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STOLZ, J. F. PWR Project Directorate 6

SUBJECT: Forwards "Small Break LOCA Analysis for B&W 177-FA Lowered-Loop Plants in Response to NUREG-0737, Item II. K. 3. 31." Evaluation concludes that small break LOCA spectrum demonstates conformance w/acceptance criteria of 10CFR50. 46.

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October 3, 1986

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. John F. Stolz, Project Director
PWR Project Directorate No. 6

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

By letter dated September 6, 1985, Duke Power Company (Duke) provided a confirmatory response to issues relating to NUREG-0737 Item II.K.3.30 and to the use of CRAFT2 code. In responding to NUREG-0737 Item II.K.3.31 requirements, Duke committed to the use of the approved Small Break LOCA (SBLOCA) Evaluation model to perform analyses for Oconee to affirm the validity of the previous SBLOCA analyses. My letter of June 27, 1986 had advised you of a delay in the submittal of results of the analyses.

Please find attached a copy of a report entitled, "Small Break Loss of Coolant Accident Analysis for B&W 177FA Lowered Loop Plants in Response to NUREG-0737, Item II.K.3.31", September 1986. This report presents SBLOCA analysis for the 177FA lowered loop plant type performed with the revised evaluation model created in response to NUREG-0737, Item II.K.3.30. This report describes SBLOCA transient behavior, compares revised evaluation model results with previous SBLOCA spectrum analyses. The evaluation concludes that the SBLOCA spectrum, inclusive of previous and revised evaluation model results, satisfies the requirements of NUREG-0737, Item II.K.3.31, demonstrating conformance with the acceptance criteria of 10 CFR 50.46.

Very truly yours,

H.B. Tucker /BW

Hal B. Tucker

PFG/28/slb

Attachment



A046
1/1

xc: Mrs. Helen Pastis
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Oconee Nuclear Station

See 1st Page of Ench

BAW-1976

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Duke tells me that there is a letter into the NRC clearing this. letter dated 1/9/87 from Chuck H-Tuck to Walt Paulson

Thanks

Jim McKnight

BAW-1976
September 1986
77-1165715-00

# SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR B&W 177-FA LOWERED-LOOP PLANTS IN RESPONSE TO NUREG-0737, ITEM II.K.3.31

Prepared For

The B&W Owners Group

Arkansas Power & Light Company
Duke Power Company
Florida Power Corporation
GPU Nuclear Corporation
Sacramento Municipal Utility District



THE BABCOCK & WILCOX COMPANY
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Report BAW-1976

September 1986

Small Break Loss-of-Coolant Accident Analysis For Babcock & Wilcox 177 Fuel Assembly (FA) Lowered Loop Plants in Response to NUREG-0737, Item II.K.3.31

G. E. Anderson, J. R. Paljug

Key Words: NUREG-0737, II K.3.31, SBLOCA

## ABSTRACT

Small break loss-of-coolant accident (SBLOCA) analyses have been performed utilizing the revised evaluation model (EM) created in response to NUREG-0737, Item II.K.3.30. The evaluation presented herein is in specific response to Item II.K.3.31 of NUREG-0737. This report describes SBLOCA transient behavior, compares revised EM results with previous results, and determines the applicable conservatism of previous SBLOCA spectrum analyses. The evaluation concludes that the SBLOCA spectrum inclusive of previous and revised EM results satisfies the requirements of NUREG-0737, Item II.K.3.31, and therefore confirms that the Babcock & Wilcox 177-fuel assembly (FA) lowered loop plants conform to the acceptance criteria of 10 CFR 50.46.

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# 1. INTRODUCTION

In response to the requirements of NUREG-0737, Section II.K.3.31, The Babcock & Wilcox Owners Group (B&WOG) has performed design-basis small break loss-of-coolant accident (SBLOCA) analyses for the lowered-loop plant design. These analyses repeat certain studies that have been previously submitted; these latest studies, however, employ updated modeling techniques, inputs and assumptions which are discussed specifically or by reference in this document. The results of these analyses, presented and described herein, confirm the findings of previous studies: the Babcock & Wilcox (B&W) designed lowered-loop 177-fuel assembly (FA) plants can be maintained within the limits of 10 CFR 50.46 should a SBLOCA occur.

## 2.0 BACKGROUND

As a result of the March 28, 1979 accident at Three Mile Island Unit 2 (TMI-2), the Bulletins and Orders Task Force was formed within the Nuclear Regulatory Commission (NRC) office of Nuclear Reactor Regulation. The Task Force was charged, in part, with reviewing the analytical predictions of feedwater transients and SBLOCAs to ensure the continued safety of all operating reactors, and with determining the acceptability of operator emergency guidelines. As a result of their reviews, the Task Force concluded that, while there were no apparent safety concerns, additional system verification of the SBLOCA model (as required by II.4 of Appendix K to 10 CFR 50) was needed in certain areas. These improvements and concerns, as they applied to each light water reactor (LWR) vendor's model, were documented in the various Task Force reports for each vendor. The review of the B&W SBLOCA model was documented in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," January 1980. The review of the reactor coolant pump model was documented in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979. On October 31, 1980, the NRC issued NUREG-0737, "Clarification of TMI Action Plan Requirements." Included in NUREG-0737 is the requirement for an industry review of NUREG-0565 and -0623 and the development of a program that addresses the NRC concerns therein. The Small Break LOCA Methods Program was developed by the B&W Owners Group to address the requirements of NUREG-0737, Section II.K.3.30.

The results of the Small Break LOCA Methods Program have been documented in References 1 and 2. These references address the revision to SBLOCA codes and models in response to the issues identified in NUREG 0565 and -0737. The NRC has reviewed and approved the results of the SBLOCA Methods Program with the issuance of the May 5, 1985 Safety Evaluation Report (SER) for the Babcock &

Wilcox Owners Group Small Break Loss-of-Coolant Accident Evaluation Model. This SER is presented on pages v. - liii of Reference 1. The preceding documentation completed the requirements of NUREG-0737, Section II.K.3.30. However, it then became necessary to address the requirements of NUREG-0737, Section II.K.3.31. As is discussed in the following sections, a program for compliance with II.K.3.31 was formulated and carried out by the B&W Owners Group.

## 3. SPECIFIC REQUIREMENTS OF NUREG-0737, SECTION II.K.3.31

Section II.K.3.31, entitled "Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46," requires that plant-specific calculations using NRC-approved models for SBLOCAs be performed to show compliance with 10 CFR 50.46 and be submitted for NRC approval. The requirements are applicable to all operating reactors and applicants for operating license.

Additional information concerning the requirements of Section II.K.3.31 was provided by the NRC letter entitled "Clarification of TMI Action Plan Item II.K.3.31 (Generic Letter No. 83-85)," D. G. Eisenhut, November 2, 1983. This letter states "...the requirements of II.K.3.31 can be satisfied by each licensee by submittal of a plant-specific analysis that demonstrates that current SBLOCA analyses using previously approved evaluation models are more limiting than analyses using the revised (II.K.3.30) models. This bounding demonstration can be done on a generic basis through the owners groups or vendors and submitted individually by each licensee."

Furthermore, the aforementioned SER (Reference 1) states, "It is the intention of the B&W Owners Group (B&WOG) to provide generic analyses, by plant configuration, in response to NUREG-0737 II.K.3.31 which will demonstrate that the current FSAR SBLOCA results are conservative. This will be accomplished by selecting a limited break spectrum for the evaluation. The break spectrum will be selected to exercise the ECC system and span the previously identified limiting break size."

In view of the requirements and clarifying documentation, the intent of the B&WOG through the analyses presented in this report is to demonstrate compliance with II.K.3.31 by showing conformance with 10 CFR 50.46. Conformance will be demonstrated through a qualitative assessment of the SBLOCA spectrum to determine the critical break sizes and then a quantitive analytical evaluation of these critical break sizes. For the purposes of this discussion, a critical break size is one that produces a cladding temperature in

excess of primary coolant saturation temperatures or provides evaluation of phenomena indicative of the transition from one break size category to another. Discussions of break size categories and of the selection of break sizes to be analyzed with the revised EM are contained in the following sections of this report.

# 4.0 SMALL BREAK LOSS-OF-COOLANT ACCIDENTS

Based on results of design-basis analyses, a loss-of-coolant accident is defined to be "small" when its cross-sectional area is 0.5 $ft^2$ or less. An SBLOCA involves a relatively slow system depressurization. Flow conditions within the reactor coolant system change gradually and smoothly. Temperature and pressure gradients between regions tend to be small. The lack of agitation allows partial phase separation of steam and water and, in some situations, countercurrent flow. Rather than the distinct blowdown and reflood phases associated with large breaks, small breaks have a smooth transition from a period of relatively high core flow to one of relatively quiescent conditions. During the early phase, heat transfer in the core is flow-controlled and is adequate to cool the fuel cladding. Later, during the quiescent period, a two-phase froth level can develop in the core region of the reactor vessel. To ensure adequate core cooling, a two-phase level must be maintained at or near the top of the core as a minimum. In this manner, the generated core decay heat can be removed from the fuel rods by pool boiling or, if the core is slightly uncovered, by convection to superheated steam. The emergency core cooling system (ECCS) has been designed to provide the necessary fluid makeup to the reactor coolant system (RCS) to ensure at least this adequate level of core decay heat removal.

Design basis SBLOCA analyses have shown that different sizes of SBLOCAs exhibit various characteristic RCS responses. For very small breaks (less than approximately 0.005 $ft^2$), natural circulation will be maintained and the primary system can be kept in, or returned to, a subcooled state. For larger small breaks (approximately 0.02 $ft^2$ and larger), the circulation flow phase will end soon after the RC pumps are tripped.

After the "forced flow" circulation portion of a small break, the reactor coolant "settles out." That is, the water falls by gravity and collects in the lower regions, and the steam separates from the liquid phase and collects

in the high points of the system. A "boiling pot" water level will exist that will vary depending on break size and location, primary-to-secondary heat transfer, ECCS performance (which depends largely on RCS pressure) and decay heat levels.

These variables can cause the response characteristics of the RCS to change in different ways after the reactor coolant system "settles" into the "boiling pot." However, the following four main categories of SBLOCAs have been designated:

1. SBLOCAs too small to interrupt natural circulation.
2. SBLOCAs that may allow the RCS to repressurize in a saturated condition.
3. SBLOCAs that allow RCS pressure to stabilize at approximately secondary side pressure.
4. SBLOCAs large enough to depressurize the RCS sufficiently to permit low pressure injection (LPI).

## 4.1. Category 1: SBLOCAs Too Small to Interrupt Natural Circulation

Typically for Category 1 breaks (sizes smaller than approximately 0.005 $ft^2$), the high-pressure injection (HPI) system can compensate for the break flow and maintain the primary coolant loops essentially full of liquid so that circulation is not interrupted.

An example of the expected RCS pressure response for breaks in this category is shown as Curve 1 of Figure 4-1. While RCS pressures may tend to remain high, the operator has equipment available (RC pumps, pressurizer sprays, and SGs) to affect a near-normal cooldown and depressurization of the RCS.

## 4.2. Category 2: SBLOCAs That May Allow the RCS to Repressurize in a Saturated Condition

Category 2 breaks (sizes between approximately 0.005 $ft^2$ and 0.02 $ft^2$) initially cause the RCS to depressurize and become saturated. Later, however, these breaks can cause the upper hot leg elbows to void sufficiently to interrupt circulation. This loss of circulation leads to a loss of primary-to-secondary steam generator (SG) heat transfer. Without SG heat transfer,

the mass and energy addition from the core and ECCS may exceed the ability of these break sizes to discharge mass and energy; as a result, the RCS may repressurize.

As the RCS continues to lose inventory, a condensing surface will be exposed inside the SG tubes. This will establish the boiler-condenser (b-c) mode of heat removal. This heat removal mode will terminate any pressure increase and will facilitate controlling RCS pressure at a value sufficient to ensure adequate HPI flow for core cooling.

The transient pressure behavior for a Category 2 break is illustrated as Curve 2 of Figure 4-1. This transient, for a 0.01 $ft^2$ break at RC pump discharge, is taken from Reference 3 and includes saturated repressurization followed by depressurization as the RCS evolves into the b-c mode of cooling.

### 4.3. Category 3: SBLOCAs that Allow RCS Pressure to Stabilize at Approximately SG Secondary Pressure

Category 3 breaks (sizes between approximately 0.02 $ft^2$ and 0.10 $ft^2$) result in RCS depressurization until the primary and secondary systems reach thermal equilibrium. At this point, SG heat removal is essentially lost. A continuous RCS depressurization is maintained. The rate of this depressurization can, however, be relatively slow and dependent on the rates of

1. Decrease in core decay heat generation.
2. Cooldown of fluid and metal components in the primary and secondary systems.
3. Release of mass and energy out of the break.
4. ECCS injection rate.

An example of the possible RCS pressure response for breaks in this category is shown as Curve 3 of Figure 4-1.

Partial core uncovering (Figure 4-2), accompanied by increases in cladding temperature (Figure 4-3) may occur for some Category 3 breaks. Previous studies (Reference 4) show that if core flood tank (CFT) injection begins while the core is still covered, then core uncovering will be prevented.

## 4.4. Category 4: SBLOCAs Large Enough to Depressurize the RCS Sufficiently to Permit Low-Pressure Injection

Category 4 breaks (sizes between 0.1 $ft^2$ and 0.5 $ft^2$) will cause the RCS to continually depressurize, first to saturation, then to an equilibrium condition with the secondary system. At that point, SG heat transfer ceases. However, the break is large enough to allow the RCS depressurization to continue without SG heat transfer. As RCS pressure falls below secondary pressure, heat energy flows in the reverse direction, from secondary to primary. This relatively insignificant amount of added energy would tend to prevent further RCS depressurization; however, Category 4 breaks are large enough to allow continued RCS depressurization, despite the slight addition of secondary heat.

Of more significance than the amount of energy added to the RCS by reverse heat transfer is the overall impact of SG modeling on transient results. The type of SG model used (previous versus revised) has essentially no impact on the remainder of an SBLOCA transient once sustained reverse heat transfer is achieved. The major dependency of this category of SBLOCAs is on break flow. Typical RCS pressure behavior for a Category 4 break is illustrated as Curve 4 of Figure 4-1. For this category of breaks, the RCS pressure continues to decrease due to the relatively large break area until it falls below the shutoff head of the LPI system, ensuring adequate core cooling.

Figure 4-1 - Characteristic RCS Pressure Response
for SBLOCA Categories 1-4

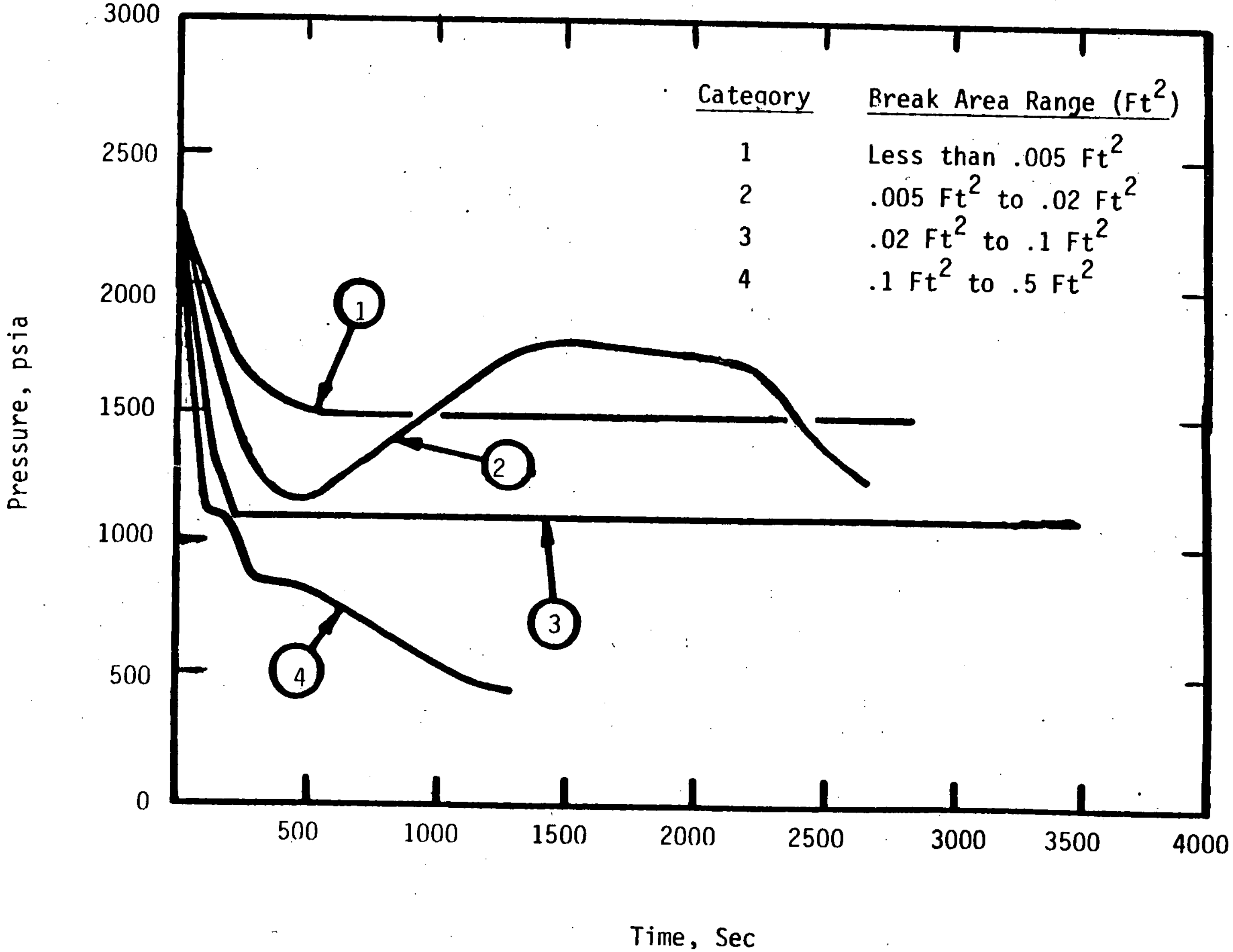


Figure 4-2 - Transient Core Mixture Height Vs.
SBLOCA Break Size

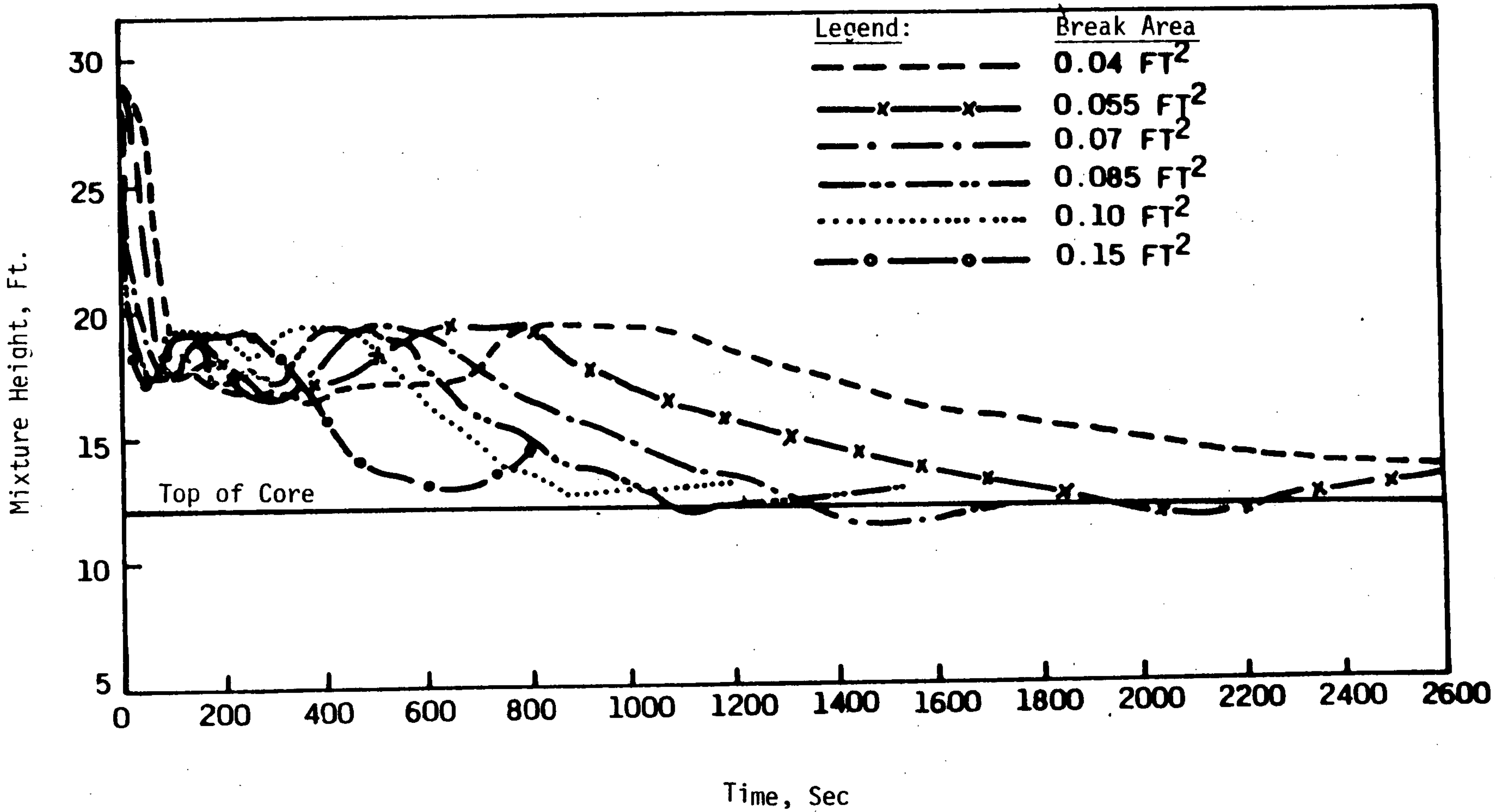
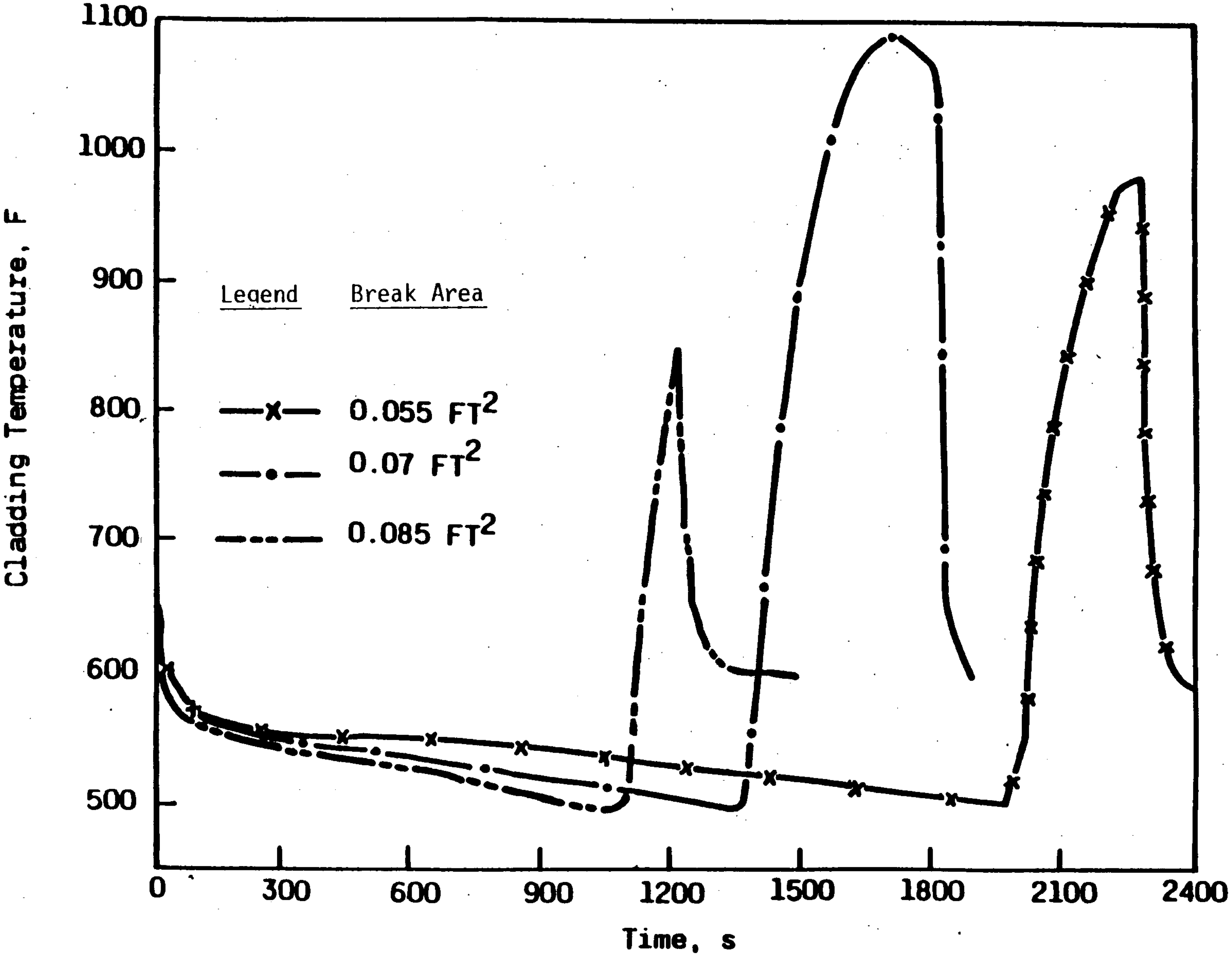


Figure 4-3 - Cladding Temperature Response - SBLOCA's Which Produce Partial Core Uncovering



# 5. SBLOCA EVALUATION MODELS

## 5.1. Previous B&W SBLOCA EM

SBLOCA analyses to date generally have utilized the previously approved SBLOCA EM which is discussed in Reference 4. The noding scheme for this model is shown in Figure 5-1.

Results for the entire SBLOCA spectrum analyzed with this model (References 4 and 5) have demonstrated that only certain sizes of Category 3 breaks result in fuel cladding temperature increases above coolant saturation temperatures. Furthermore, the maximum cladding temperature for SBLOCAs has been determined (Reference 4) to be far below the 2200F limit of the acceptance criteria of 10 CFR 50.46. The minimum core mixture level calculated with the previous EM is 11 ft for the 0.07 $ft^2$ break size (Figure 4-2). The cladding temperature responses for the SBLOCAs, analyzed with the previous EM, that produced the most limiting partial core uncovering are shown in Figure 4-3.

## 5.2. Revised B&W SBLOCA EM

As a result of the SBLOCA Methods Program developed to address the requirements of NUREG-0737, Section II.K.3.30, certain code modifications were made to the existing SBLOCA EM. These modifications are described in detail in Reference 1, and include the following models:

1. A non-equilibrium pressurizer model.
2. A two-phase RC pump model.
3. A mechanistic steam generator model.
4. A revised auxiliary feedwater (AFW) model.

The noding scheme for this revised EM is shown in Figure 5-2.

In addition to the code modifications, the following key input parameters differ as noted between the previous and revised EMs.

1. <u>Steady-state loop flow resistances</u>
Loop flow resistances were increased slightly to enable the revised model to more closely match existing plant data for flow and pressure distributions in steady-state conditions.

2. <u>Pressurizer surge line form loss coefficients</u>
The previous EM accounted for frictional but not form losses in the surge line. For the revised EM, both form and frictional losses were modeled. Best estimate values were input for form loss coefficients that are appropriate for SBLOCA transients in which forced RC flow is of relatively short duration.

3. <u>Low RCS Pressure Engineered Safety Features Actuation System Setpoint:</u>
The low RCS pressure Engineered Safety Features Actuation System (ESFAS) setpoint was changed from 1365 psia in the previous EM to 1495 psia in the revised EM. The change was made after earlier SBLOCA analyses (Reference 1) performed with the revised model indicated an elevated RCS saturation pressure response after the subcooled blowdown phase compared to the pressure response with the previous model. A setpoint os 1495 psia was determined to be necessary to ensure that ESFAS actuates at "early" transient times with the new model as in cases with the old model. Current plant operating setpoints support the use of the higher ESFAS setpoint.

4. <u>AFW Actuation Delay Time</u>
The delay time from reactor and turbine trip (and the assumed coincident loss of offsite power) to AFW actuation was changed from 36.0 seconds to a more conservative value of 40.0 seconds in the revised EM. This change is considered inconsequential to transient results.

5. <u>AFW Flow Rates</u>
The AFW flow rates in the revised EM differ from those in the previous model. The revised flow rates are lower because the new model accounts for AFW recirculation flow and reflects additional line losses. AFW flow rates for the previous and revised EMs are given in Table 5-1.

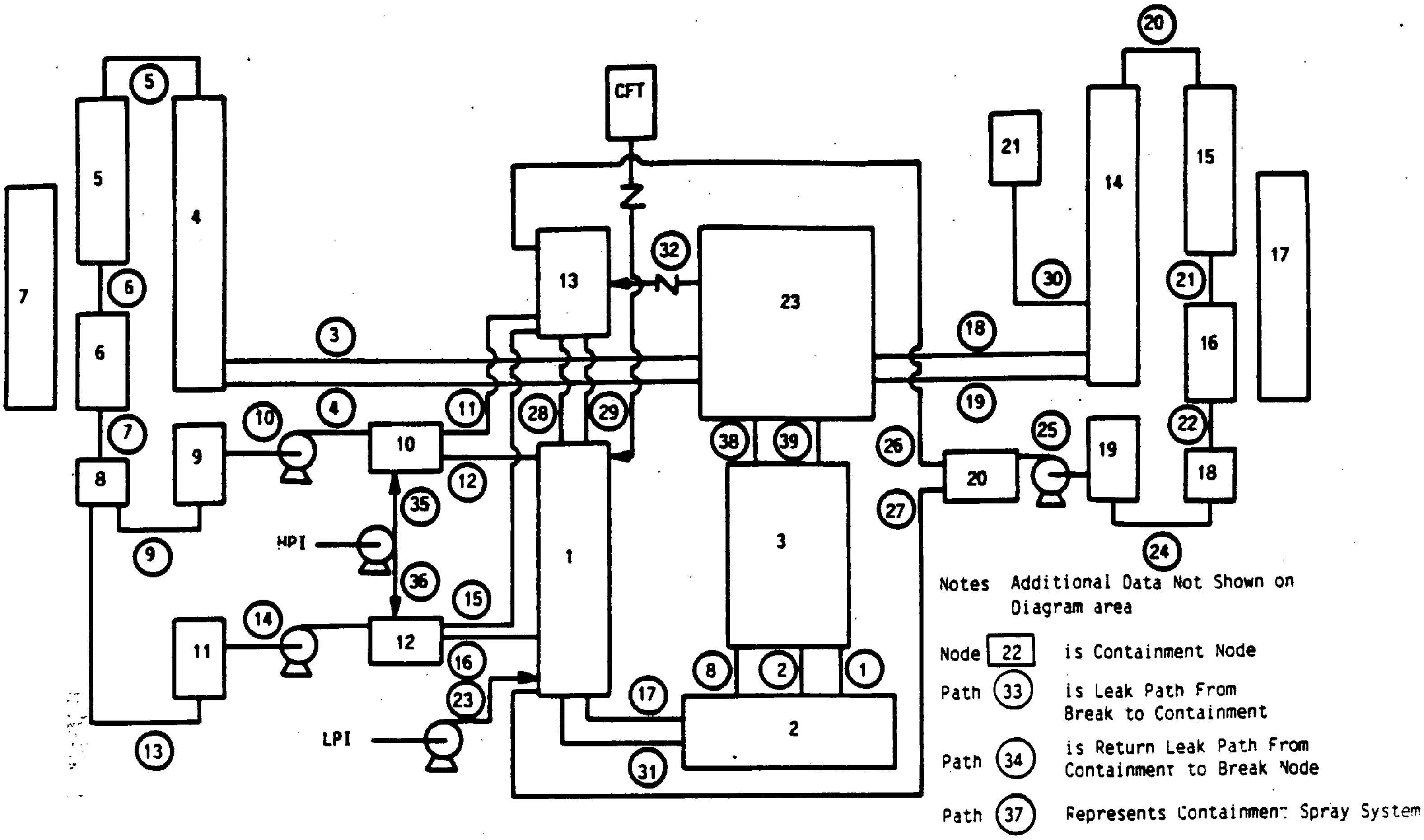
6. AFW Level Setpoint

The secondary level setpoint, at which AFW flow will be automatically stopped, was changed from 17 ft in the previous model to 20.7 ft in the revised model. 20.7 ft corresponds to 50% on the secondary operate range (OR) level scale and more realistically represents the natural circulation level AFW cutoff setpoint utilized in the operating B&W 177-FA lowered loop plants.

Table 5-1. Auxiliary Feedwater Flow Rates For Previous and Revised Evaluation Models

| Secondary Pressure (psia) | Auxiliary Feedwater Flow Rate Per SG (gpm) Previous Model | Revised Model |
|---|---|---|
| 0 | 725 | 570 |
| 929 | 725 | 570 |
| 1136 | 500 | 370 |
| 1244 | 400 | 170 |

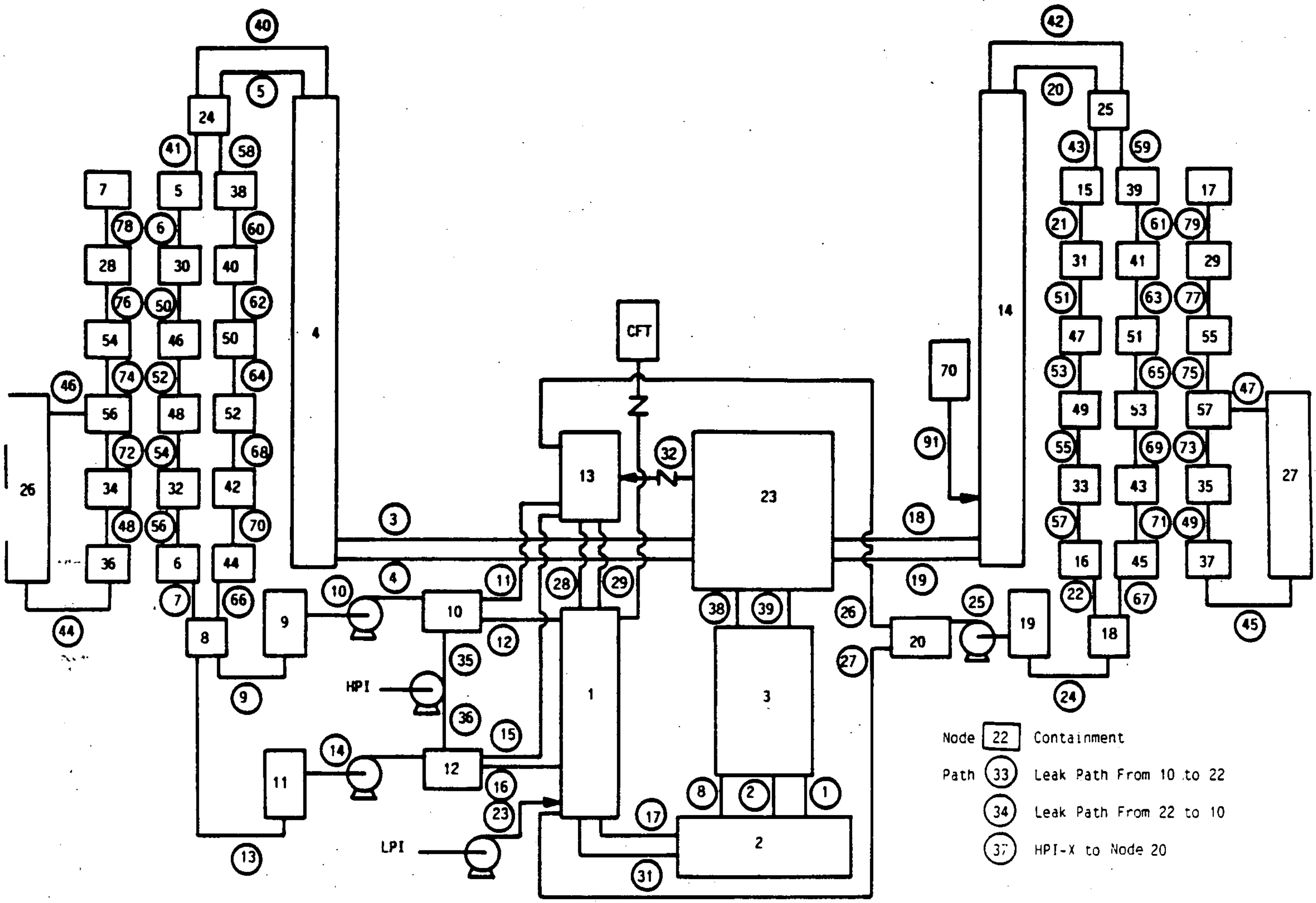
Figure 5-1. CRAFT2 Noding Diagram For Small Breaks - Previous Model




| Node No. | Identification |
|---|---|
| 1 | Downcomer |
| 2 | Lower Plenum |
| 3 | Core |
| 4,14 | Hot Leg Piping |
| 5,15 | SG & Upper Head |
| 6,16 | Steam Generator Tubes |
| 7,17 | Secondary, SG |
| 8,18 | SG Lower Head |
| 9,11,19 | Cold Leg Piping |
| 10,12,20 | Cold Leg Piping |
| 13 | Upper Downcomer |
| 21 | Pressurizer |
| 22 | Containment |
| 23 | Upper Plenum |

| Path No. | Identification |
|---|---|
| 1,2 | Core |
| 3,4,18,19 | Hot Leg Piping |
| 5,20 | Hot Leg, Upper |
| 6,21 | SG Tubes |
| 7,22 | SG Lower Head |
| 8 | Core Bypass |
| 9,13,24 | Cold Leg Piping |
| 10,14,25 | Pumps |
| 11,12,15,16,26,27 | Cold Leg Piping |
| 17,31 | Downcomer |
| 23 | LPI |
| 28,29 | Upper Downcomer |
| 30 | Pressurizer |
| 32 | Vent Valve |
| 33,34 | Leak & Return Path |
| 35,36 | HPI |
| 37 | Containment Sprays |

Figure 5-2 - CRAFT2 Noding Diagram For Small Breaks - Revised Model




| Node No. | Identification |
|---|---|
| 1 | Downcomer |
| 2 | Lower Plenum |
| 3 | Core |
| 4,14 | Hot Leg Piping |
| 5,6,30,32 38,40,42,44, 46,48,50,52 | SG Tubes |
| 7,26,34,36,54,56 | SG-Secondary Side |
| 8,18 | SG Lower Head |
| 9,11,19 | Cold Leg Piping |
| 10,12,20 | Cold Leg Piping |
| 13 | Upper Downcomer |
| 15,16,31,33,39,41 43,45,47,49,51,53 | SG Tubes |
| 17,29,35,37,55,57 | SG-Secondary Side |
| 22 | Containment |
| 23 | Upper Plenum |
| 24,25 | SG Upper Head |
| 26,27 | SG-Downcomer |
| 70 | Pressurizer |

| Path No. | Identification |
|---|---|
| 1,2 | Core |
| 3,4,18,19 | Hot Leg Piping |
| 5,20,40,42 | Hot Leg, Upper |
| 6,41,50,52,54,56,58, 60,62,64,68,70 | SG Tubes |
| 7,66 | SG Lower Head |
| 8 | Core Bypass |
| 9,13,24 | Cold Leg Piping |
| 10,14,25 | Pumps |
| 11,12,15,16,26,27 | Cold Leg Piping |
| 17,31 | Downcomer |
| 21,43,51,53,55,57,59, 61,63,65,69,71 | SG Tubes |
| 22,67 | SG Lower Head |
| 23 | LPI |
| 28,29 | Upper Downcomer |
| 32 | Vent Valve |
| 33,34 | Leak and Return Path |
| 35,36 | HPI |
| 37 | Containment Sprays |
| 38,39,44,46,48 | Upper Plenum |
| 72,74,76,78 | SG-Secondary |
| 45,47,49,73,75,77,79 | SG-Secondary |
| 91 | Pressurizer Surge |

# 6.0 SBLOCA SPECTRUM ANALYSIS WITH REVISED EM

## 6.1. Justification of SBLOCA Cases Selected

In accordance with the May 5, 1985 NRC SER (Reference 1), the requirements of NUREG-0737 II.K.3.31 will be met by

1. Selecting a limited break spectrum for analyses that exercises the ECC and spans the previously identified limiting SBLOCA size, the 0.07 $ft^2$ break.

2. Performing analyses that show that current licensing SBLOCA results presented in References 1, 3, 4 and 5 are conservative.

The limited break spectrum for B&W's 177-FA lowered-loop plants that meets the requirements of item 1 above consists of the 0.01, 0.04 and 0.07 $ft^2$ break sizes, all located in the RC pump discharge piping. This has been previously established as the worst-case break location for B&W plants (Reference 6).

As with the previous analyses, the revised analyses will be conducted assuming the design-basis assumptions required by 10 CFR 50 Appendix K and discussed in Reference 4. These assumptions include

1. 120% of the 1971 ANS 5.1 decay heat generation standard.

2. Failure of one emergency diesel generator to start, leading to the inoperability of one HPI pump and one LPI pump.

3. AFW availability to both SGs until SG secondary levels reach the natural circulation setpoint of 50% OR. At that point AFW will automatically be stopped, and will be reinitiated only if the secondary level again decreases below 50% OR.

The sufficiency of this spectrum will be justified in the following paragraphs. In all cases, demonstration of reasonably close similarities in transient behavior between the previous and revised models will

1. Verify the acceptability and conservatism of the previous EM and of the results obtained using that model.
2. Demonstrate that the revised EM is an acceptable analytical tool for use in future SBLOCA studies.
3. Verify that a complete break spectrum that meets the requirements of 10 CFR 50.46 now exists for B&W's operating plants.

### 6.1.1. 0.01 ft$^2$ Break

The 0.01 ft$^2$ break was chosen because it is representative of Category 2 break sizes. Historically, Category 2 breaks require SG heat removal for RCS depressurization. If SG heat removal is interrupted, Category 2 breaks are small enough to allow saturated RCS repressurization. Therefore, this is the category of most interest when considering the potential effects of the EM revision, particularly those pertaining to SG and AFW modeling. This category is analyzed mainly to demonstrate the repressurization/depressurization phenomena since no core uncovering is predicted for this break size range. Minimum AFW flow rate and SG level requirements deemed essential to help mitigate Category 2 breaks are defined in References 7 and 8.

### 6.1.2. 0.04 ft$^2$ Break

The 0.04 ft$^2$ break is considered to be a transition break size between Categories 2 and 3. Evaluation of such a transition break is necessary to determine how the transient is influenced by changes incorporated in the revised EM. Analyses (Reference 1) with the old EM indicate that this particular size limits the probability of saturated RCS repressurization, a characteristic of Category 2 breaks that indicates primary-to-secondary decoupling, while maximizing the potential for SG influence on a Category 3 break. Therefore, the possibility of significant changes in transient response resulting from the revisions to the EM and the historical interest in breaks in Categories 2 and 3 dictate the selection of the 0.04 ft$^2$ break for reanalysis.

### 6.1.3. 0.07 ft$^2$ Break

The 0.07 ft$^2$ break has been shown by previous design basis analyses (Reference 1) to result in the maximum amount of core uncovering for SBLOCAs. Thus, even

though the 0.07 $ft^2$ break is only an intermediate size (Category 3), it resulted in the most limiting SBLOCA with regard to core cooling and ECCS capacity.

Comparison of the responses of the revised EM relative to the previous model in situations of significant inventory loss and possible core uncovering, as well as loss of and possible reversal of SG heat transfer are desirable. Based on previous analyses, the 0.07 $ft^2$ break transient exhibited all of these characteristics. Therefore, the 0.07 $ft^2$ break size is an appropriate choice for reanalyses to demonstrate the conformance of the revised EM with 10 CFR 50.46 in limiting SBLOCA situations.

## 6.2. Justification for Excluding Category 1 and 4 Breaks From Reanalysis Spectrum

### 6.2.1. Category 1

Analyses of Category 1 breaks with the previous EM (Reference 3) have demonstrated the capability of the HPI system under design-basis conditions to maintain sufficient RCS liquid inventory to sustain natural circulation and prevent core uncovering. Revisions made to the EM are not expected to result in significant changes in transient behavior for breaks in this category. At most, a slight shift may occur in the maximum break size defined for Category 1. This shift would only involve movement between Categories 1 and 2. A representative Category 2 break has been reanalyzed. In comparison with other break categories, Category 1 breaks have been shown to be insignificant in terms of transient severity and the potential for core uncovering. The reanalysis of a Category 1 break is therefore not required to ensure that the break spectrum is complete.

### 6.2.2. Category 4

Category 4 breaks exhibit a fairly rapid and constant depressurization to pressures below the low pressure injection (LPI) shutoff pressure, thus assuring long-term core protection. Saturated RCS pressure rapidly decreases below secondary pressure. When this occurs, reverse heat transfer begins; revisions to the SG model will therefore have only limited and short-term effects on the overall transient.

For these break sizes, the model features that have the most significant influence on the transient are the critical flow and leak discharge models; neither of these have been revised. Therefore, the reanalysis of Category 4 breaks is not considered necessary to demonstrate compliance with NUREG-0737 II.K.3.31.

# 7.0 RESULTS OF THE SBLOCA SPECTRUM ANALYSIS

## 7.1. Introduction

The results of the reanalysis of the 0.01 $ft^2$ break are discussed in detail in Reference 1. However, for purposes of continuity, analytical results for the 0.01 $ft^2$ break are presented again herein. This section will:

1. Present an overview of the transient results obtained using the previous model.
2. Provide a separate, similar overview of the transients from the reanalysis using the revised EM.
3. Compare the two sets of results for each break size and discuss the significant differences.

## 7.2. 0.07 $ft^2$ Break

### 7.2.1. Results Using The Previous EM

The sequence of significant events and the RCS pressure and mixture level history for this case are presented in Table 7-1 and Figures 7-1 and 7-2, respectively. Following break occurrence, the subcooled fluid blowdown, characterized by a rapid RCS depressurization, resulted in reactor trip. Turbine and RC pump trips occurred immediately thereafter as a result of the assumption that a loss of offsite power (LOOP) coincides with the reactor trip.

After RC pump trip, RCS flow was maintained at an appreciable rate. This was due in part to the noding scheme of this EM, which did not separate the "underside" of the hot leg U-bend from the SG tube region. This arrangement allowed a direct link to exist for heat transfer between those two regions. This modeling, therefore, had the effect of prolonging system flow. Previous results have justified the model for all except the Category 2 breaks (those which cause saturated RCS repressurization). For analyses of Category 2

breaks, the previous EM was modified (See Reference 3 and Figure 7-15) to include a separate control volume for the hot leg U-bend and SG upper plenum. This region is distinctly separated from the SG heat transfer region so that the correct sequence of heat transfer interruption and loss of circulation is predicted. For Category 3 breaks (the 0.07 $ft^2$ break being the most severe), loss of circulation and RCS repressurization have been demonstrated as being minor factors in the overall conclusion. Therefore, use of the previous model yielded acceptable results for the 0.07 $ft^2$ break and for other Category 3 breaks.

Another major contributor to the high RCS flow rate was the large rate of SG heat transfer that existed for the part of the transient during which main feedwater (MFW) was available. This high rate of SG heat transfer existed as a result of the simplistic nature of the SG model, as is discussed in detail in Section 5 of Reference 1. The significant heat transfer rate, due to MFW and large RC flows, enhanced the early rate of RCS depressurization.

Even after MFW flow ceased, the old SG model maintained a relatively large cold driving head, resulting in high RCS circulation flow rates. Those large flow rates continued until eventually being interrupted by voiding in the upper hot leg elbows (U-bends). A slowdown in the RCS depressurization rate accompanied the decrease in system flow as the hot legs voided.

The high rate of SG heat transfer early in the event caused secondary pressure to remain at the safety valve setpoint as RCS pressure decreased. By ~200 seconds, the primary-to-secondary differential pressures and temperatures, and hence the SG heat sink capacity, were reduced. Decreased SG heat transfer resulted, allowing AFW flow to raise SG levels. By ~300 seconds, secondary levels reached the automatic natural circulation setpoint of 50% OR which caused the shut AFW injection to be shut off. The SG heat transfer rates decreased accordingly; the break then became the primary mechanism for depressurizing the RCS. Consequently, the RCS depressurization rate became relatively slow for the remainder of the transient.

RCS inventory loss continued at a faster rate than the HPI flow rate until, by about 1370 seconds, core uncovering began. The core mixture level continued to decrease, falling to a minimum level approximately 1 ft below the top of the core at about 1495 seconds. By that time, RCS pressure had decreased

below 615 psia, initiating core flooding tank (CFT) injection. CFT flow, in conjunction with HPI flow, overcame RCS liquid losses and increased the reactor vessel liquid inventory thereafter, with the core becoming recovered after about 1750 seconds. With the core recovered and liquid replenishment exceeding liquid losses, long-term cooling was assured. The minimum core mixture level was approximately 11 ft, resulting in fuel cladding temperatures reaching a maximum value of approximately 1100F at 1700 seconds.

### 7.2.2. Results Using The Revised EM

The RCS pressure and mixture level transients and the key event sequence for the reanalysis of the 0.07 $ft^2$ break with the revised EM are shown in Figures 7-3 and 7-4, and in Table 7-2. Break opening caused RCS pressure to initially decrease rapidly in the subcooled blowdown phase, initiating reactor and turbine trip. A LOOP coincided with reactor trip, resulting in the tripping of the RC pumps and in the start of MFW coastdown.

The revised hot leg/SG noding scheme of the new EM resulted in decreasing RCS flow rates after the RC pumps tripped and hot leg voiding began to occur. As RCS flow and primary-to-secondary temperatures decreased, and as RCS fluid conditions changed, the revised EM continually updated, i.e. decreased the primary-to-secondary heat transfer coefficient. This decrease in heat transfer slowed the rate of primary depressurization; the depressurization rates slowed further after MFW flow ceased.

During this portion of the transient, the RCS pressure response was highly sensitive to the overall RCS energy balance (energy input versus energy removal). Between approximately 40 to 60 seconds (see Figure 7-3) RCS flow and SG heat transfer decreased, but when combined with the still-subcooled break flow and the cooling effects of HPI, overall energy removal was roughly in balance with core decay heat production. A temporary RCS pressure plateau resulted.

However, as SG heat transfer continued to decrease, this balance was upset in favor of energy addition, and between about 60 to 80 seconds a slight repressurization occurred. After saturation conditions developed at the break, energy removal out the break increased, allowing the RCS to again depressurize. This depressurization rate was enhanced after approximately 135

seconds when AFW spray began condensing primary steam as the plant entered the b-c cooling mode.

Between approximately 120 seconds (the end of the flow-controlled phase of the transient) and 750 seconds, RCS depressurization continued at a fairly constant rate. At 750 seconds, the natural circulation SG level setpoint was reached and AFW flow ceased. Primary and secondary systems reached thermal equilibrium shortly thereafter. Subsequently, the primary depressurization rate slowed as the break and HPI became the only mechanisms for reducing primary pressures.

RCS pressure continued downward, although at a slower rate than when SG cooling existed, until CFT actuation occurred at 1042 seconds. After 1042 seconds, RCS pressure continued to decrease, enhancing HPI and CFT flows and decreasing leak flow. With the core decay heat generation also continuing to decrease, the RCS liquid replacement rate increased while the liquid loss rate decreased. As a result, by 1600 seconds the liquid replacement rate had exceeded the liquid loss rate, and the RCS liquid inventory and mixture levels began to increase. This ensured that the core would remain covered and justified the termination of the reanalysis at 1600 seconds.

### 7.2.3. .07 $ft^2$ Break: Comparison of Results From Previous and Revised Studies

A comparison of the RCS pressure and mixture level histories and other significant parameters obtained using the previous and revised EMs is shown in Figures 7-5 and 7-6 and in Table 7-3. The RCS pressure transients for the two cases differ from the outset, as evidenced by the differences in the times required to trip the reactor, turbine, and RC pumps, and to reach saturated conditions in the RCS. Since break size is the same for both cases, early (pre-reactor trip) transient differences in depressurization rate were due mainly to differences in surge line modeling between the two EMs, as discussed previously in section 5.2. The greater surge line flow resistance in the new model lessened the pressurizer's effect on the RCS subcooled depressurization, allowing for a more rapid pressure decrease until after the reactor had tripped. This faster initial drop in pressure occurred in the new model, even though the SG heat transfer rate was less than in the old model, as was discussed in sections 7.2.1 and 7.2.2.

After the reactor, turbine, and RC pumps tripped, the effects of pressurizer surge line modeling on the RCS pressure response became insignificant. The RCS depressurization rate in both cases was dominated by core decay heat, SG heat transfer rate, RCS flow, and the break. In the revised case, the rates of SG heat transfer were lower during the MFW coastdown and were additionally influenced by reduced system flow. Therefore, after surge line effects had diminished, the revised depressurization rate became lower, mainly in response to the lower heat transfer rates. The reduced SG heat transfer in the new model occurring for approximately the first 80 seconds resulted in the hot legs' saturating at a higher pressure due to higher RCS energy levels. RCS depressurization in the new case became still slower than in the old case after MFW flow stopped. The times that MFW flow ceased reflect the differences in reactor trip times since the delay time to MFW flow shutoff was the same in both models.

ESFAS actuation occurred 11 seconds earlier in the revised case due to the higher actuation pressure assumed. AFW flow initiation occurred later in this case because a longer delay time from LOOP was assumed than in the previous model.

After RCS saturation, larger SG heat transfer rates in the previous case, due in part to increased system flow, allowed the RCS depressurization to continue. However, between 40 and 80-seconds in the new case, the relatively lower primary-to-secondary heat flow resulted in an energy balance and somewhat stable RCS pressure. At 80 seconds, saturation of the break region allowed increased energy removal and RCS depressurization in the new case.

During the middle portion of the transient (approximately 100 to 750 seconds), the new case depressurized faster than did the previous case. This was the result of SG and AFW modeling differences between the two cases, and the effects each had before and during this middle transient period. The old model utilized a single, large secondary control volume per SG, which contained a large quantity of saturated fluid that did not change properties significantly when small amounts of AFW were added. Also, the old model predicted high heat transfer rates prior to 100 seconds. These two factors resulted in secondary pressure remaining relatively high in the old case (see

Figure 7-7). The high secondary pressures and saturation temperatures allowed the RCS to reach thermal equilibrium with the secondary SGs relatively quickly (Table 7-3). The resultant low differential temperature and low heat transfer rates during the middle and latter stages of the transient meant that the SG had little influence on RCS pressure, which was mainly affected by core decay heat, the break and HPI. Without SG effects, the RCS therefore depressurized relatively slowly.

In the revised model, the more detailed SG secondary noding scheme allowed fluid properties within each node to be changed more readily. This effect, combined with lower predicted early transient RCS circulation rates and SG heat removal rates, allowed AFW to lower secondary pressures and saturation temperatures. Significant primary-to-secondary differential temperatures developed by approximately 100 seconds.

At about 135 seconds, sufficient RCS liquid inventory had been lost through the break to lower the primary mixture level in the SGs below the AFW injection elevation. At that point, the AFW spray began to directly condense primary steam, initiating b-c cooling. With the large primary-to-secondary differential temperatures that existed, heat transfer began through b-c cooling and continued until about 750 seconds, resulting in a faster RCS depressurization rate (Figure 7-5). At approximately 750 seconds, secondary levels reached the AFW level setpoint of 50% on the OR and AFW flow was shut off. Soon after AFW flow stopped, the primary and secondary systems reached thermal equilibrium. Thereafter, SG cooling and differences in SG modeling had no further effect on the transient. The break became the dominant factor affecting RCS pressure. After 750 seconds, the pressure in both cases decreased at a similar, comparatively slow rate.

The differences in RCS pressure at the time of AFW shutoff in both cases had an impact on the final recovery stages of the events. With the slower RCS pressure decrease after AFW stopped, the time to actuate CFT injection, relative to the time of core uncovering became a function of the pressure at AFW shutoff in both studies. The lower RCS pressure at AFW shutoff in the revised case enabled RCS pressure to decrease to CFT pressure before core uncovering started. With the added inventory from the CFTs, the reactor

vessel (RV) mixture height was maintained above the top of the core in the new case.

## 7.3. 0.04 ft$^2$ Break

### 7.3.1. Results Using The Previous EM

Significant data regarding RCS pressure and mixture level transients, plus other parameters are found in Figures 7-8 and 7-9 and in Table 7-4. This transient progressed in much the same manner as that for the .07 ft$^2$ case discussed in section 7.2.1, except that the rates at which the transient parameters changed were relatively slower. Once again, the SG and hot leg U-bend modeling tended to maintain RCS flow for an extended period of time, resulting in sufficient heat transfer that, when coupled with break flow, allowed for a fairly constant RCS depressurization.

Relative to the depressurization transient of the previous 0.07 ft$^2$ case, the 0.04 ft$^2$ case caused the RCS to depressurize in a similar manner. AFW was shut off when the SG level reached the 50% OR setpoint at ~350 seconds. After that, the SG heat transfer rate was reduced so much that the energy removed from the RCS via the break became less than the energy added from the core decay heat. The excess energy addition resulted in a slight RCS repressurization that lasted from approximately 350 to 700 seconds. At that time, break flow and HPI were sufficient to offset the decreasing core decay heat, allowing RCS pressure to decrease again.

Once resumed, depressurization continued steadily for the remainder of the transient. The inner RV mixture level also declined during the latter part of the event. However, liquid boiloff due to core decay heat decreased significantly with time while liquid losses due to the break decreased at lower RCS pressures. By approximately 3100 seconds, the HPI flow, which had been increasing as RCS pressure fell, matched the rate of liquid loss. This matchup occurred prior to core uncovering; afterward, the reactor vessel liquid volume increased.

An important conclusion of these results is that, under the design-basis assumptions previously listed, one HPI system has the injection capability to keep the core covered for the 0.04 ft$^2$ break, without additional injection from the CFTs. Furthermore, since smaller break sizes result in less inven-

tory loss, design-basis HPI flow is capable of preventing core uncovering from breaks <u>less than</u> 0.04 $ft^2$ in area as well.

### 7.3.2. Results Using The Revised EM

RCS pressure and mixture responses for this case are presented in Figures 7-10 and 7-11 and in Table 7-5. The transient was similar in most respects to that of the revised 0.07 $ft^2$ case described in section 7.2.2, although events are of different magnitude and occur later in time. As with the revised 0.07 $ft^2$ case, the revised 0.04 $ft^2$ break allowed pressure to decrease after the break opening, with the rate of decrease slowing after the hot legs began to saturate.

The impact of the revised model was to decrease SG heat transfer during the subcooled blowdown phase due to degraded RCS flow and increased noding detail in the hot leg U-bend and upper SG regions. At approximately 70 seconds, after the end of the subcooled blowdown, an energy imbalance existed that resulted in a slight RCS repressurization that lasted until about 120 seconds. At 120 seconds, the break region saturated, permitting increased energy discharge. This greater energy loss out of the break, combined with the cooling effects of AFW spray, altered the energy imbalance in favor of RCS energy losses.

The RCS depressurization was further enhanced when the condensation of primary steam (b-c cooling) began at approximately 480 seconds. The RCS lost pressure at a relatively rapid rate from approximately 120 to 730 seconds. At 730 seconds secondary levels reached their 50% OR setpoint and AFW was shut off. After 730 seconds, RCS depressurization continued, but at a relatively slower rate. The mechanisms causing depressurization for the remainder of the transient were the break flow and the cooling effects of HPI. Pressure and mixture levels continued to decrease in the RCS until the liquid replacement rate matched the liquid loss rate. This matchup occurred at approximately 3100 seconds, with the RV mixture height slightly more than 3.5 feet above the top of the core. Liquid injection (HPI only) exceeded liquid losses after 3100 seconds, verifying that long-term cooling had been established, and allowing the analysis to end.

### 7.3.3. Comparison of The Results From The Previous and Revised Studies

Pertinent information for the 0.04 $ft^2$ cases analyzed with both the previous and revised models is compared in Figures 7-12, 7-13 and 7-14 and in Table 7-6. As with the old and new analyses for the 0.07 $ft^2$ break, compared in section 7.2.3, the transient responses of the 0.04 $ft^2$ break are different for the two evaluation models. The major causes for these differences have been discussed previously and are summarized briefly below.

1. Differences in surge line modeling
   The revised case experienced a faster depressurization early in the transient. Reactor, turbine, and RC pump trips, and the beginning of MFW coastdown were slightly affected.

2. SG model differences
   During and after the initial subcooled blowdown phase of fairly rapid RCS depressurization, the revised model predicted less primary-to-secondary heat transfer. This prediction was mainly due to the modeling detail of the hot leg U-bend and the SG upper plenum. The net effect is to allow for steam formation in the hot leg U-bends, thereby degrading RCS flow and decreasing heat transfer rates.

3. AFW model differences
   The AFW model influence within the SG model allows for steam pressure reductions during times of little primary-to-secondary heat transfer when AFW injection is being supplied to raise the SG secondary level. Reduced SG pressure and saturation temperature resulted in prolonged b-c cooling once a condensation surface had been established. This condition was demonstrated from 480 to 730 seconds in the revised 0.04 $ft^2$ case.

The relatively large SG heat sink potential that existed during b-c in the revised case more than offset the relatively low subcooled blowdown heat transfer coefficients predicted by the new model. Enhanced b-c SG heat removal in the new case caused RCS pressure to fall below that of the previous case before AFW flow was stopped when the SG 50% OR level setpoint was reached. At the lower RCS pressures, HPI was able to deliver more flow and

thus to maintain more liquid in the system for the new case. Thus, the newer model predicts a greater mixture height margin above the top of the core for this break size.

## 7.4. 0.01 ft$^2$ Break

The 0.01 ft$^2$ break was selected as a representative size for Category 2 breaks. SG performance is important for energy removal in Category 2 break transients. This conclusion is based on analyses (Reference 4) using the previous EM, and is the basis for defining AFW spray and pool level requirements (References 7 and 8) to ensure core protection and long-term cooling for these break sizes. Thus, for the 0.01 ft$^2$ break, the effects of SG and AFW modeling may be significant. The 0.01 ft$^2$ break was analyzed with the previous and revised EMs to demonstrate that Category 2 SBLOCA phenomena can be predicted with both models. The break was not analysed to show that the core does or does not remain covered. For this break category, the ECCS has been shown (References 3, 7, and 8) to prevent core uncovering, thus maintaining fuel cladding temperatures within a few degrees of the RCS saturation temperature. Therefore, this category of breaks does not constitute a challenge to the requirements of 10 CFR 50.46.

### 7.4.1. Results Using The Previous EM

The RCS pressure, SG pressure and hot leg mixture level histories, and sequence of key events are shown in Figures 7-16 and 7-17 and Table 7-7. The noding diagram used for this study is shown in Figure 7-15. As previously discussed (section 7.2.1) this noding arrangement is modified from the standard EM schematic of Figure 5-1 to separate the hot leg U-bend and SG upper plenum from the SG heat transfer region. This arrangement facilitates the prediction of the repressurization phenomenon that begins when natural circulation and SG heat transfer are lost. The repressurization phase lasts until a condensation surface is established on the primary side of the SG tubes, restoring heat transfer (b-c) and depressurizing the RCS, thus providing long-term cooling.

In this analysis, the RCS depressurized rapidly over the first 100 seconds to reach a saturation pressure of about 1400 psia (Figure 7-16). At this point,

steam formation in the hot leg and upper plenum slowed the rate of depressurization.

As a result of continuous SG heat transfer, RCS depressurization continued until about 650 seconds, when the hot legs voided and natural circulation ceased. The loss of SG heat removal and the small break size caused the primary system pressure to began increasing. At 1500 seconds, the maximum system pressure reached 1750 psia and began to decrease slowly because steam condensation by the SG was established. This additional energy removal resulted in a decreasing RCS pressure transient. The RCS pressure was then controlled by b-c cooling.

7.4.2. Results Using The Revised EM

A listing of the sequence of key events for this case is shown in Table 7-8, and the transient response is shown in Figures 7-18 and 7-19. The new case initially depressurized to approximately 1500 psia, after which the new SG model exhibited less heat transfer, effectively stopping the RCS depressurization after the initial blowdown.

As the system reached saturation, the energy produced was balanced by the energy removed, and the pressure response remained essentially steady at 1520 psia through 350 seconds. As the transient continued, more inventory was lost through the break, which caused a reduction in the hot leg level and reduced two-phase flow into the SG. Consequently, reactor vessel steam relief was maintained through the internals vent valves, which saturated the upper downcomer, thereby supplying saturated fluid to the break node with condensation in the cold legs ensuing. This sequence of events allowed for system depressurization to occur at 375 seconds. However, as SG heat transfer decreased, this depressurization was short-lived due to an overall imbalance in energy removal versus energy production. This imbalance quickly changed the pressure response to an upward trend, which continued until 440 seconds. At 420 seconds, steam in the upper downcomer mixture region separated and cold leg condensation terminated. Then, reactor vessel steam relief increased to the hot legs, which caused an increase in the two-phase circulation through the SG. SG heat transfer was increased as steam in the two-phase flow condensed in the SG. Energy removal had thus been increased above the amount of energy produced, and a depressurization resulted at 440 seconds. By 530

seconds, the energy terms had again balanced, causing a steady system pressure at 1500 psia.

From 500 to 550 seconds, enough system inventory had been lost to disrupt natural circulation again which resulted in a decrease in SG energy removal and an increase in RCS pressure. By 630 seconds, the decrease in steam flow to the saturated cold legs caused both an increased flow to the hot legs and a hot leg level swell, which returned two-phase circulation to the SG. With the return of SG heat transfer, a depressurization occurred until 670 seconds. By that time, however, the system liquid inventory was not sufficient to maintain the necessary hot leg level, the circulation pattern was lost again and the depressurization stopped. System pressure remained steady at 1470 psia as the energy terms balanced once again. The main contributors to energy removal were steam relief out of the break and ECCS flow, with a minor contribution by SG heat transfer. During the next 3 transient minutes, intermittent two-phase circulation existed, allowing for enough SG heat transfer to aid in the system energy balances.

At 940 seconds, the intermittent two-phase circulation was lost. At 950 seconds, the secondary side level setpoint (50% on the OR) was reached, and AFW was turned off. The SGs were then unable to remove heat, and the energy balance was upset, causing a continuous increase in the RCS pressure. This analysis was similar to the previous one in which the loss of natural circulation caused a repressurization at 650 seconds. The repressurization continued until the level in the SG primary side decreased sufficiently to expose a condensation surface at approximately 1500 seconds. At that time AFW was injected, and the expected result of primary side steam condensation was established, which brought about an abrupt end to the repressurization, enabling the system pressure to eventually be controlled near the SG secondary pressure and assuring RCS inventory recovery, core protection, and long-term cooling.

### 7.4.3. Comparison of The Results From The Previous and Revised Studies

A comparison of the RCS pressures and mixture levels is shown in Figures 7-20, 7-21 and 7-22. A comparison of the sequence of events is presented in Table 7-9. Differences in the transient response are due primarily to the influence of the SG and AFW model revisions. The arguments presented in earlier

discussions of SBLOCA results comparisons, sections 7.2.3 and 7.3.3, are also valid for the 0.01 ft$^2$ results. Basically, the previous SG model allowed for excessive heat transfer during the initial subcooled blowdown portion of these transients, which resulted in larger RCS flow rates and decreased RCS pressures. Once circulation was lost, the influence of the SG was negligible until a condensation surface was established. This was evidenced by the fairly early repressurization exhibited by the previous EM. In the previous SG model, condensation heat transfer is only a function of the primary-to-secondary differential temperature, resulting in a slow recovery to lower RCS pressures.

The revised SG model provided a more mechanistic response, and therefore the initial blowdown period did not continue after the RCS depressurized to saturation pressure. Due to AFW injection, a fairly stable saturation pressure was maintained for the majority of the transient. The characteristic repressurization for Category 2 breaks was only demonstrated after AFW was turned off (50% OR level setpoint reached). The common feature of both 0.01 ft$^2$ analyses is that b-c cooling was predicted to result in the RCS pressures at being controlled low values, enabling the HPI system to provide for core protection and long-term cooling. Fuel cladding temperatures will be maintained at or below RCS saturation temperatures for the entire transient. To ensure this result, References 7 and 8 provide minimum AFW spray and SG level requirements that are necessary based on conservative EM results and calculations.

Table 7-1. Sequence Of Key Events For The 0.07 $Ft^2$ Break At RC Pump Discharge, Analyzed With Previous SBLOCA Evaluation Model

| Event | Time, Sec |
|---|---|
| Break Occurs | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia) Turbine Trips, RC Pumps Trip on coincident LOOP, MFW Coastdown Begins, AFW system Actuates | 6 |
| Hot Legs Saturate | 15 |
| MFW Flow Coastdown Ends | 20 |
| ESFAS Actuates on Low RCS Pressure, 1365 psia | 32 |
| AFW Injection Commences | 42 |
| HPI Commences | 67 |
| Break Region Saturates | 84 |
| Flow Controlled Phase Ends | 190 |
| SG Natural Circulation Level Setpoint Reached (50% O.R.) AFW Shuts Off (Broken Loop SG/Intact Loop SG) | 270/300 |
| Primary and Secondary Systems Reach Thermal Equilibrium (Broken Loop/Intact Loop) | 290/320 |
| Core Uncovering Begins | 1370 |
| CFT Injection Begins | 1495 |
| Core Recovered | 1800 |

Table 7-2. Sequence of Key Events for a 0.07 $Ft^2$ Break At RC Pump Discharge, Analyzed With The Revised SBLOCA Evaluation Model

| Event | Time, Sec |
|---|---|
| Break Occurs | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on coincident LOOP, MFW Coastdown Begins, AFW System Actuates | 3 |
| Hot Leg Saturates | 12 |
| MFW Coastdown Ends | 17 |
| ESFAS Actuates on Low RCS Pressure, 1495 psia | 21 |
| AFW Injection Commences | 44 |
| High Pressure Injection Commences | 56 |
| Break Region Saturates | 82 |
| Flow Controlled Phase Ends | 120 |
| Primary Steam Condensation (Boiler-Condenser) Begins | 135 |
| SG Natural Circulation Level Setpoint Reached (50% OR); AFW Flow Shuts off (Broken Loop SG/Intact Loop SG) | 750/755 |
| Primary and Secondary Systems Reach Thermal Equilibrium | 780 |
| CFT Injection Begins | 1042 |

Table 7-3. Comparison of The Results of The Previous and Revised Evaluation Models For The 0.07 $Ft^2$ Break At RC Pump Discharge

| Event (Parameter)/Units | Previous Model | Revised Model |
|---|---|---|
| Break Occurs/sec | 0 | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia) Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuates/sec | 6 | 3 |
| Total Heat Transferred to SGs at t = 10 sec/Btux$10^7$ | 2.422 | 1.988 |
| Hot Leg Saturates/sec | 15 | 12 |
| RCS Pressure at Time of Hot Leg Saturation/psia | 1570 | 1590 |
| MFW Coastdown Ends/sec | 20 | 17 |
| Total Heat Transferred to SG's at t = 20 sec/Btux$10^7$ | 3.779 | 3.076 |
| ESFAS Actuates on Low RCS Pressure 1365 psia Previous Model, 1495 psia Revised Model/sec | 32 | 21 |
| AFW Injection Commences | 42 | 44 |
| Primary-to-Secondary Differential Temperature at t = 80 sec/F | 1 | 11 |
| Break Region Saturates/sec | 84 | 82 |
| Flow Controlled Phase Ends/sec | 190 | 120 |
| AFW Flow Shuts Off (When Secondary Level Setpoint Reached)/sec | 300 | 750 |
| Primary and Secondary Systems Reach Thermal Equilibrium/sec | 320 | 780 |
| Total Heat Transferred to SGs at time Thermal Equilibrium reached/Btu X$10^7$ | 8.020 | 8.434 |
| RCS Pressure at Time Thermal Equilibrium Reached/psia | 960 | 640 |
| CFT Injection Begins/sec | 1495 | 1042 |
| Core Uncovering Begins/sec | 1370 | (a) |
| Core Recovery and Long-Term Cooling Begins/sec | 1800 | (a) |

(a) Core was still covered when transient was ended at 1600 sec.

Table 7-4. Sequence of Key Events for a 0.04 $Ft^2$ Break at RC Pump Discharge, Analyzed With The Previous SBLOCA Evaluation Model

| Event | Time, Sec |
|---|---|
| Break Occurs | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuates | 11 |
| MFW Coastdown Ends | 25 |
| Hot Leg Saturates | 27 |
| ESFAS Actuates on Low RCS Pressure (1365 psia) | 45 |
| AFW Injection Commences | 47 |
| High Pressure Injection Commences | 80 |
| Break Region Saturates | 140 |
| Flow Controlled Phase Ends | 334 |
| SG Natural Circulation Level Setpoint Reached, AFW Flow Ceases (Broken Loop SG/Intact Loop SG) | 320/360 |
| Primary and Secondary Systems Reach Thermal Equilibrium (Broken Loop/Intact Loop) | 730/740 |
| Minimum Mixture Height Above Top of Core, ft @ sec | 1.58 @ 3030 |

Table 7-5. Sequence of Key Events for a 0.04 $Ft^2$ Break at RC Pump Discharge, Analyzed With The Revised SBLOCA Evaluation Model

| Event | Time, Sec |
|---|---|
| Break Occurs | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuated | 10 |
| MFW Coastdown Ends | 24 |
| Hot Leg Saturates | 25 |
| ESFAS Actuates on Low RCS Pressure, 1495 psia | 30 |
| AFW Injection Commences | 50 |
| High Pressure Injection Commences | 65 |
| Break Region Saturates | 121 |
| Flow Controlled Phase Ends | 317 |
| SG Natural Circulation Level Reached (50% OR), AFW Flow Shuts Off (Broken Loop SG/Intact Loop SG) | 720/730 |
| Primary and Secondary Systems Reach Thermal Equilibrium (Broken Loop/Intact Loop) | 1275/1100 |
| Minimum Mixture Height Above Top of Core, ft @ sec | 3.75 @ 3065 |

Table 7-6. Comparison of The Results of The Previous and Revised Evaluation Models For a 0.04 $Ft^2$ Break at RC Pump Discharge

| Event (Parameter)/Units | Previous Model | Revised Model |
|---|---|---|
| Break Occurs/sec | 0 | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuates/sec | 11 | 10 |
| Total Heat Transferred to SGs at t = 15 sec/Btu $x10^7$ | 3.772 | 3.444 |
| NFW Coastdown Ends/sec | 25 | 24 |
| Total Heat Transferred to SGs at t = 25 sec./Btu $x10^7$ | 5.156 | 4.611 |
| Hot Leg Saturates/sec | 27 | 25 |
| RCS Pressure at Time of Hot Leg Saturation/psia | 1515 | 1530 |
| ESFAS Actuation on Low RCS Pressure, 1365 psia Previous Model, 1495 psia Revised Model/sec | 45 | 30 |
| AFW Injection Commences/sec | 47 | 50 |
| High Pressure Injection Commences/sec | 80 | 65 |
| Primary-to-Secondary Differential Pressure at t = 115 sec/psid | 80 | 480 |
| Break Region Saturates/sec | 140 | 121 |
| Flow Controlled Phase Ends/sec | 334 | 317 |
| AFW Flow Shuts Off (When Secondary level Setpoint Reached)/sec | 360 | 730 |
| Primary and Secondary Systems Reach Thermal Equilibrium/sec | 740 | 1275 |
| Total Heat Transferred to SGs at Time Thermal Equilibrium Reached/Btu $x10^7$ | 10.449 | 10.538 |
| RCS Pressure At Time Thermal Equilibrium Reached/psia | 1060 | 880 |
| Minimum Mixture Height <u>Above</u> Top Of Core,/ft @ sec | 1.58 @ 3030 | 3.75 @ 3065 |

Table 7-7. Sequence of Key Events for a 0.01 $Ft^2$ Break At RC Pump Discharge, Analyzed With The Previous SBLOCA Evaluation Model

| Event | Time, Sec |
|---|---|
| Break Occurs | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuates | 50 |
| MFW Coastdown Ends | 65 |
| AFW Injection Commences | 90 |
| Hot Leg Saturates | 100 |
| ESFAS Actuates on Low RCS Pressure, 1365 psia | 155 |
| High-Pressure Injection Begins | 190 |
| Flow-Controlled Phase Ends (Unbroken Loop/Intact Loop) | 340/650 |
| Break Region Saturates | 470 |

Table 7-8. Sequence of Key Events For a 0.01 $Ft^2$ Break at RC Pump Discharge, Analyzed With The Revised SBLOCA Evaluation Model

| Event | Time, Sec |
|---|---|
| Break Occurs | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuates | 43 |
| MFW Coastdown Ends | 58 |
| AFW Injection Commences | 85 |
| ESFAS Actuates on Low RCS Pressure, 1495 psia | 115 |
| Hot Leg Saturates | 130 |
| High Pressure Injection Begins | 150 |
| Flow Controlled Phase Ends | 505 |
| Break Region Saturates | 530 |
| SG Natural Circulation Level Setpoint Reached 50% OR AFW Shuts Off | 950 |

Table 7-9. Comparison of The Results of Previous and Revised Evaluation Models For The 0.01 $Ft^2$ Break at RC Pump Discharge

| Event (Parameter)/Units | Previous Model | Revised Model |
|---|---|---|
| Break occurs/sec | 0 | 0 |
| Reactor Trips on Low RCS Pressure (1900 psia), Turbine Trips, RC Pumps Trip on Coincident LOOP, MFW Coastdown Begins, AFW System Actuates/sec | 50 | 43 |
| MFW Coastdown Ends/sec | 65 | 58 |
| AFW Injection Commences/sec | 90 | 85 |
| Hot Leg Saturates/sec | 100 | 130 |
| ESFAS Actuates on Low RCS Pressure 1365 psia Previous Model, 1495 psia Revised Model/sec | 155 | 115 |
| High Pressure Injection Begins/sec | 190 | 150 |
| Flow Controlled Phase Ends/sec | 650 | 505 |
| Break Region Saturates/sec | 470 | 530 |

Figure 7-1 - RCS and Secondary Pressures, .07 $Ft^2$ Break at RC Pump Discharge, Previous Model

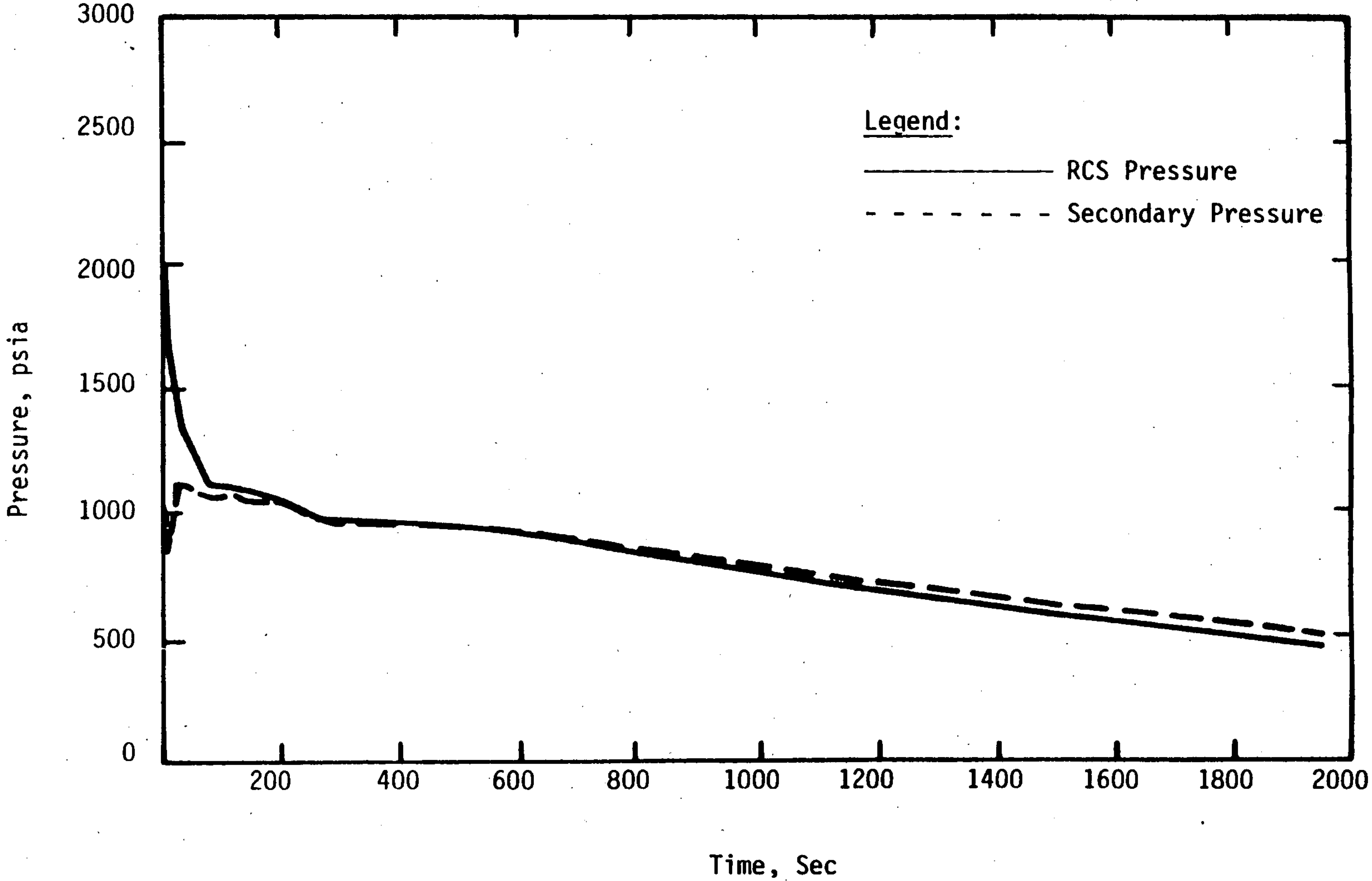


Figure 7-2 - Hot Leg and Reactor Vessel Mixture Heights, .07 Ft$^2$ Break at RC Pump Discharge, Previous Model

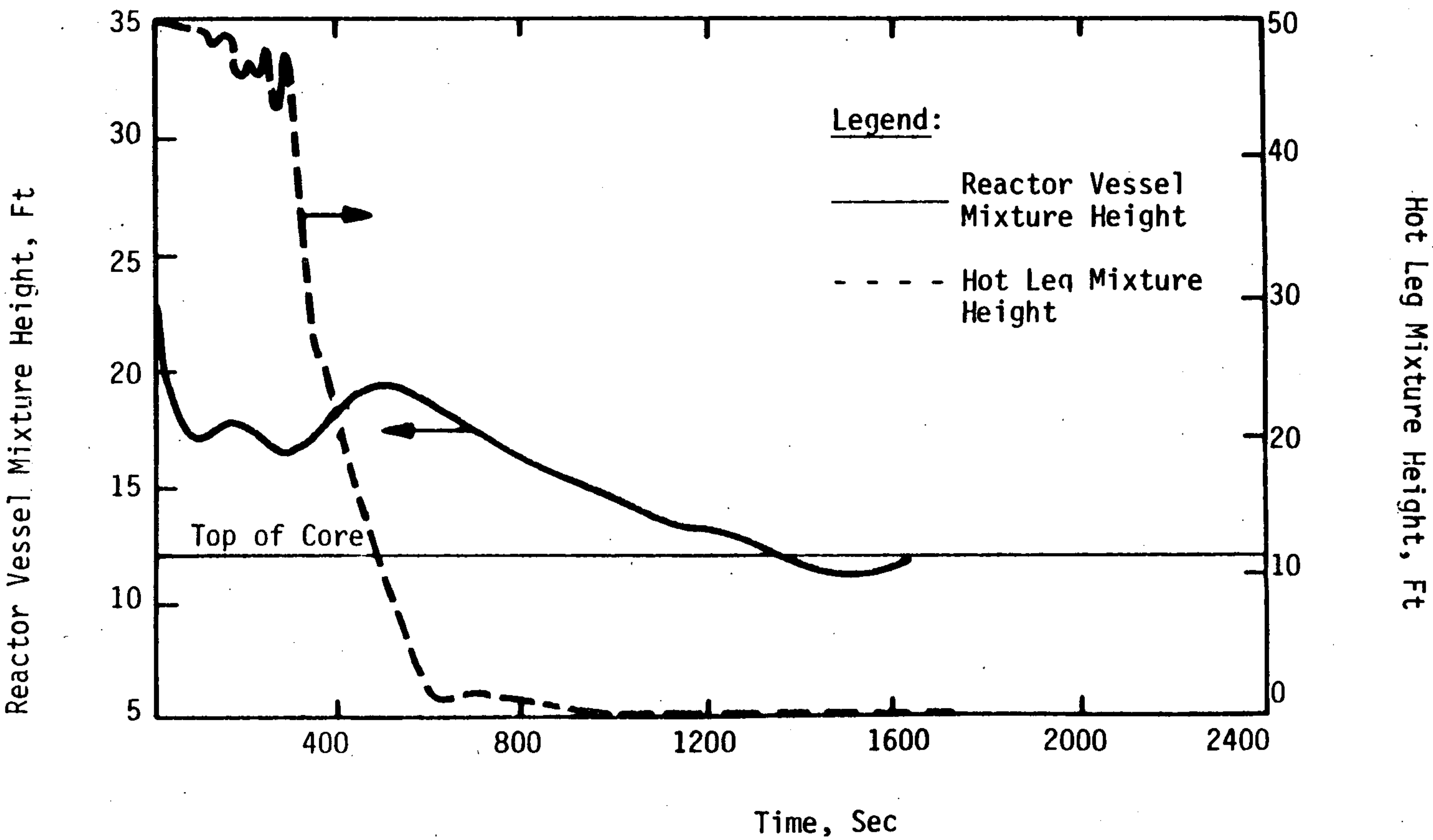


Figure 7-3 - RCS and Secondary Pressures, .07 $Ft^2$ Break at RC Pump Discharge, Revised Model

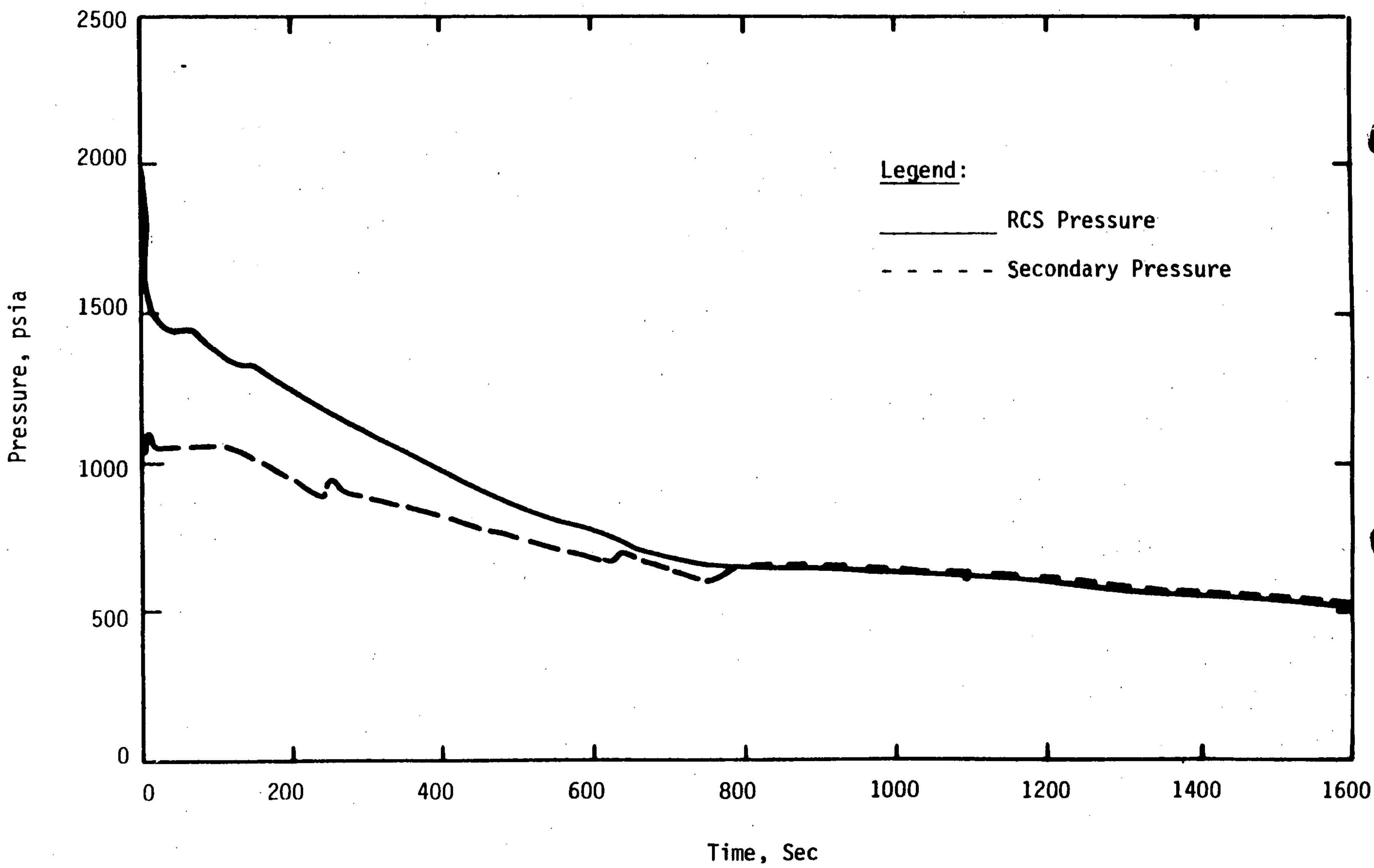


Figure 7-4 - Hot Leg and Reactor Vessel Heights, .07 $Ft^2$
Break at RC Pump Discharge, Revised Model

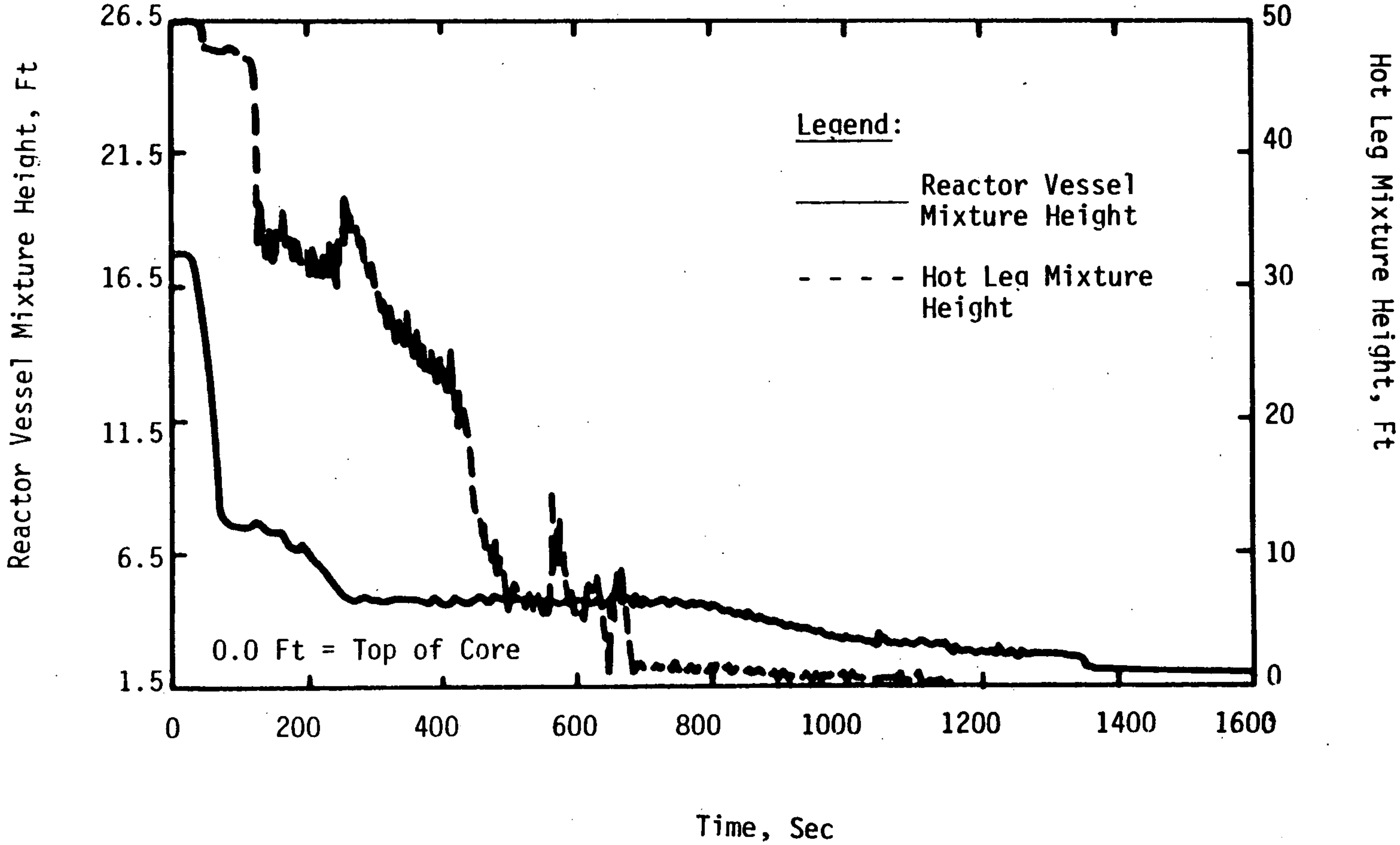


Figure 7-5 RCS Pressure, .07 $Ft^2$ Break at RC Pump Discharge, Comparison
Comparison of Results of Previous and Revised Models

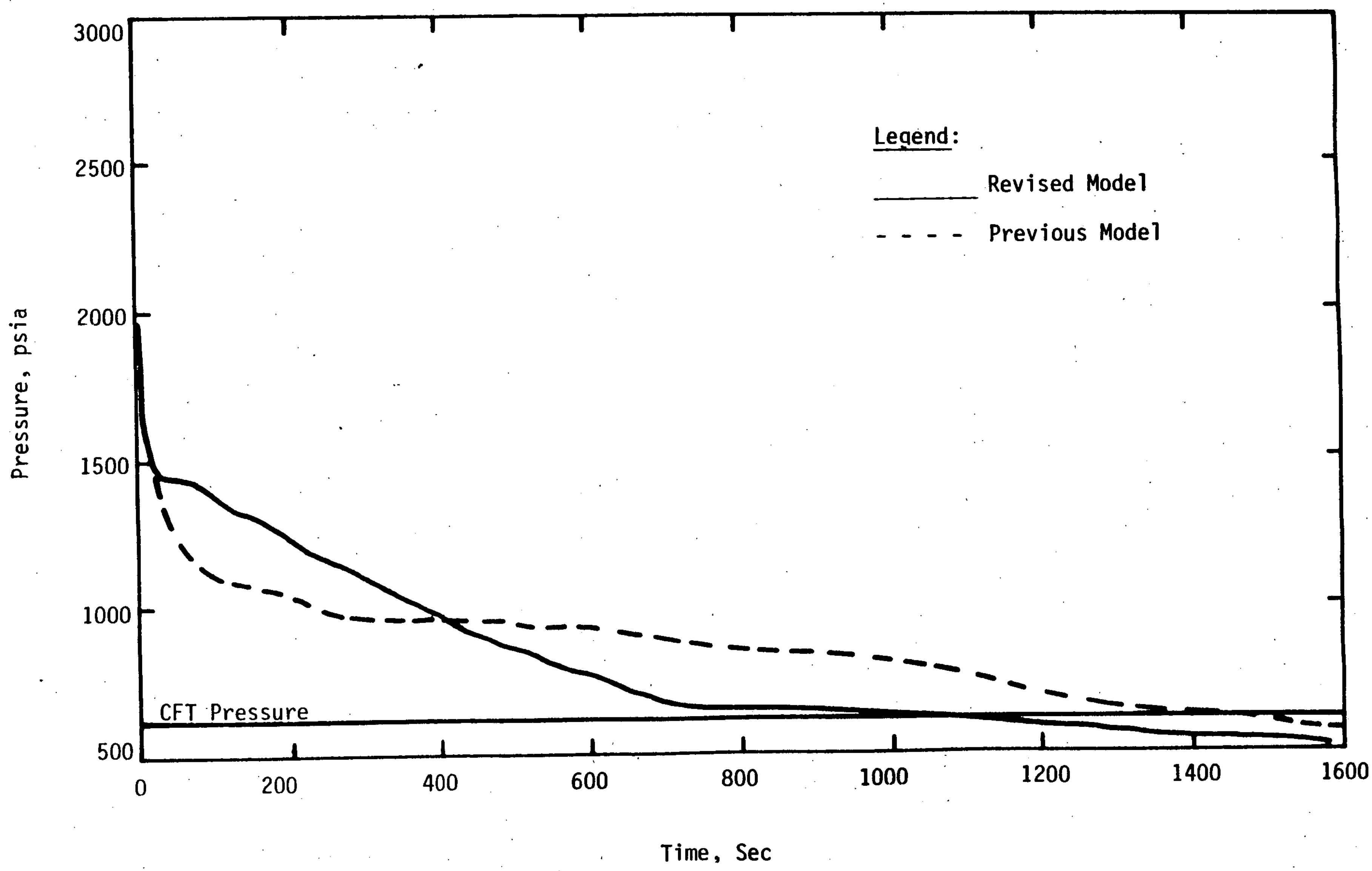


Figure 7-6 - Inner Vessel Mixture Height, .07 $Ft^2$ Break at RC Pump Discharge, Comparison of Results of Previous and Revised Models

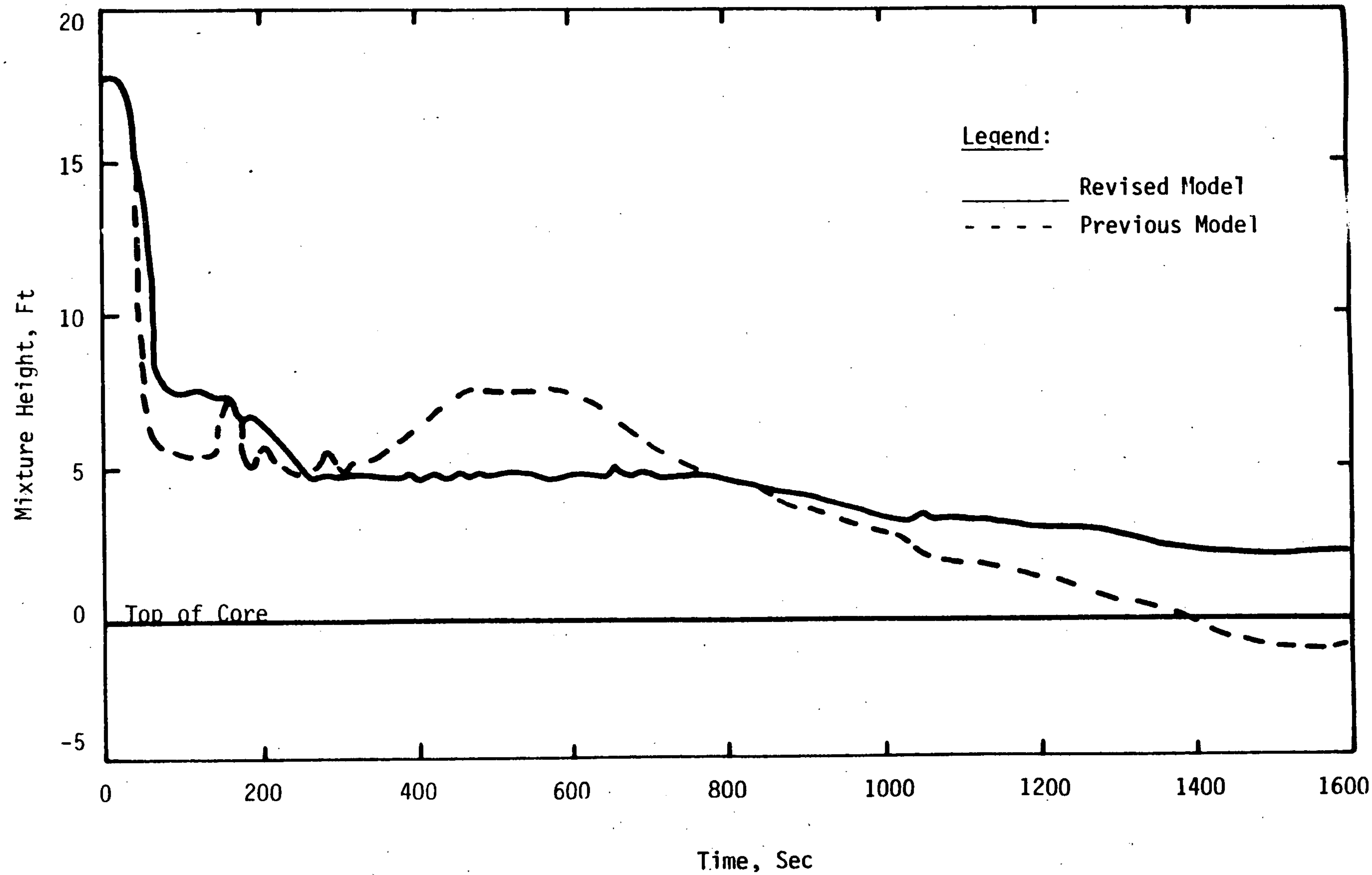


Figure 7-7 - Secondary Pressures, .07 Ft$^2$ Break at RC Pump Discharge, Comparison of Results of Previous and Revised Models

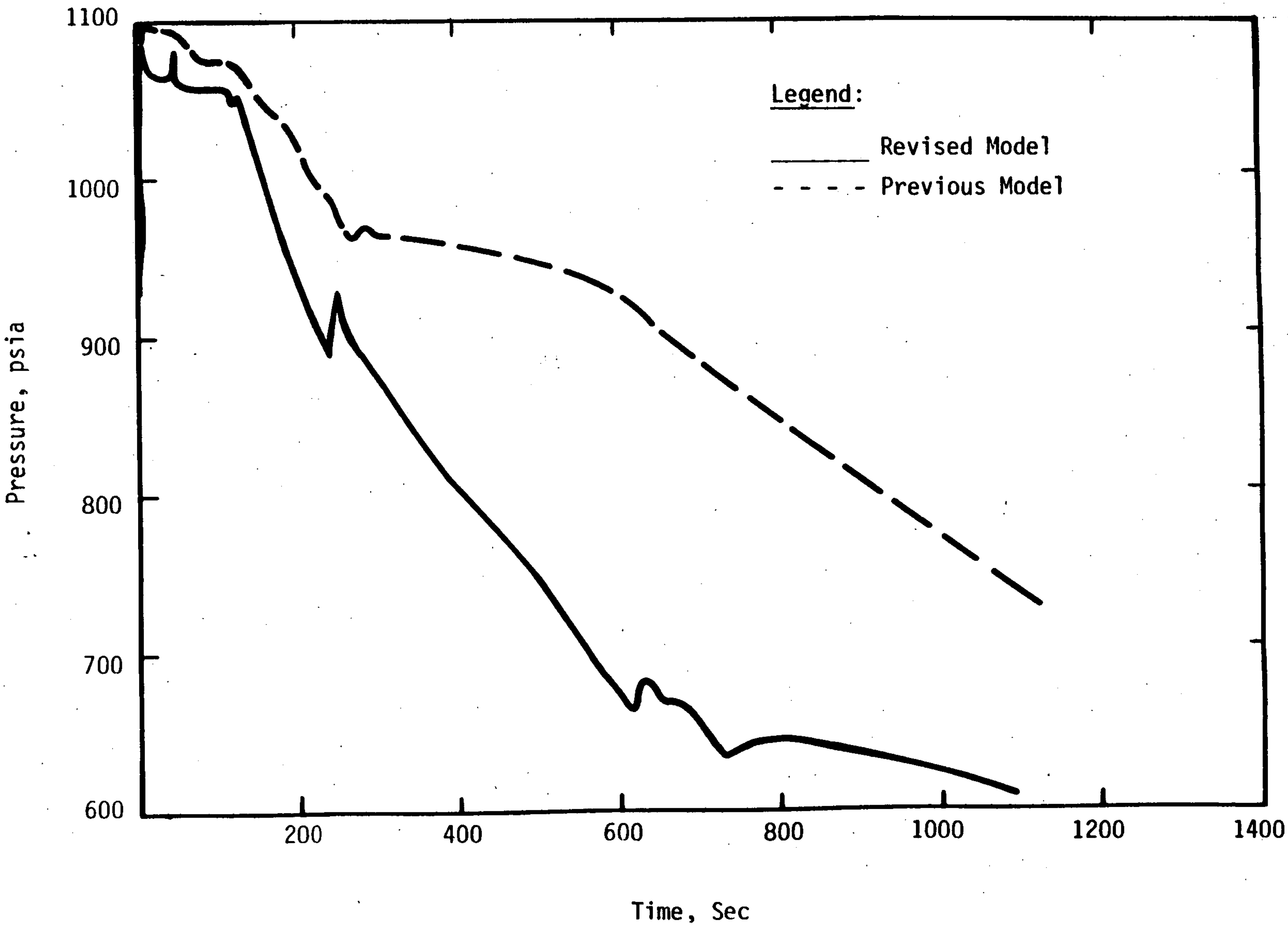


Figure 7-8 - RCS and Secondary Pressures, .04 $Ft^2$ Break at RC Pump Discharge, Previous Model

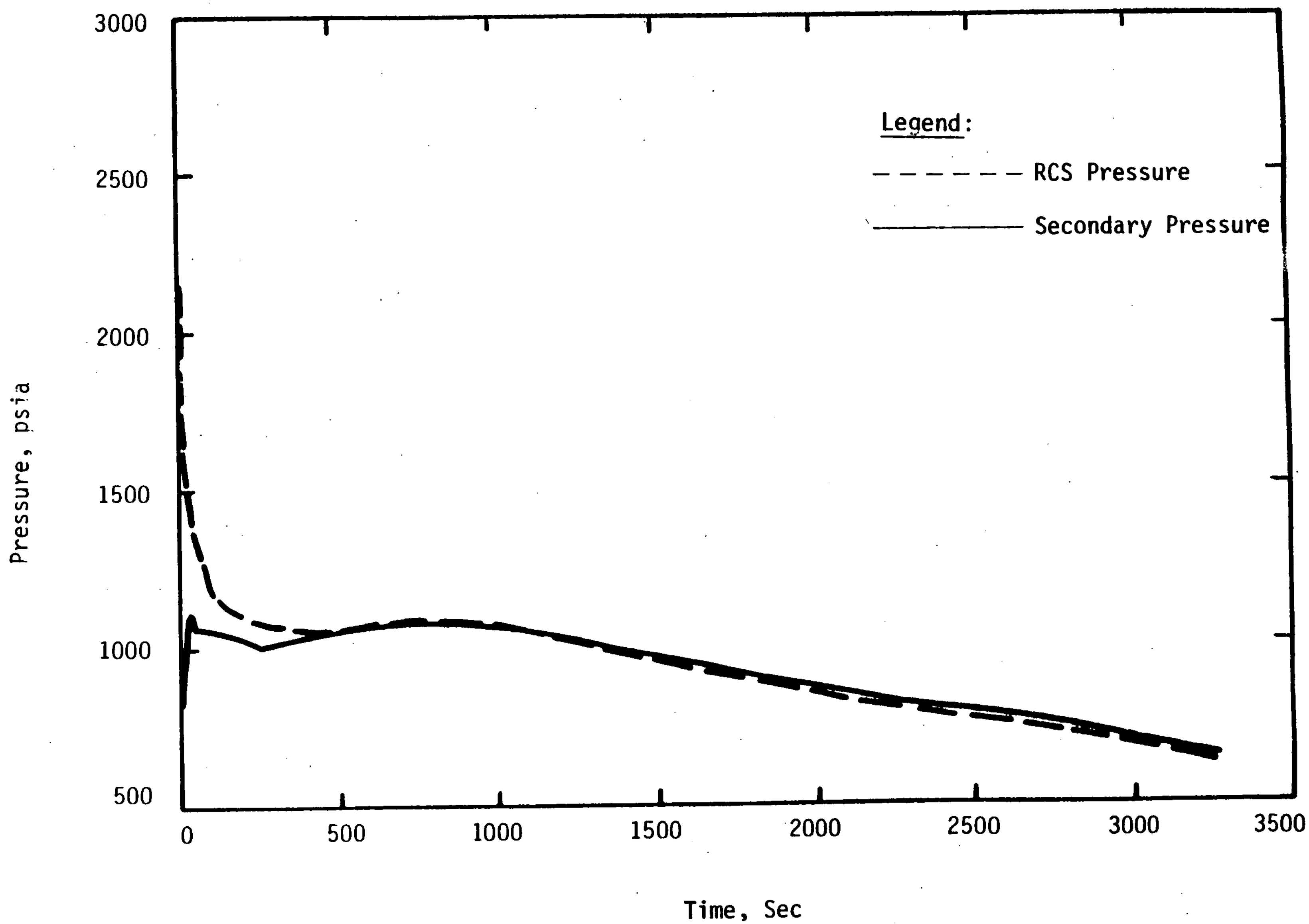


Figure 7-9 - Hot Leg and Reactor Vessel Mixture Heights, .04 $Ft^2$ Break at RC Pump Discharge, Previous Model

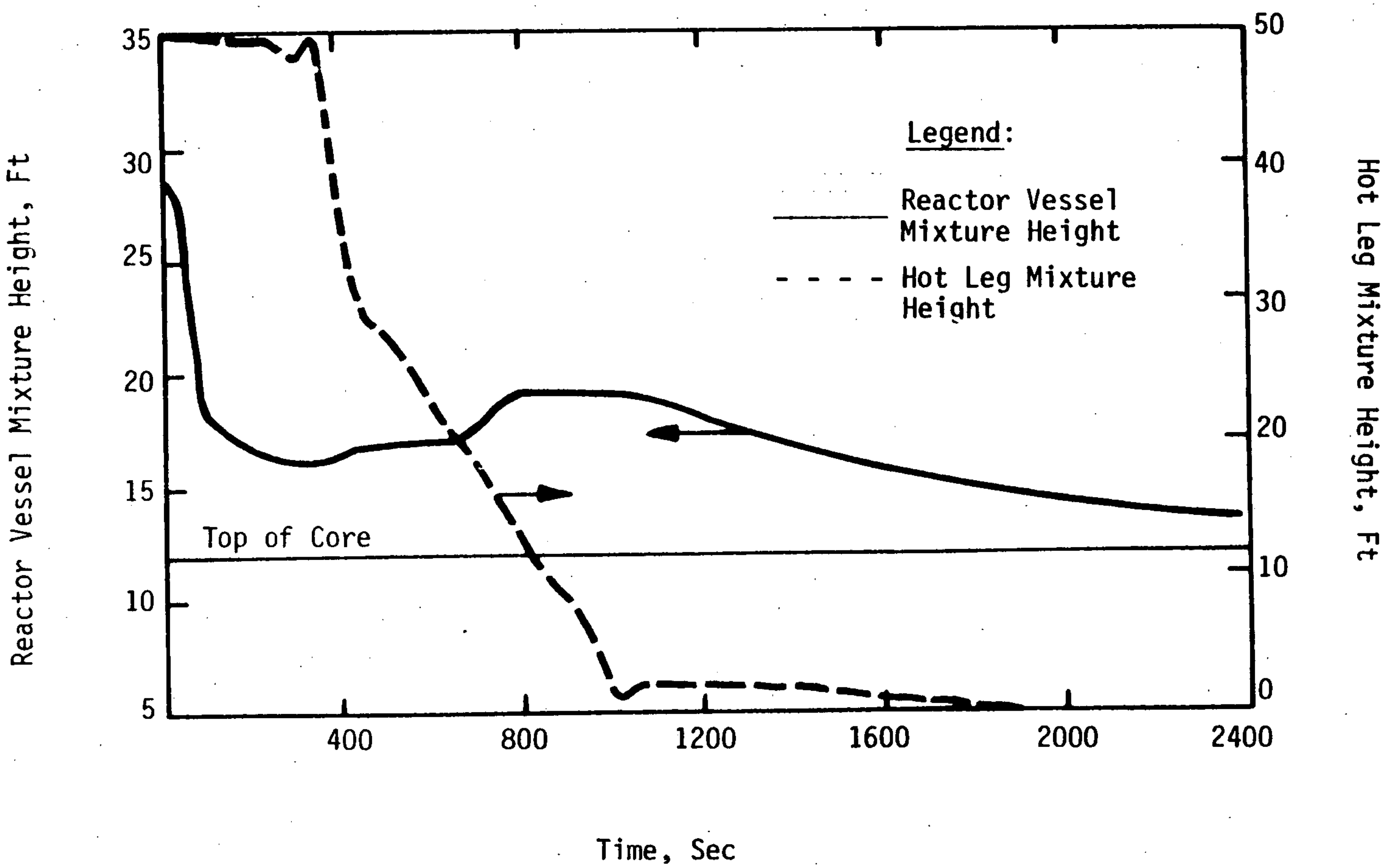


Figure 7-10 - RCS and Secondary Pressures, .04 $Ft^2$ Break at RC Pump Discharge, Revised Model

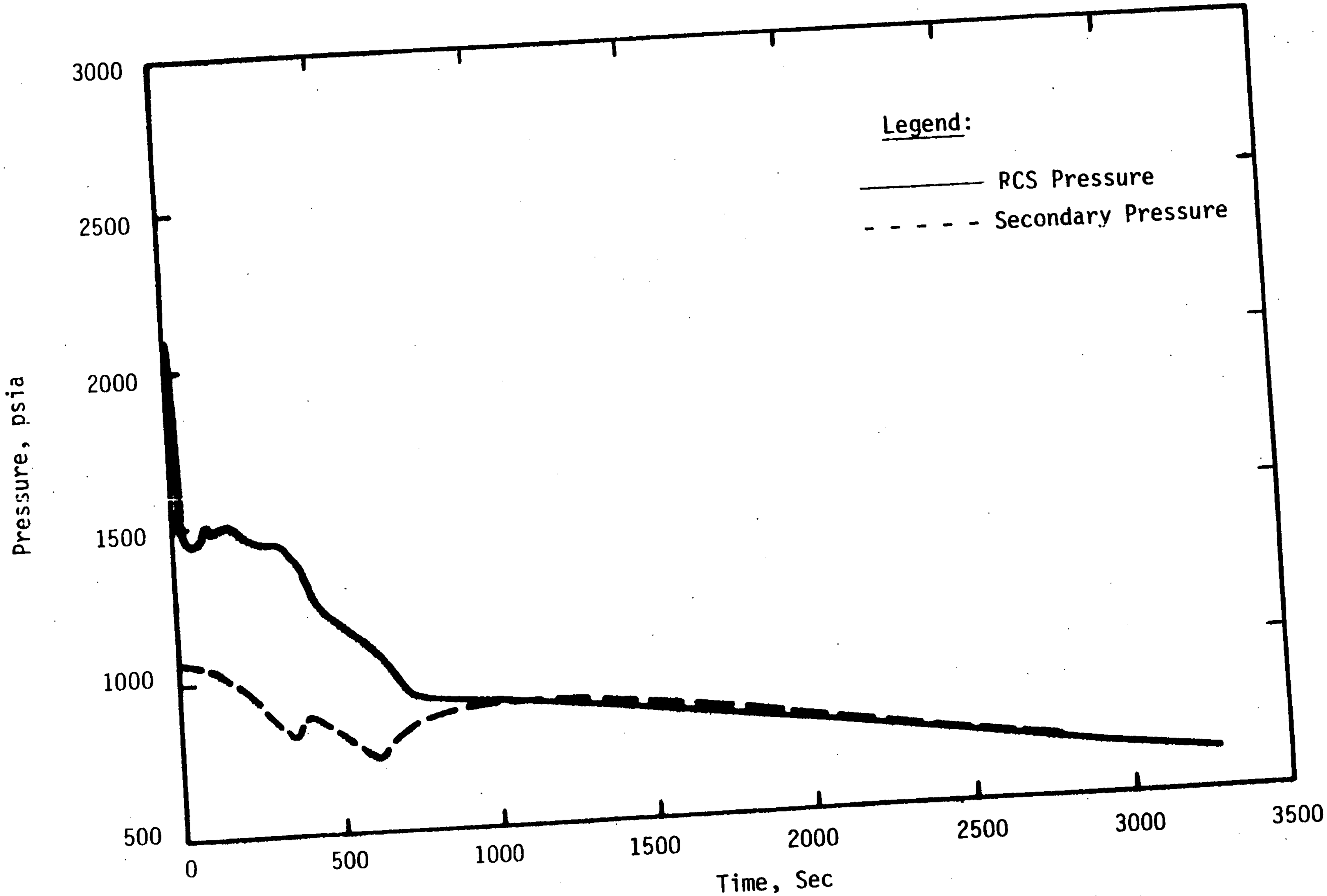


Figure 7-11 - Hot Leg and Reactor Vessel Mixture Heights, .04 $Ft^2$ Break at RC Pump Discharge, Revised Model

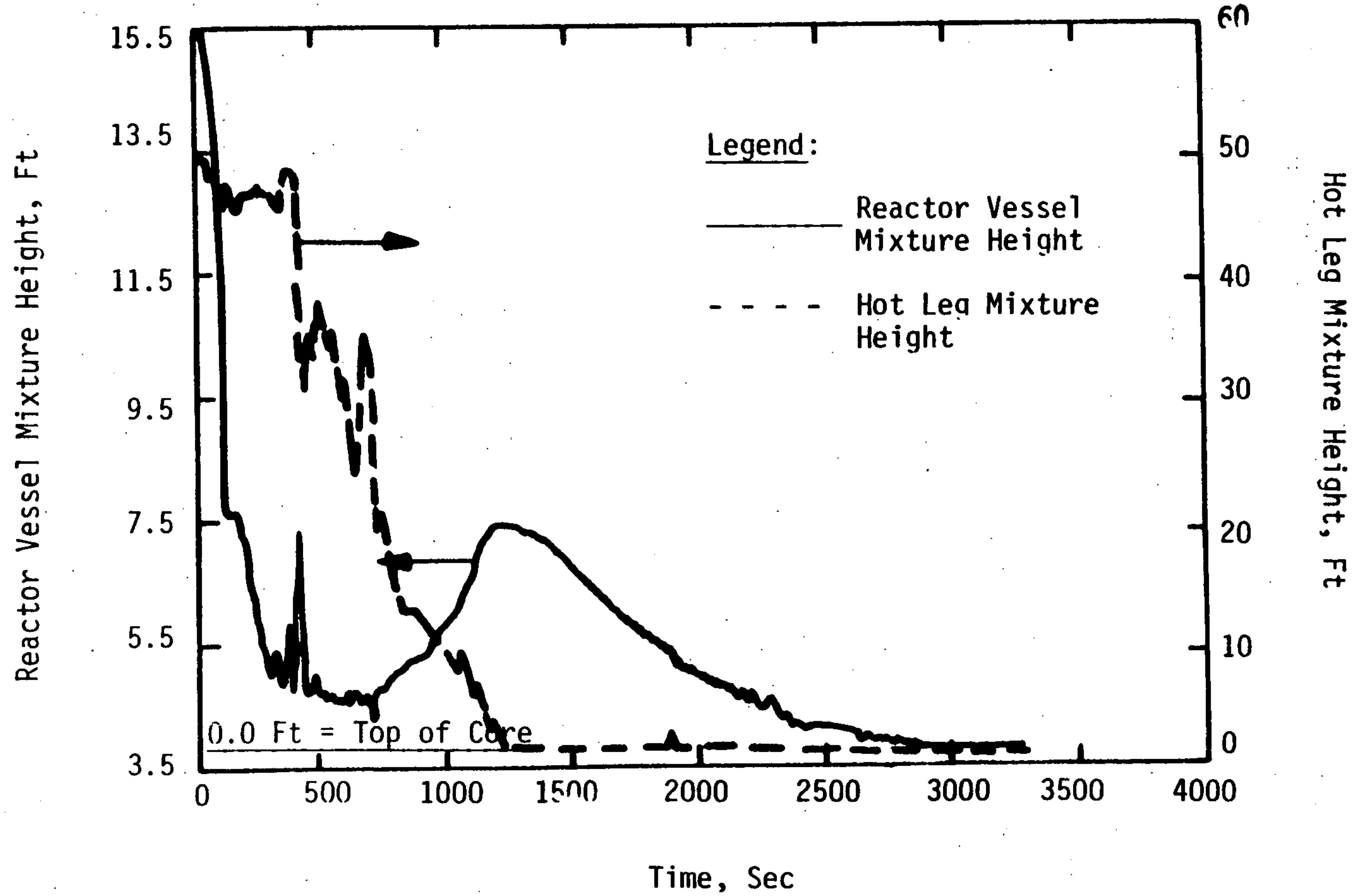


Figure 7-12 - RCS Pressure, .04 Ft$^2$ Break at RC Pump Discharge
Comparison of Results of Previous and Revised
Models

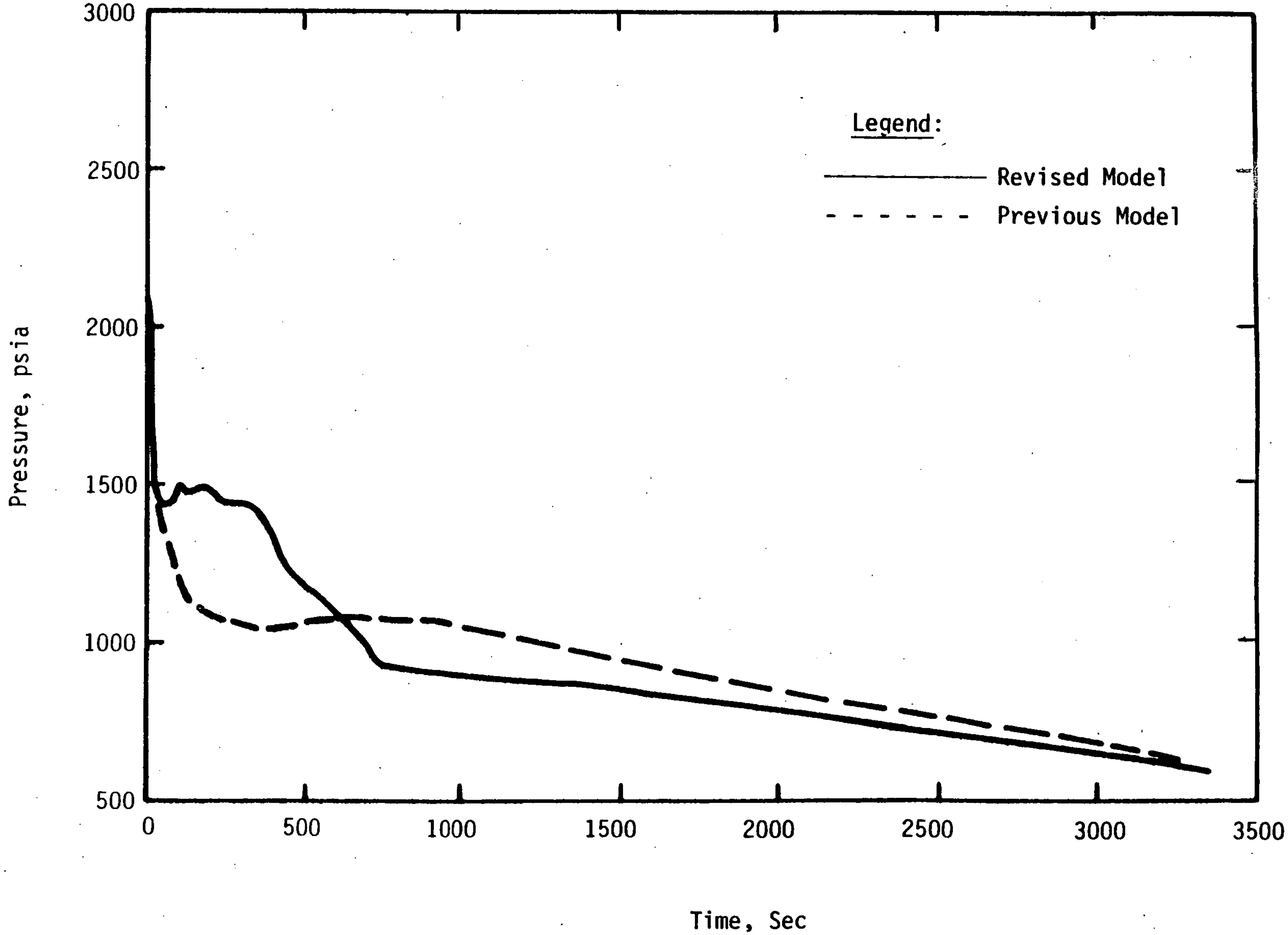


Figure 7-13 - Inner Vessel Mixture Heights, .04 $Ft^2$ Break at RC Pump Discharge, Comparison of Results of Previous and Revised Models

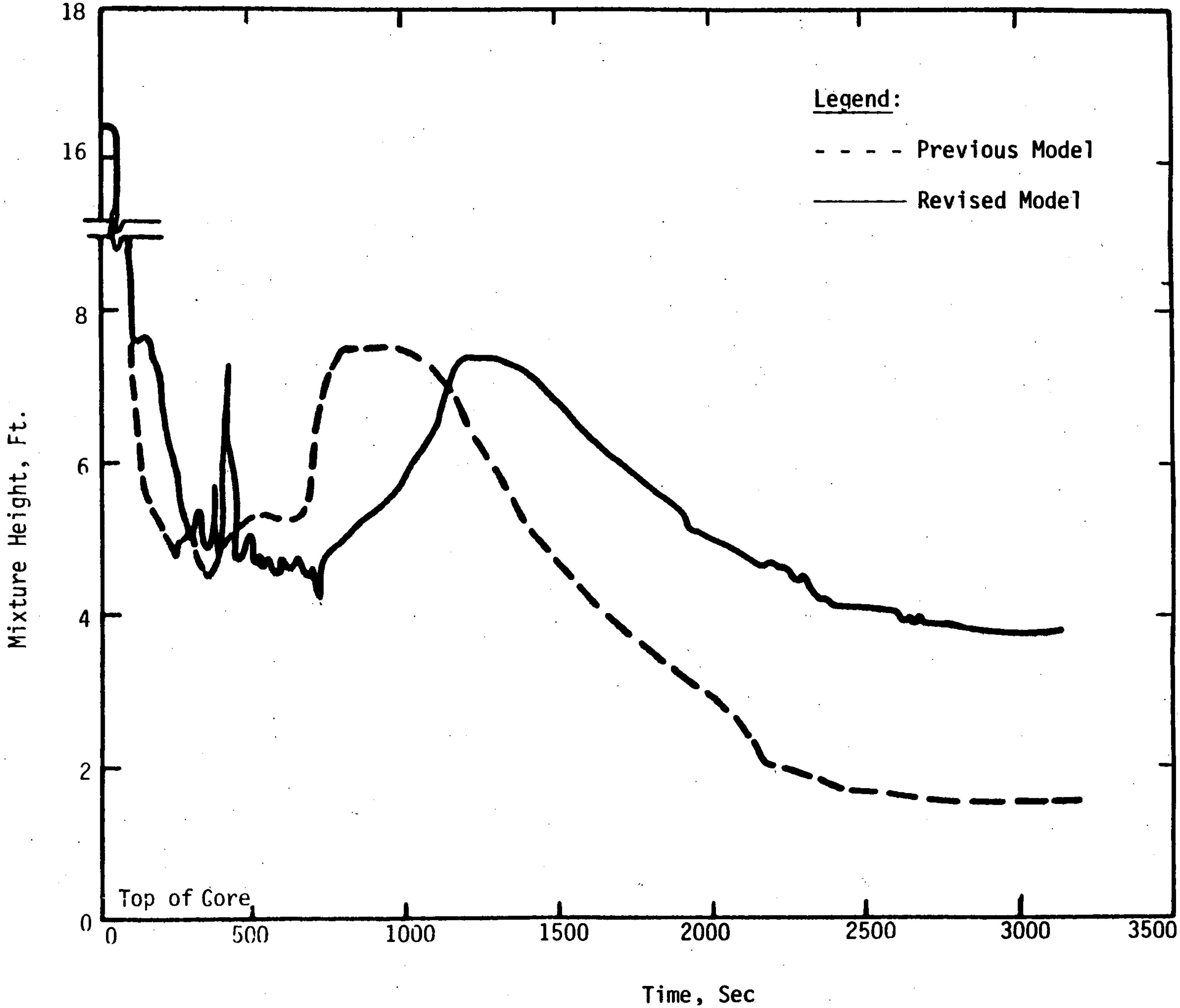


Figure 7-14 - Secondary Pressures, .04 $Ft^2$ Break at RC Pump Discharge, Comparison of Results of Previous and Revised Models

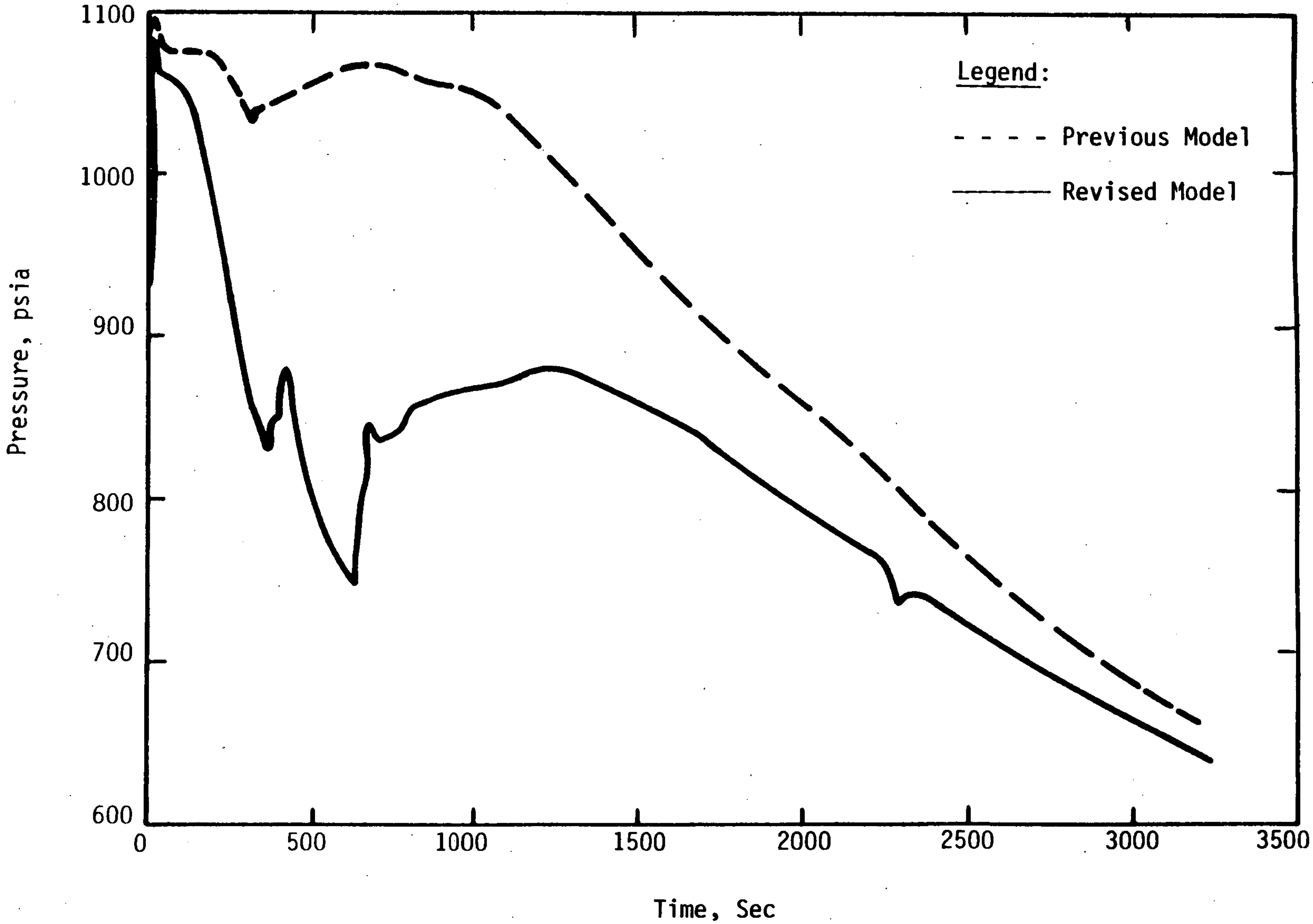
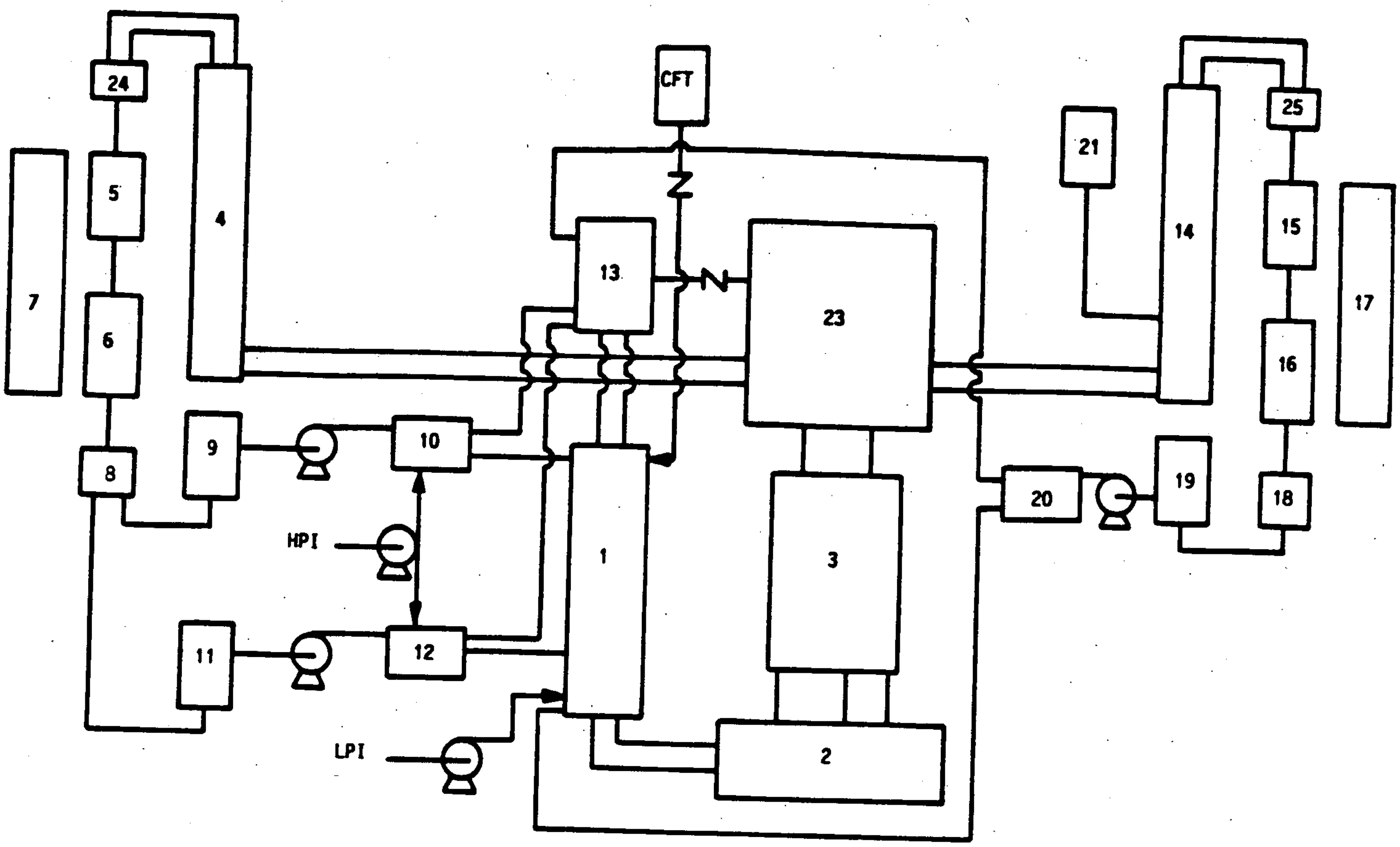


Figure 7-15 - Previ[illegible] EM Noding Diagram for Categor[illegible] SBLOCA's (.005 $ft^2$ < A < .02 $Ft^2$)




| Node No. | Identification |
|---|---|
| 1,13 | Downcomer |
| 2 | Lower Plenum |
| 3 | Core |
| 4,14 | Hot Leg Piping |
| 5,6,15,16 | Steam Generator |
| 7,17 | Secondary, SG |
| 8,18 | SG Lower Head |
| 9,11,19 | Cold Leg Piping |
| 10,12,20 | Cold Leg Piping |
| 21 | Pressurizer |
| 22 | Containment |
| 23 | Upper Plenum |
| 24,25 | SG Upper Head/Top of HL |

Figure 7-16 - RCS and Secondary Pressure, .01 Ft$^2$ Break at RC Pump Discharge, Previous Model

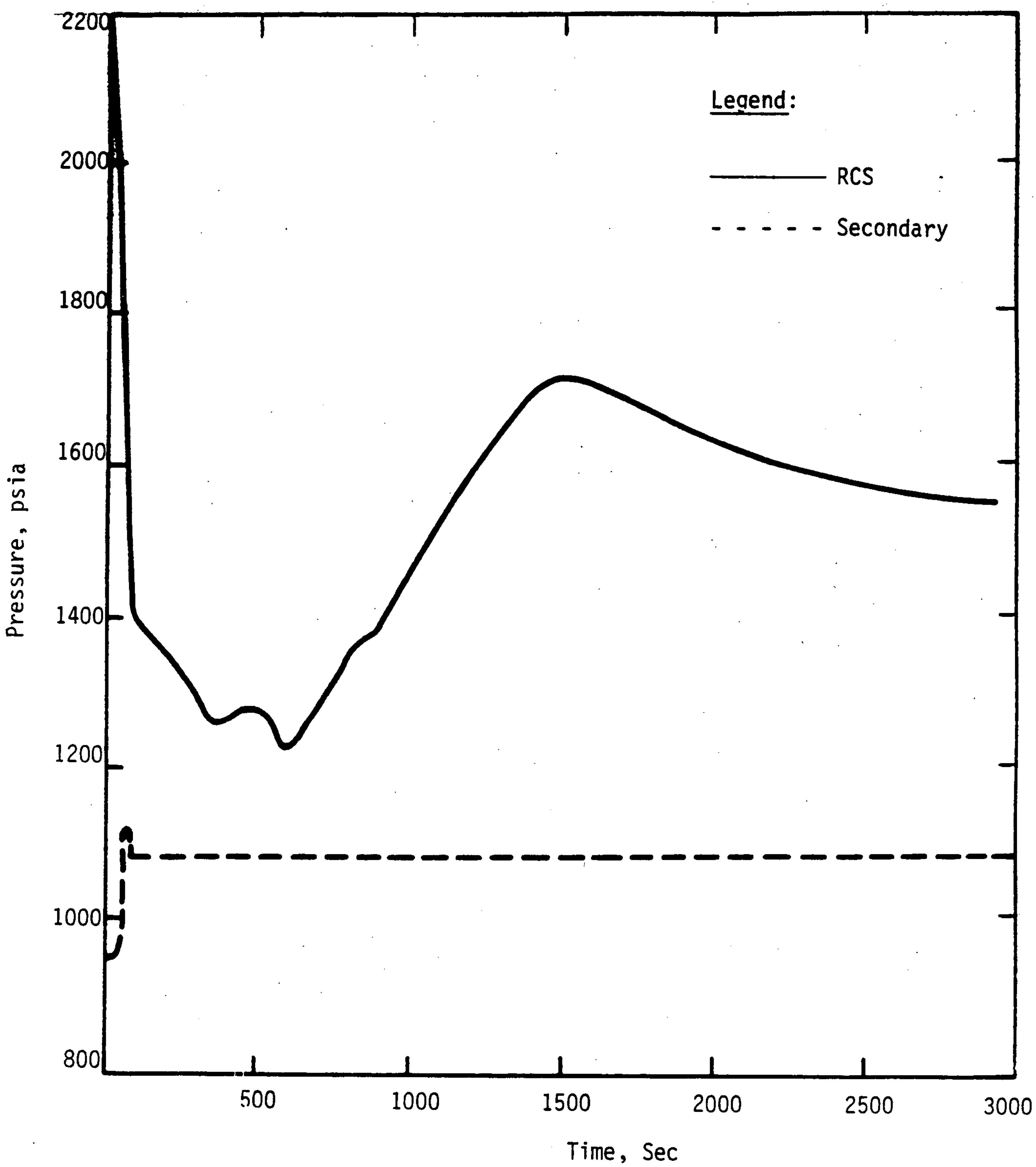


Figure 7-17 - Hot Leg Mixture Heights, .01 $Ft^2$ Break at RC Pump Discharge, Previous Model

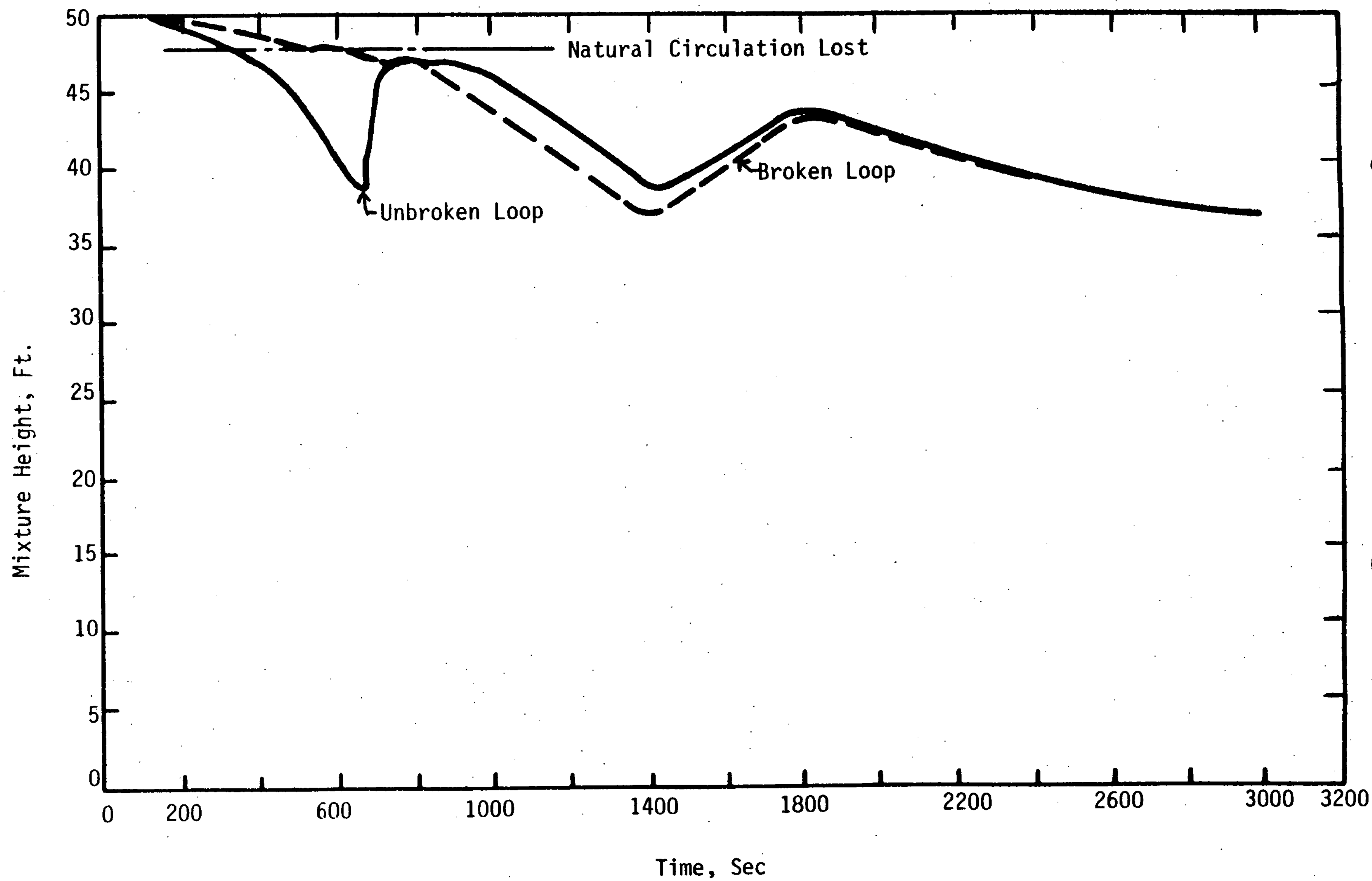


Figure 7-18 - RCS and Secondary Pressures, .01 $Ft^2$ Break at RC Pump Discharge, Revised Model

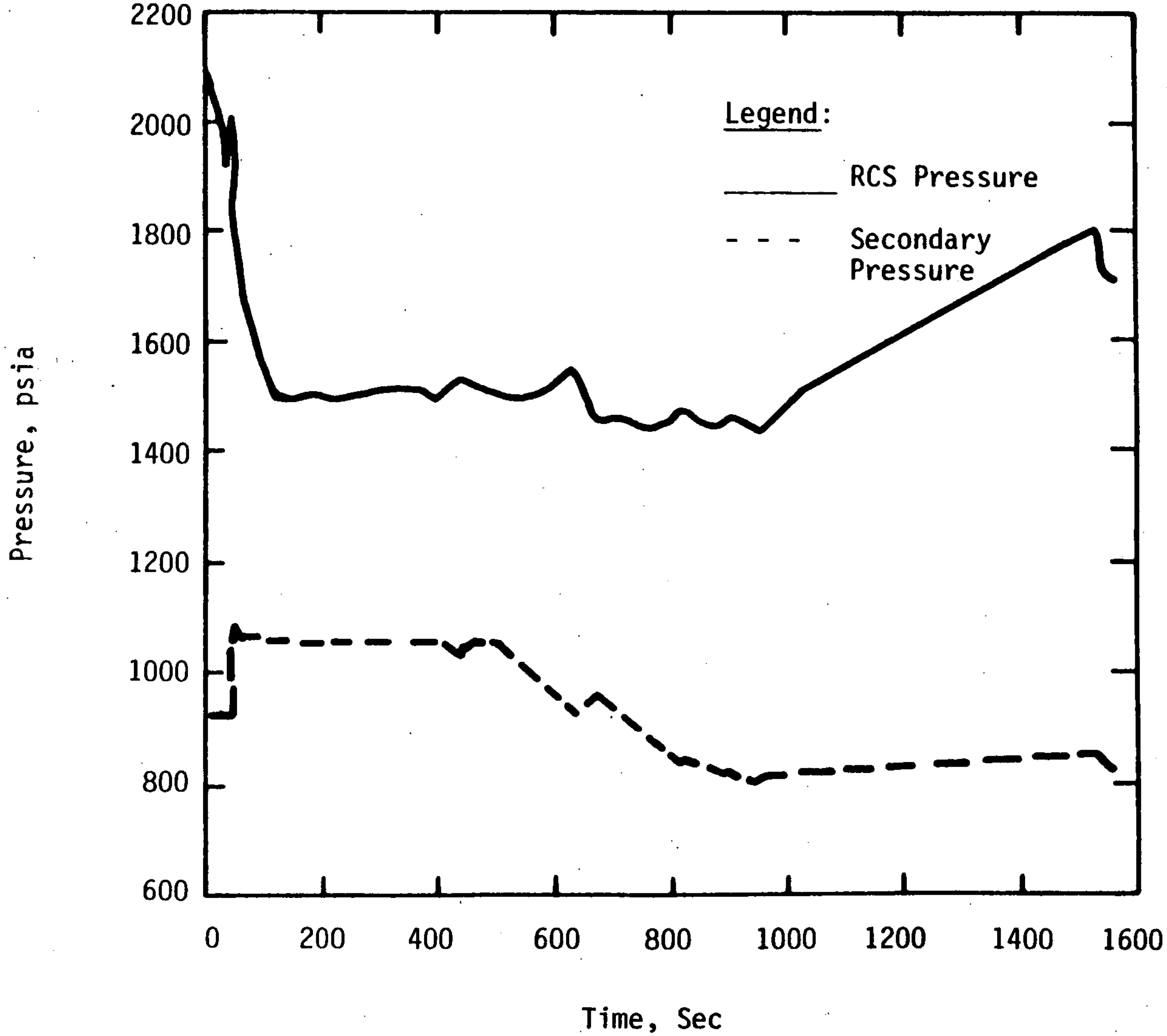


Figure 7-19 - Hot Leg Mixture Heights, .01 $Ft^2$ Break at RC Pump Discharge, Revised Model

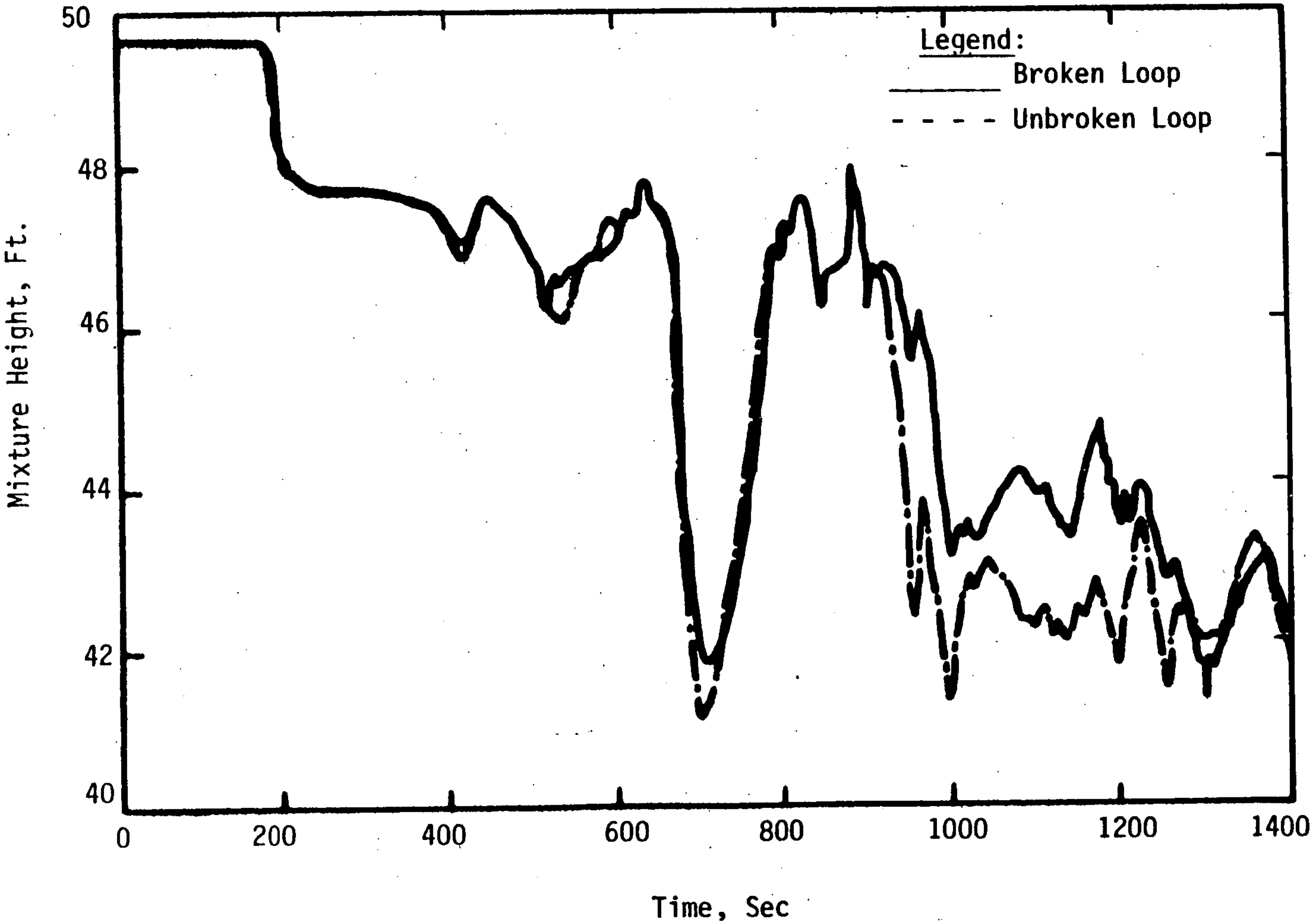


Figure 7-20 - RCS Pressure, .01 $Ft^2$ Break at RC Pump Discharge, Comparison of Results of Previous and Revised Models

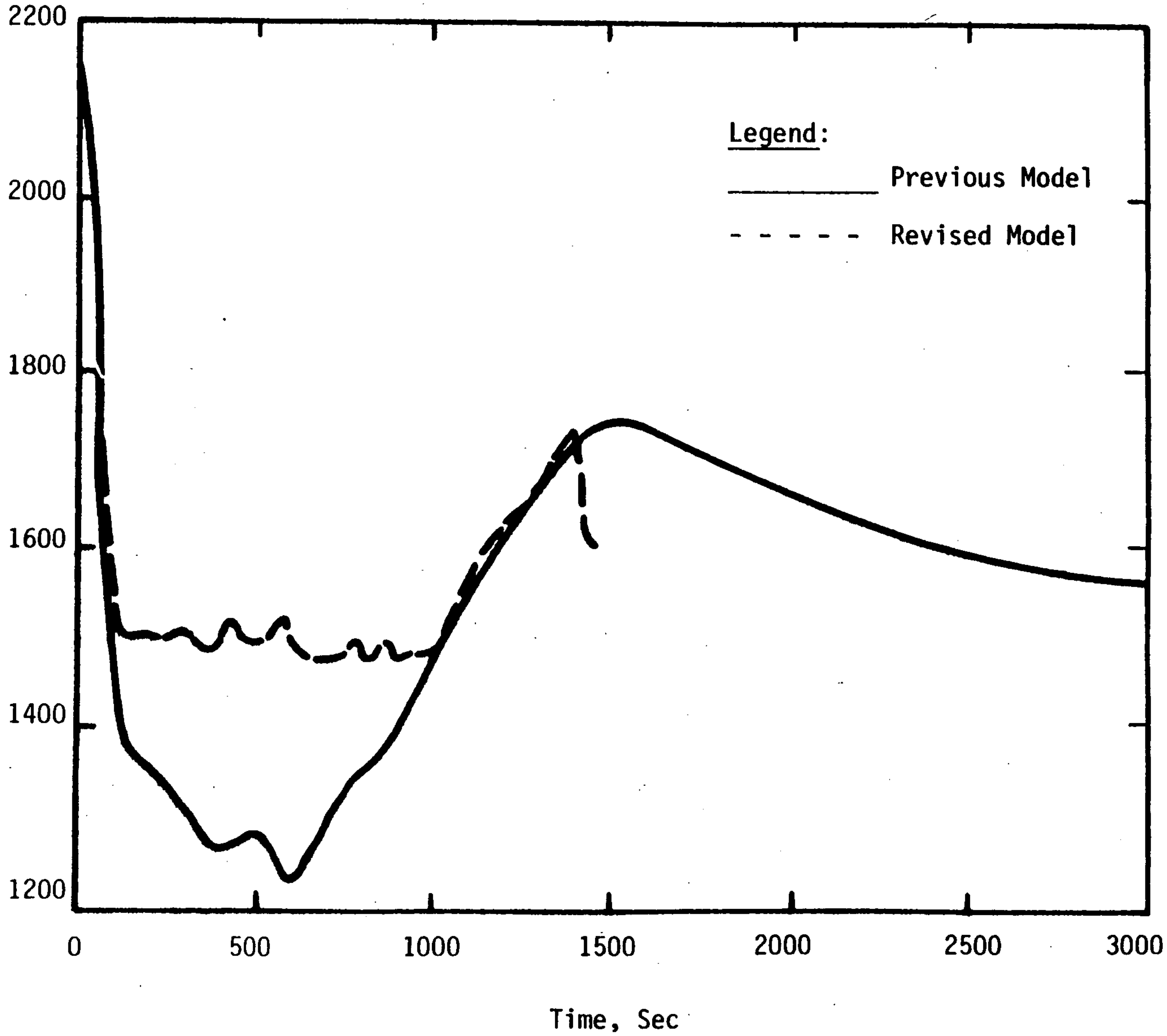


Figure 7-21 - Inner Vessel Mixture Heights, .01 $Ft^2$ Break at RC Pump Discharge, Comparison of Previous and Revised Models

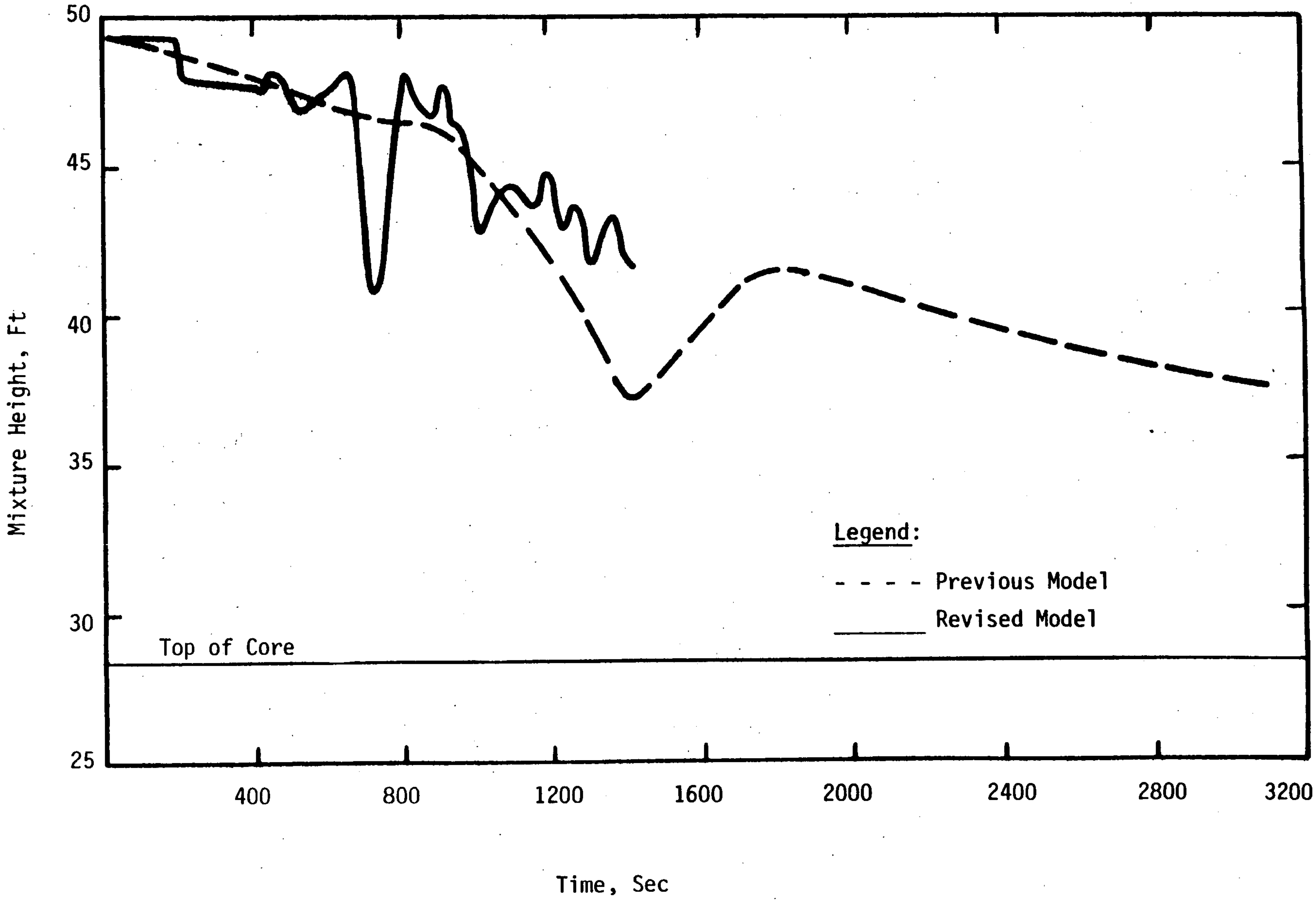
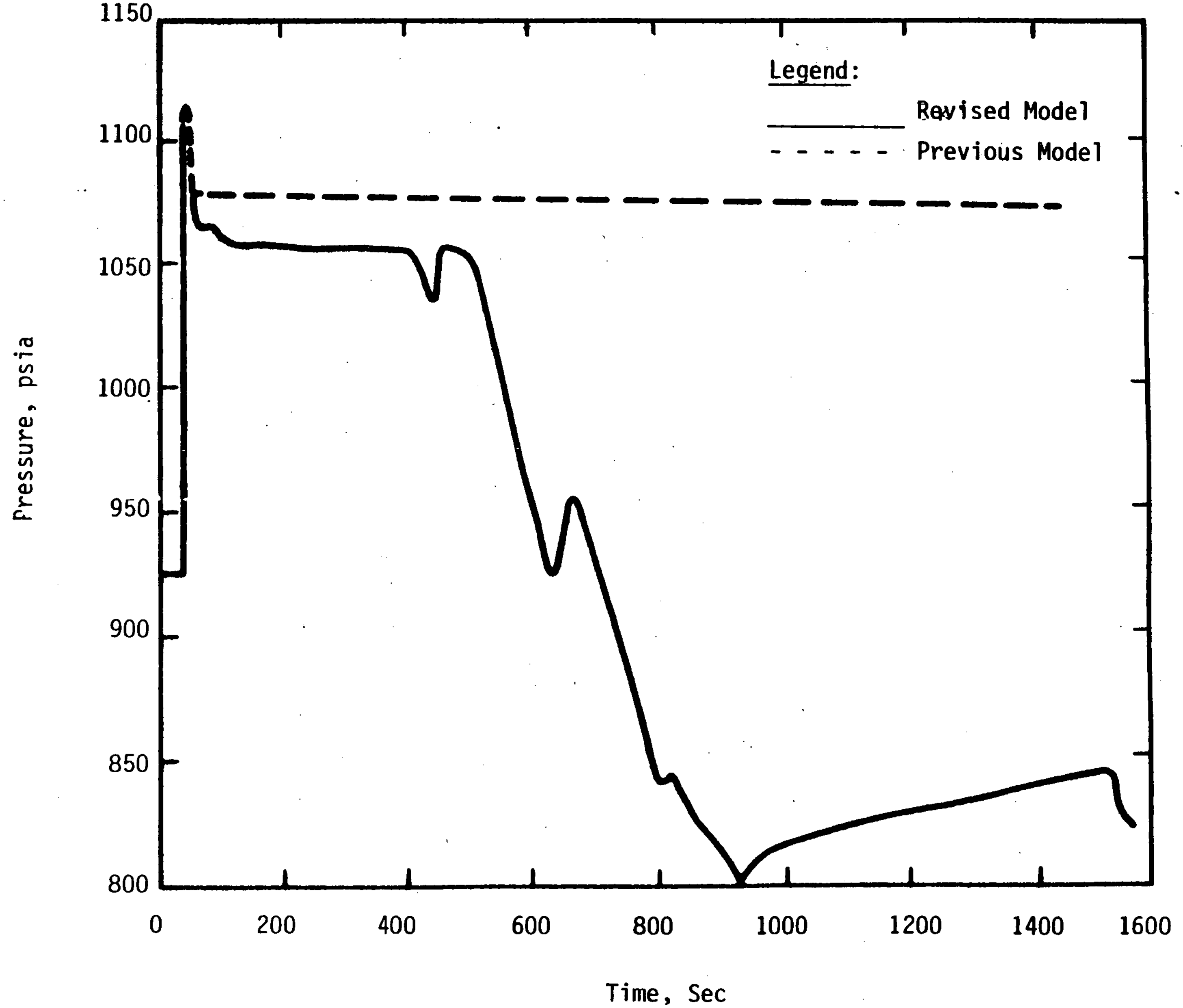


Figure 7-22 - Secondary Pressures, .01 $Ft^2$ Break at RC Pump Discharge, Comparison of Results of Previous and Revised Models



## 8.0 CONCLUSIONS

Three RC pump discharge breaks were reanalyzed with the revised EM and using design-basis assumption. The breaks reanalyzed were of the following sizes:

1. 0.01 $ft^2$ (Category 2).
2. 0.04 $ft^2$ (Category 2-3 transition).
3. 0.07 $ft^2$ (Category 3).

In each case, comparisons of the analytical results from the previous and revised models showed that the transients exhibited the same general trends and phenomena, although differing somewhat in timing and magnitudes. Significant conclusions for each beak size analyzed are as follows.

1. For the 0.01 $ft^2$ break, in both analyses, the same major responses occurred:
   a. An initial RCS depressurization to saturation.
   b. A saturated RCS repressurization.
   c. An RCS depressurization when primary steam condensation began in the SG tubes.
2. For the 0.04 ft break:
   a. The revised EMs SG model predicted a lower heat transfer rate than did the previous model early in the event.
   b. In the new case, the relatively lower early SG heat transfer rate, combined with effects of SG noding and AFW modeling, resulted in a comparably large primary-to-secondary differential temperature ( T) developing by approximately 115 seconds.
   c. This large T resulted in a higher rate of b-c heat transfer in the new case, and also allowed a longer duration for b-c.

d. As a result of the enhanced b-c cooling, pressure in the new case fell below that in the old case during the middle and late portions of the transient. For the revised case, this lower pressure resulted in:

(1) Increased HPI flow.

(2) Decreased leak flow.

(3) A resultant larger liquid volume remaining in the RCS by the time long-term cooling was established.

3. For the 0.07 $ft^2$ break:

a. The RCS pressure response in the revised analysis was similar to the pressure response for the revised 0.04 $ft^2$ break, in that

(1) Pressure was higher early in the event.

(2) Once b-c cooling was established, a higher primary-to-secondary T resulted in a larger rate of heat transfer over a longer period of time.

(3) The enhanced b-c cooling caused RCS pressure to become lower and remain lower in the revised case.

b. As in the 0.04 $ft^2$ case, lower RCS pressure late in the transient resulted in

(1) More liquid remaining in the RCS.

(2) CFT injection commencing earlier than in the previous case and, more importantly, beginning while the core was still covered.

c. The combination of lower pressure, greater HPI flow, CFT injection and lower break flow allowed the core to remain covered in the new case. In comparison, the previous 0.07 $ft^2$ break case resulted in the most severe amount of core uncovering of any design basis SBLOCA case examined by B&W.

Overall, while results of studies with the previous and revised EMs are fundamentally similar, those of the newer model tended to be <u>less severe</u> for

the liquid inventory loss from the primary system and the potential for core uncovering. These results confirm that studies performed with the previous model are conservative and remain valid for continued licensing applications. For B&W's 177-FA lowered-loop plants, a documented SBLOCA spectrum is therefore considered to exist that meets all NRC requirements germane to SBLOCAs, as specified in 10 CFR 50 Appendix K, 10 CFR 50.46, NUREG-0565, and NUREG-0737.

## 9. REFERENCES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2. J. Cudlin, et. al., BAW-10092A, Rev. 3, "CRAFT2 Fortran Program for Digital Simulation of a Multinode Reactor During Loss of Coolant", Babcock & Wilcox, Lynchburg, Virginia, July, 1985.

3. B&W Report, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks In The 177 Fuel Assembly Plant," May 7, 1979 via Letter from J. H. Taylor to R. J. Mattson (NRC) dated May 7, 1979, NRC Public Document Room, Accession #79051901C4.

4. J. H. Taylor (B&W) to S. A. Varga (NRC), Letter, 7/18/78. Subject: "B&W's SBLOCA Spectrum Analysis".

5. R. C. Jones, et. al., BAW 10103, Rev. 3, Appendix C, "ECCS Analysis of B&W's 177FA Lowered Loop NSS," Babcock & Wilcox, Lynchburg, Virginia, July, 1977.

6. J. H. Taylor (B&W) to Dr. Ernst Volgenau (NRC), Letter, 4/14/78. Subject: Evaluation of 177 FA Lowered Loop ECCS Concern.

7. Report 77-1141270-00, "Evaluation of SBLOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W Designed Operating NSSS," Babcock and Wilcox, Lynchburg, Virginia, February 1983.

8. Report 77-1150445-00, "Evaluation of Minimum EFW Requirements Following Small Break LOCA," Babcock & Wilcox, Lynchburg, Virginia, May, 1984.