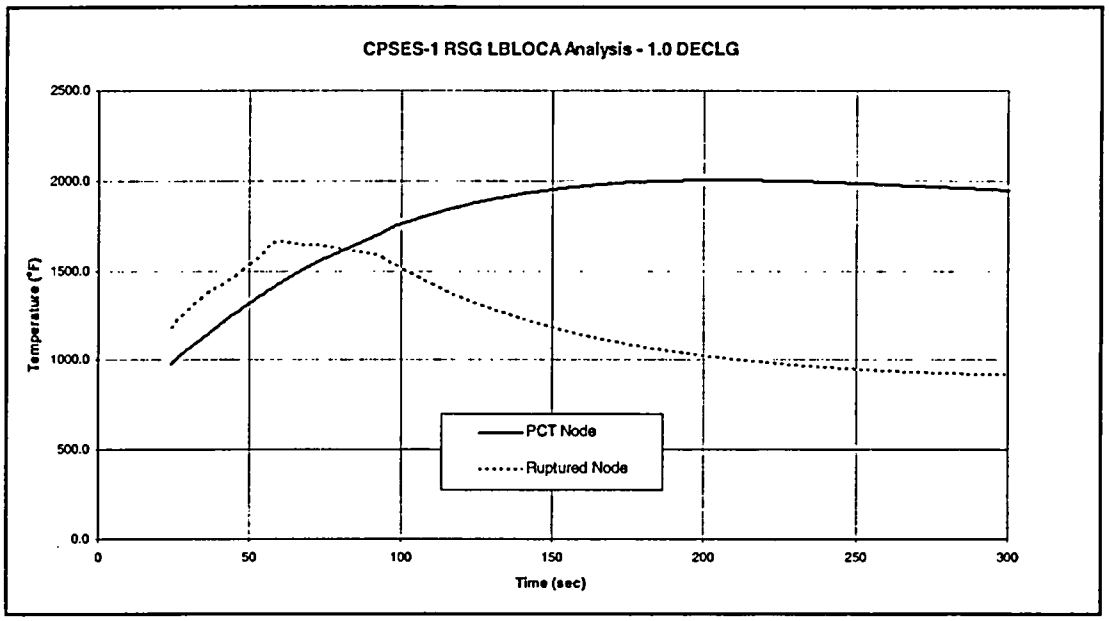

Figure 6.17 - PCT/Ruptured Node Cladding Temperature (1.0 DECLG)

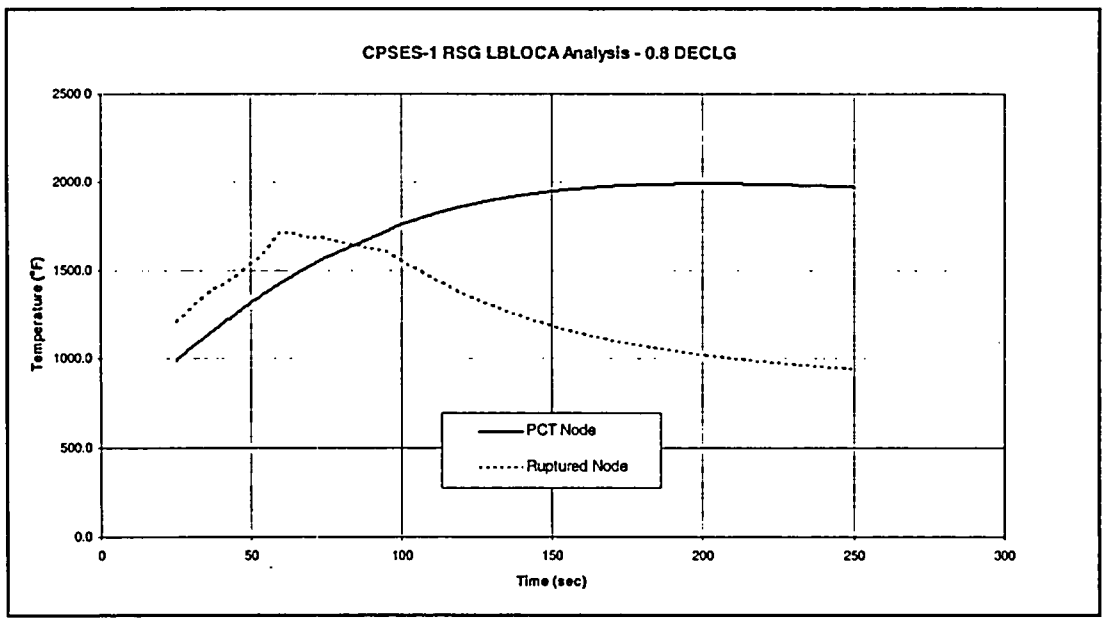

Figure 6.18 PCT/Ruptured Node Cladding Temperature (0.8 DECLG)

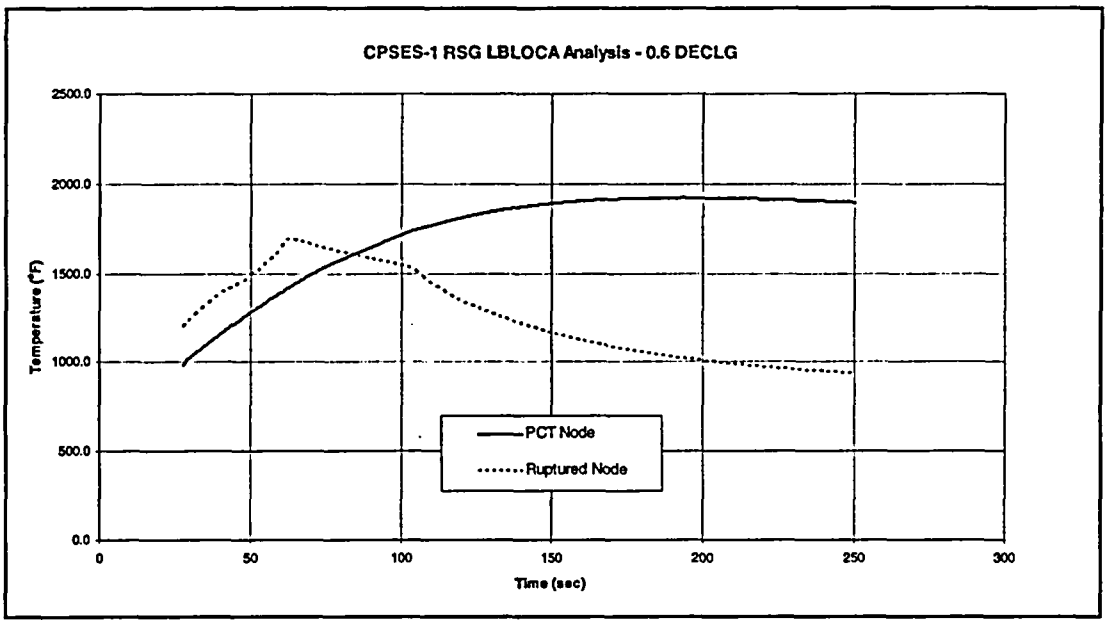

Figure 6.19 - PCT/Ruptured Node Cladding Temperature (0.6 DECLG)

CHAPTER 7

CONCLUSION

The objective of this report is to obtain NRC approval for changes to TXU Power's large and small Break LOCA ECCS Evaluation Models so they may be used to analyze CPSES-1 with the replacement Δ-76steam generators. The current NRC-approved large (Reference 1) and small (Reference 5) break LOCA Evaluation Models will continue to be used for CPSES-2, which has D-5 steam generators. Therefore these changes, once approved, will supplement rather than replace the current methodologies.

The relevant differences between the Δ-76 and the D-4 and D-5 steam generators modeled in the currently NRC-approved large and small Break LOCA ECCS Evaluation Models were summarized in Chapter 2. Briefly: the Δ-76 has a larger primary side volume, a larger secondary side inventory, a larger heat transfer area, a lower tube thermal conductivity, which does not offset the previous features, no pre-heater and the auxiliary feedwater does not share delivery piping with the main feedwater.

The SBLOCA model changes needed to model the Δ-76 RSGs were described in Chapter 3. The only code affected was ANF-RELAP. The proposed ANF-RELAP model is essentially the same as the approved model of Reference 5, with the exception of the new steam generator geometry, minor nodalization adjustments and revised downcomer to vessel head orifice dimensions.

Chapter 4 presented five SBLOCA demonstration analyses. It also presented, for comparison purposes, the CPSES-1 Cycle 11 3 inch break and a 4 inch break analysis with the D-4 steam generator model. The demonstration analyses were discussed in depth and the accident progression was similar to that presented for the D-4. Where there were differences, these were quantitative not

qualitative and consistent with the physical differences between the steam generators. Tables 4.6 through 4.9 summarize the key results of the analyses.

The LBLOCA model changes needed to model the Δ-76 RSGs were described in Chapter 5. Here two codes were affected: RELAP4 and REFLEX, but the only changes involved the Δ-76 steam generator geometry replacing that of the D-4.

Chapter 6 presented nine LBLOCA demonstration analyses, not including the CPSES-1 Cycle 11 DEG break analysis of record with the D-4 steam generator model, presented for comparison purposes. The demonstration analyses were also discussed in depth and the accident progression was similar to what was seen for the D-4. Where there were differences, these were quantitative not qualitative and consistent with the physical differences between the steam generators. Tables 6.5 through 6.8 summarize the key results of the analyses.

In each of the cases presented in this report, the calculated results show the following:

1. The calculated peak clad temperature is lower than the 2200^{0}F peak clad temperature limit set forth in 10 CFR 50.46 (b)(1).

2. The total cladding oxidation at the peak location[16] is under the 17% limit specified in 10 CFR 50.46 (b)(2).

3. The hydrogen generated in the core by cladding oxidation is less than the 1% limit of 10 CFR 50.46 (b)(3).

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[16]This includes the initial, pre-transient oxidation calculated with RODEX2.

4. Finally, the limiting SBLOCA case shows that the baseline break flow is matched by the pumped injection flow, indicating that stable recovery is underway. Stable recovery for the LBLOCA limiting case is similarly indicated.

These analyses demonstrate the proper implementation of the changes described in Chapters 3 and 5 and the overall conclusion from these analyses is that the changes to the methodologies are appropriate.

TXU Power will therefore incorporate these changes into its large and small break LOCA methodologies for use on CPSES-1 with the Δ-76 RSGs. These changes include all codes, input decks, results, conclusions, and application procedures presented in this report to perform large and small break LOCA analyses and evaluations in compliance with 10 CFR 50.46 criteria and Appendix K requirements, for CPSES-1 with the Δ-76 RSGs. The current NRC-approved large (Reference 1) and small (Reference 5) break LOCA Evaluation Models will continue to be used for CPSES-2, which has D-5 steam generators. Therefore these changes, once approved, will supplement rather than replace the current methodologies. Finally, all methodologies will remain supplemented by Reference 3.

CHAPTER 8

REFERENCES

1. TXU Electric, "Revised Large Break Loss of Coolant Accident Analysis Methodology," ERX-2000-002-P-A, March 2000.

2. Siemens Power Corporation, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," EMF-2087 (P), Revision 0 , August 1998.

3. TXU Electric, "ZIRLO™ Cladding and Boron Coating Models for TXU Electric's LOCA Methodologies," ERX-2001-005-P-A, September 2002.

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6. Siemens Power Corporation, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model Applications," XM-NF-82-49 (P) (A), Revision 1 , Supplement 1 , May 1992.

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