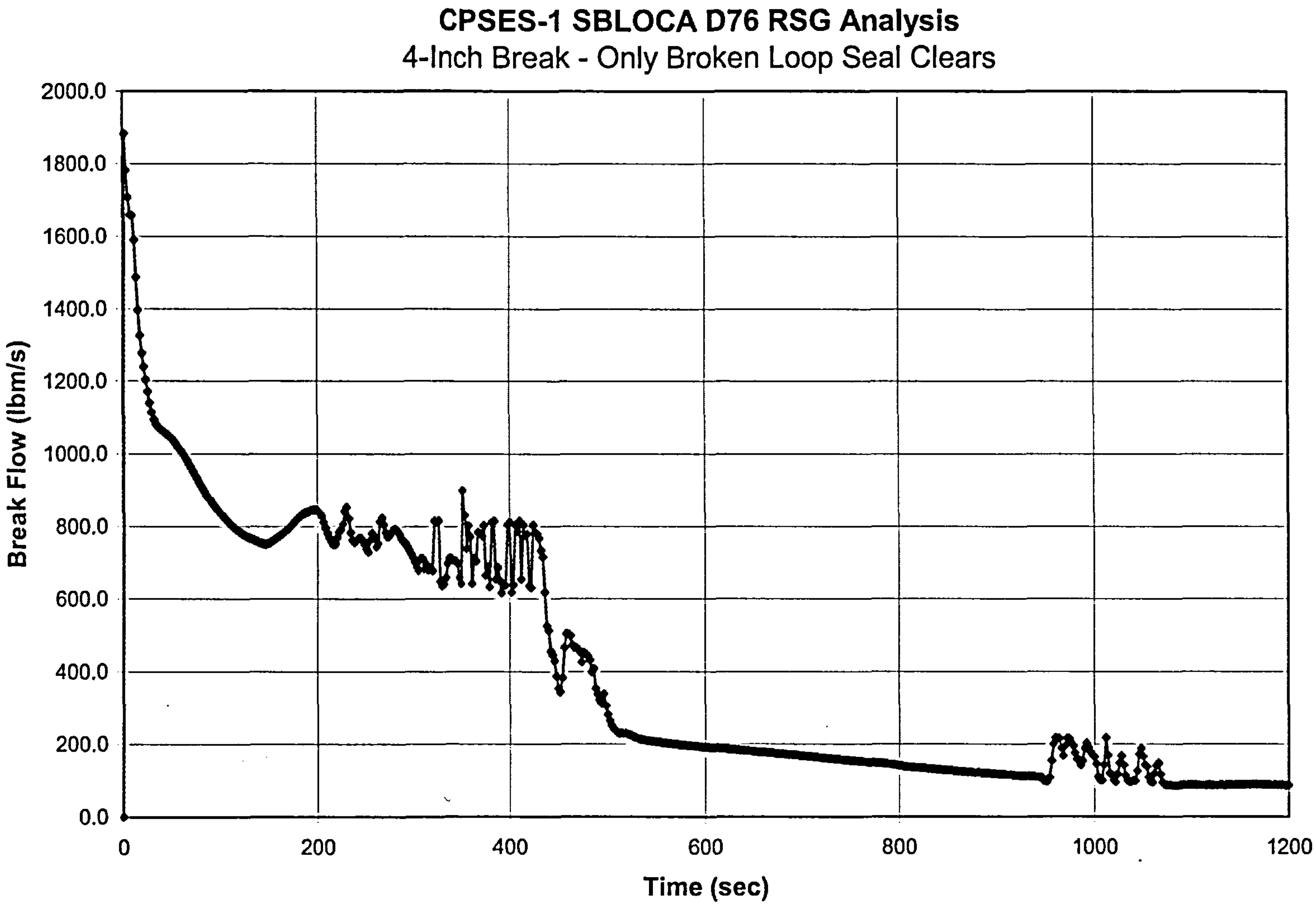

Figure H-10 Break Flow Rate – 4-in Break (Only Broken Loop Seal Clears)

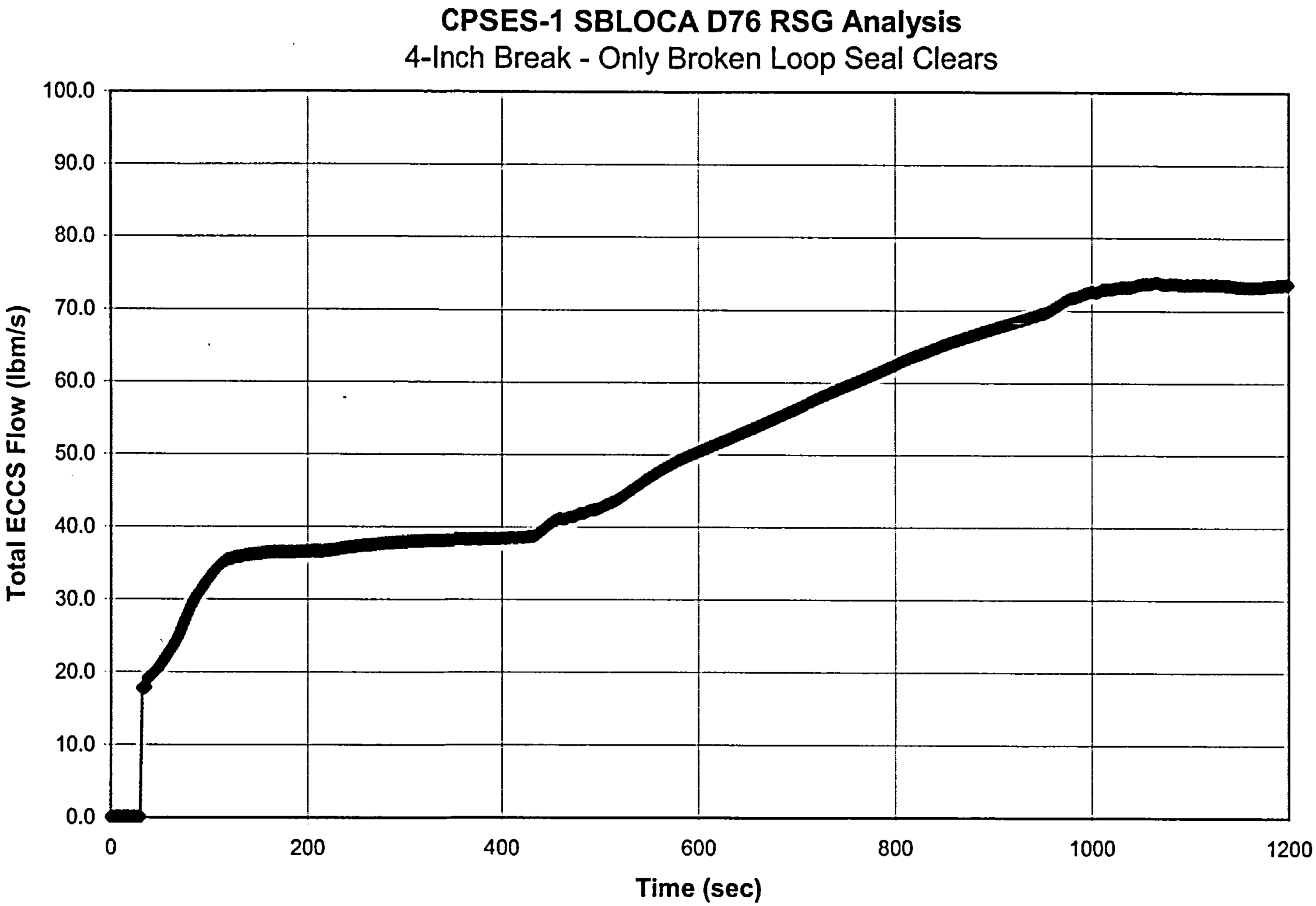

Figure H-11 Total Pumped ECCS Flow Rate – 4-in Break (Only Broken Loop Seal Clears)

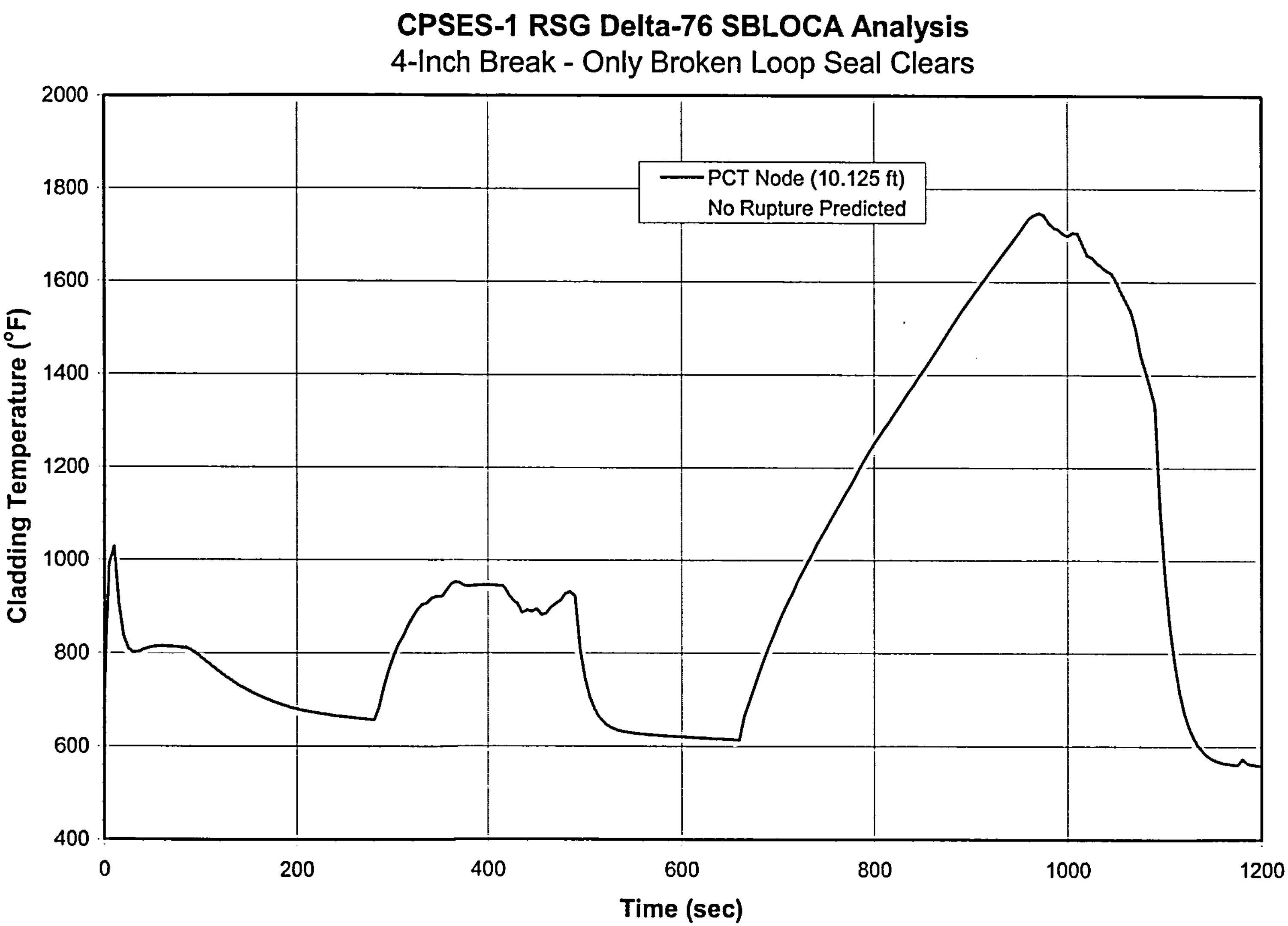

Figure H-12 TOODEE2 Clad Temperature – 4-in Break (Only Broken Loop Seal Clears)

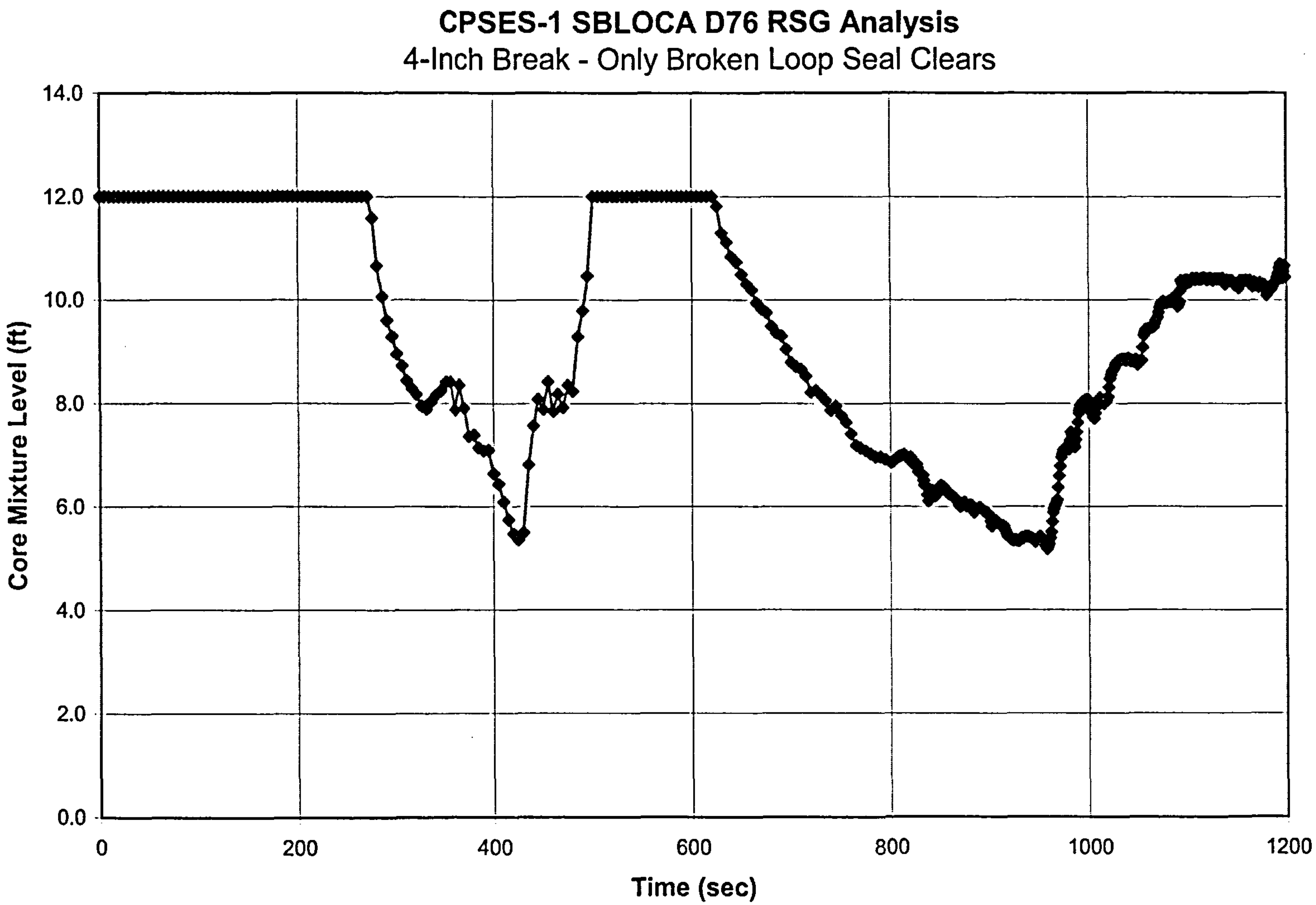

Figure H-13 Core Mixture Level – 4-in Break (Only Broken Loop Seal Clears)

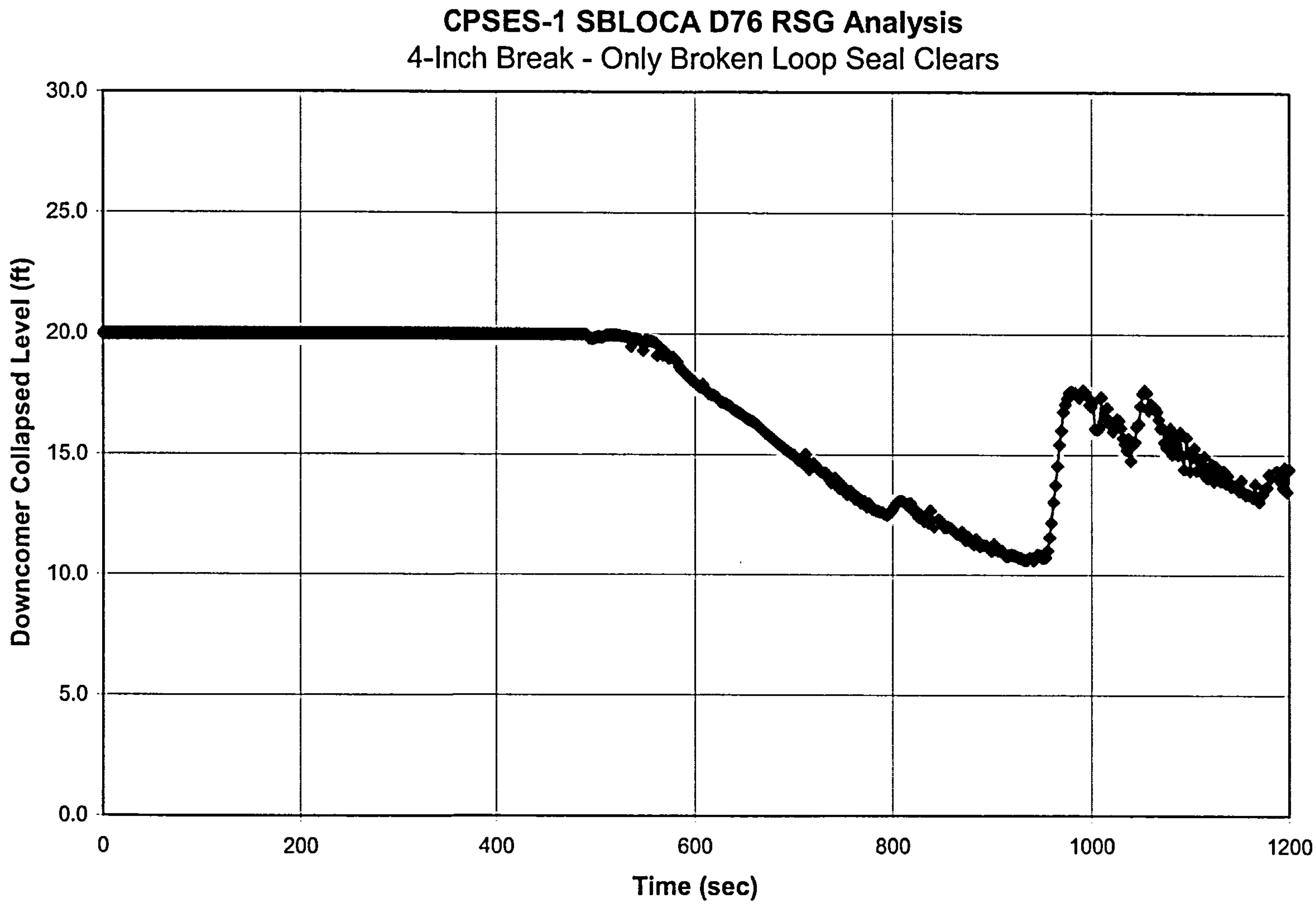

Figure H-14 Downcomer Liquid Level – 4-in Break (Only Broken Loop Seal Clears)

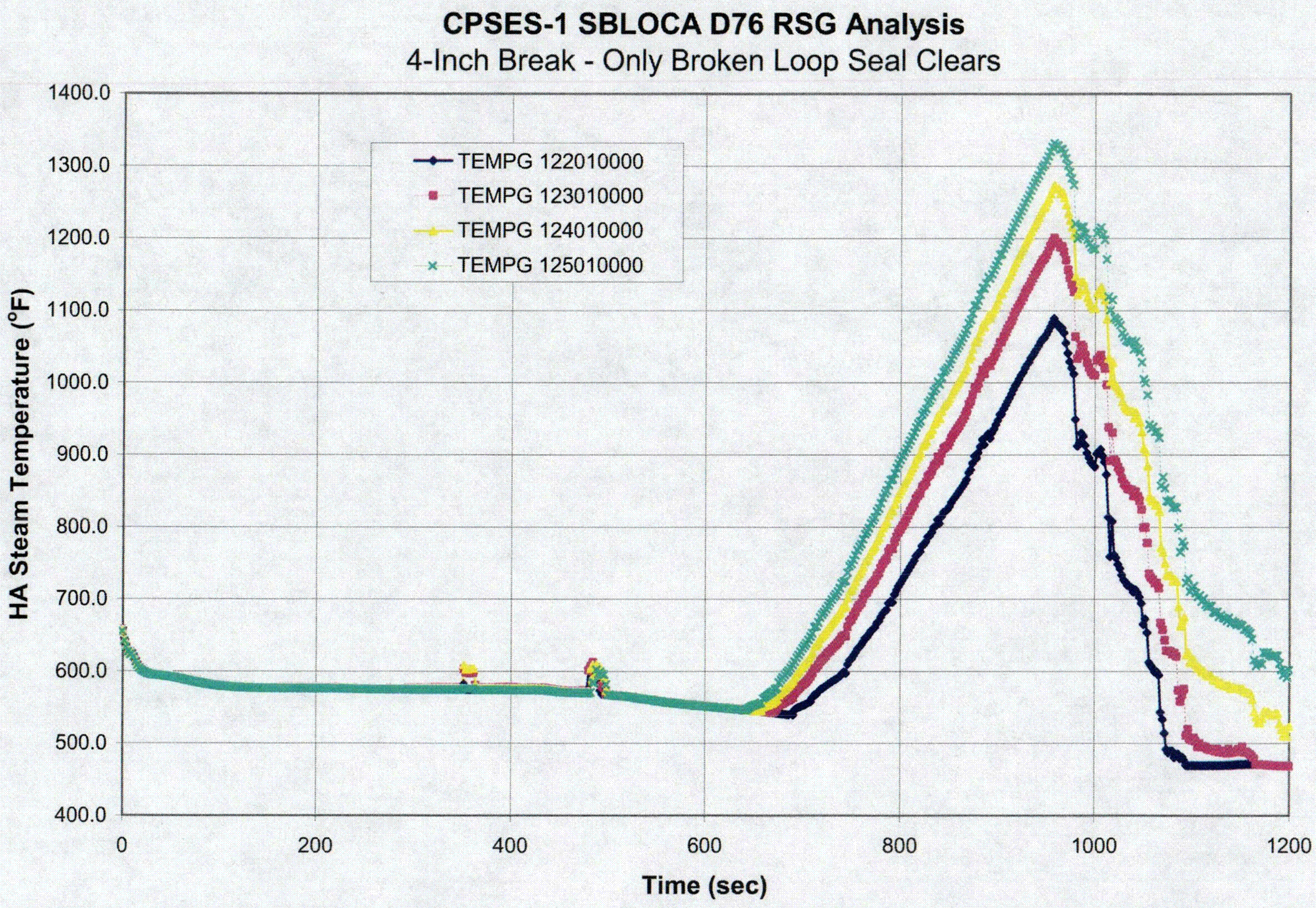

Figure H-15 Hot Assembly Steam Temperatures – 4-in Break (Only Broken Loop Seal Clears)

C86

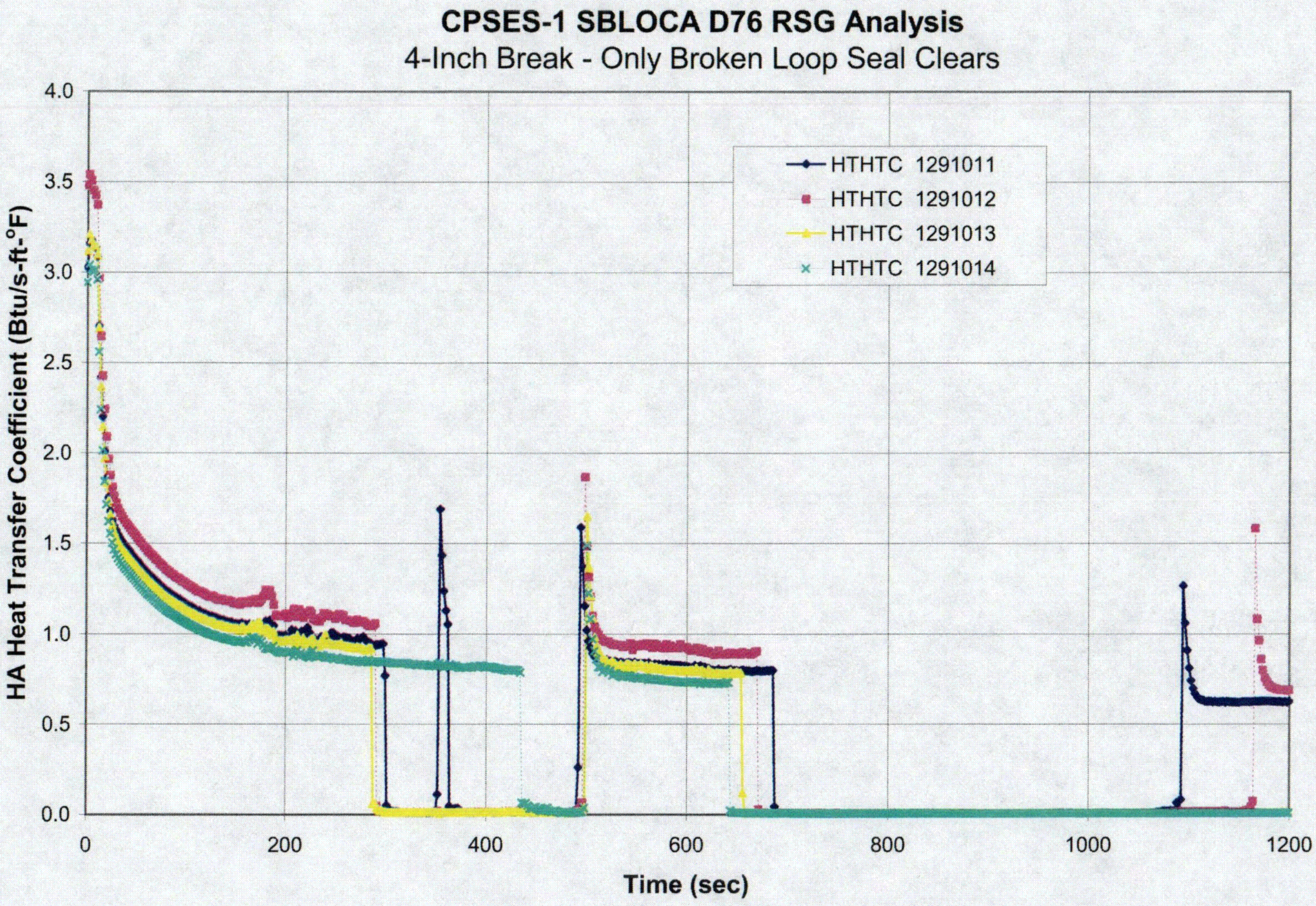

Figure H-16 Hot Assembly Heat Transfer Coefficients – 4-in Break (Only Broken Loop Seal Clears)

C97

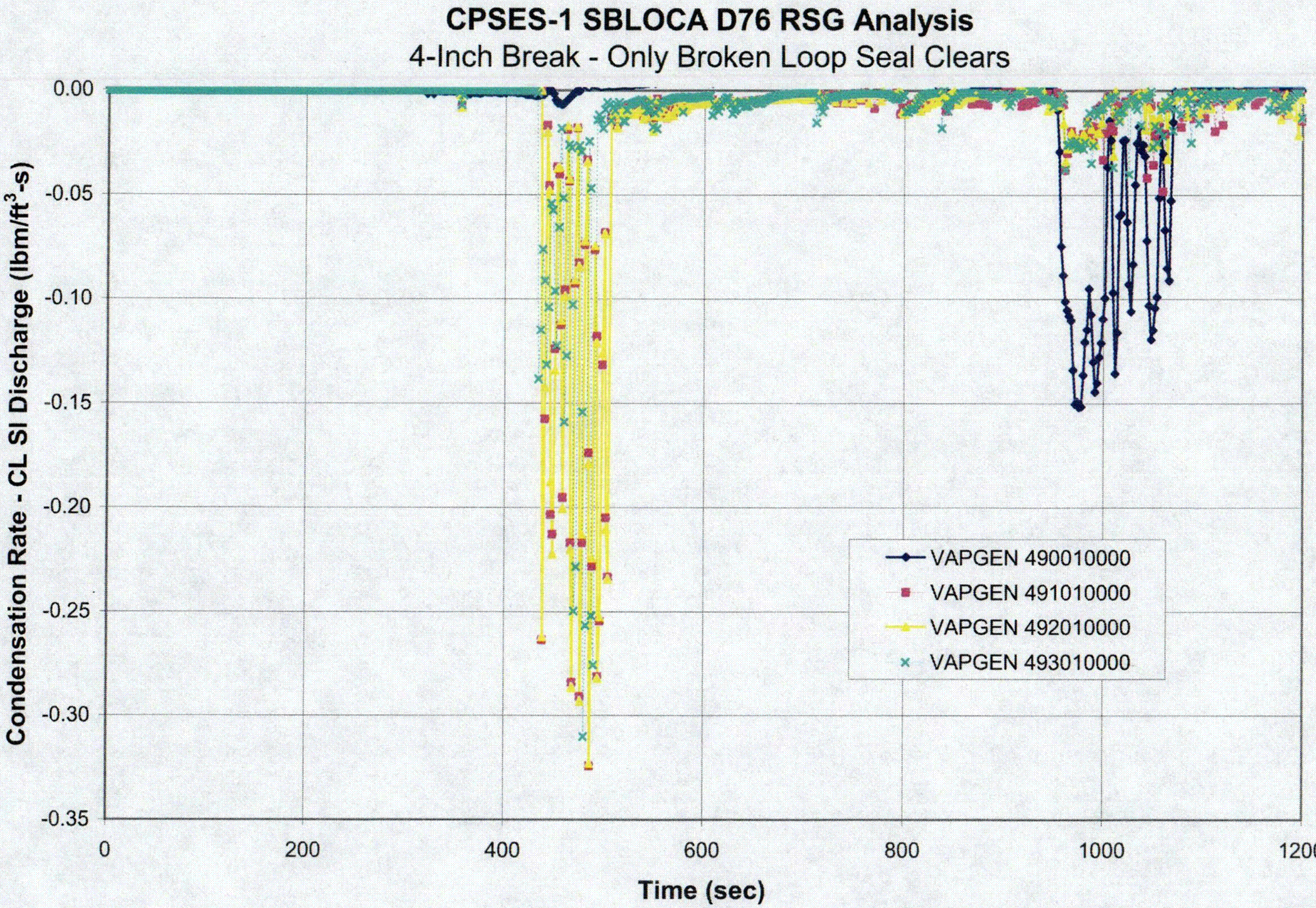

Figure H-17 Condensation Rate in Cold Leg Discharge – 4-in Break (Only Broken Loop Seal Clears)

C08

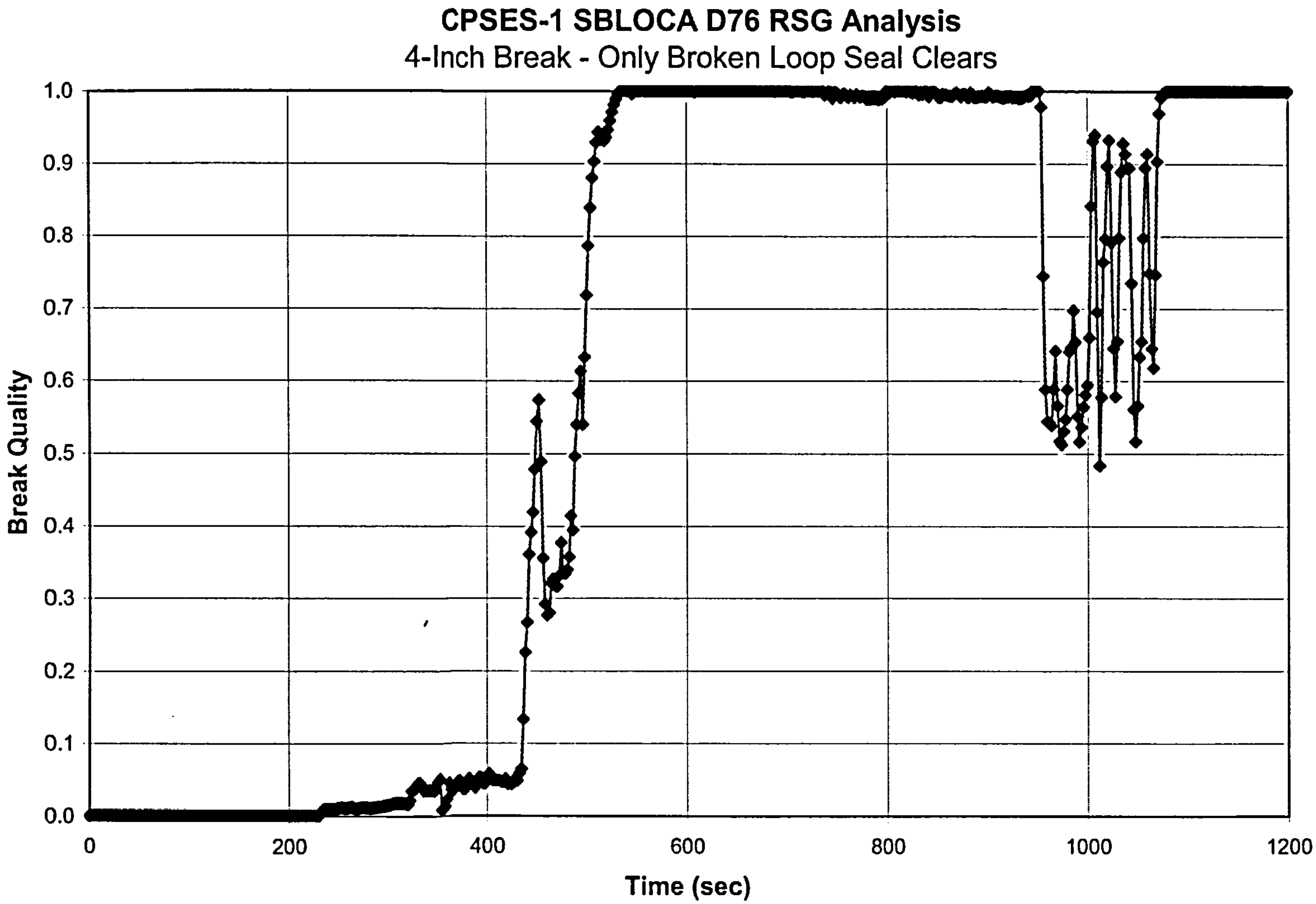

Figure H-18 Break Quality – 4-in Break (Only Broken Loop Seal Clears)

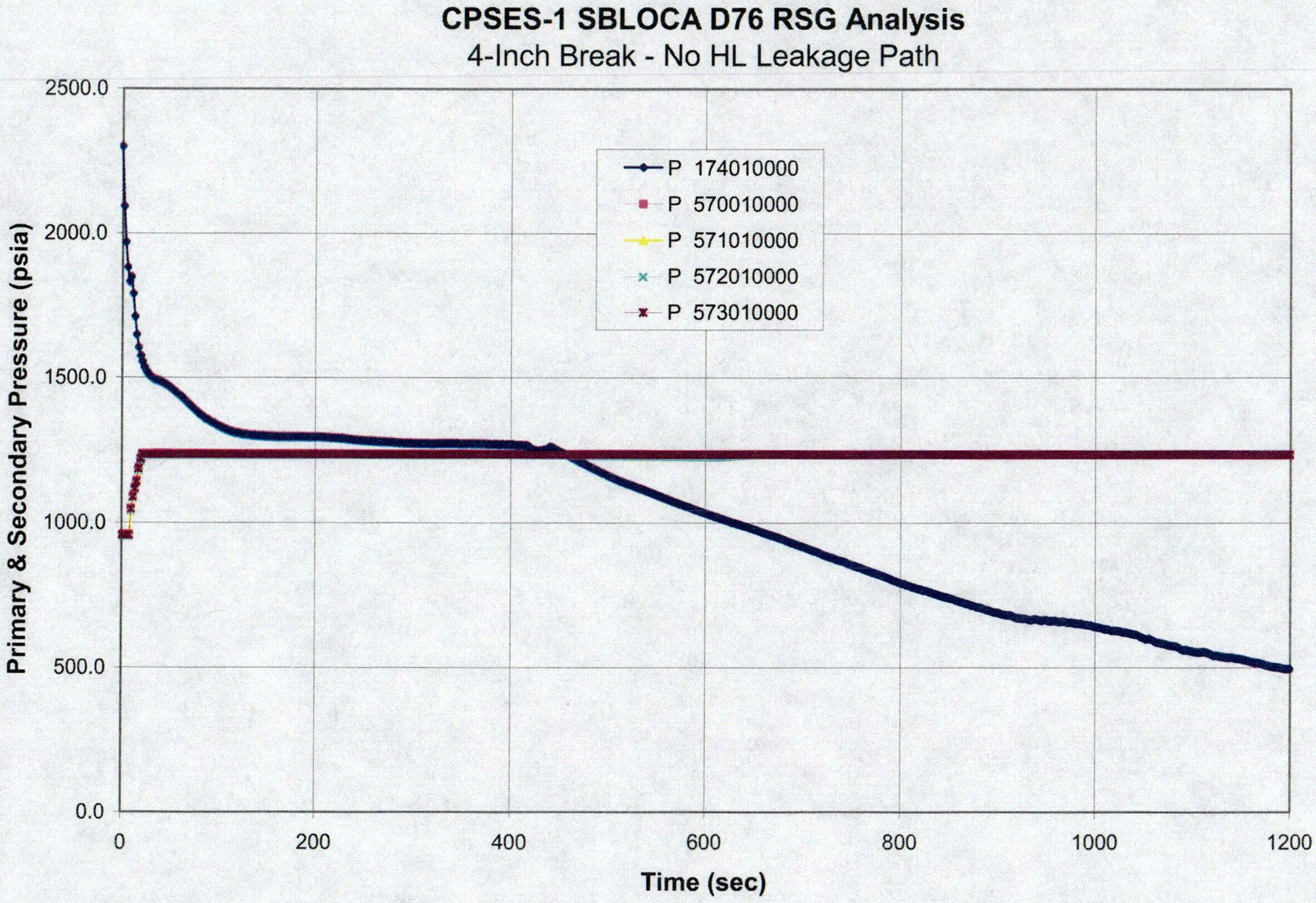

Figure I-1 Primary and Secondary System Pressures – 4-in Break (No HL Leakage Path)

C89

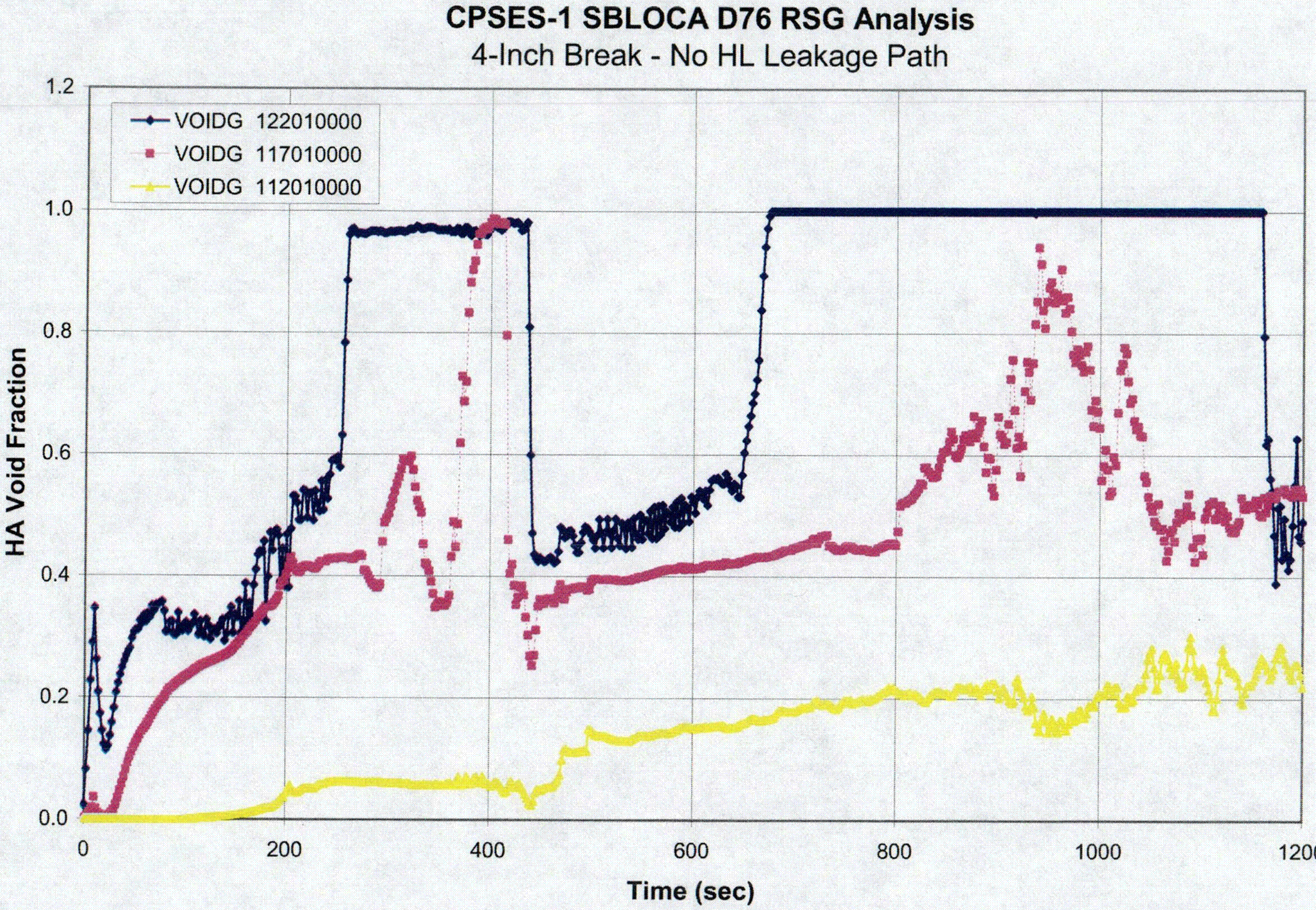

Figure I-2 Hot Assembly Region Void Fractions – 4-in Break (No HL Leakage Path)

C90

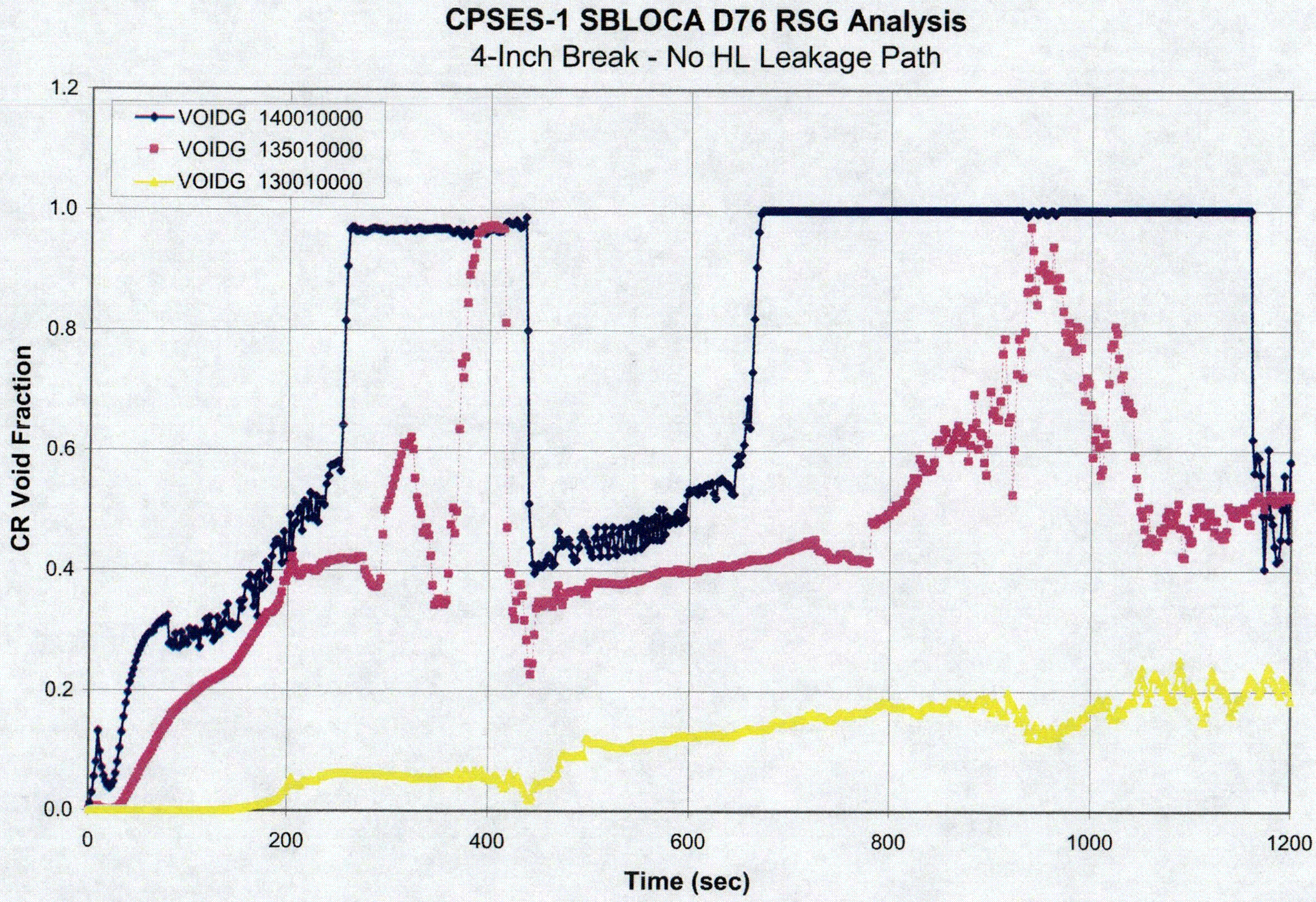

Figure I-3 Central Core Region Void Fractions – 4-in Break (No HL Leakage Path)

c91

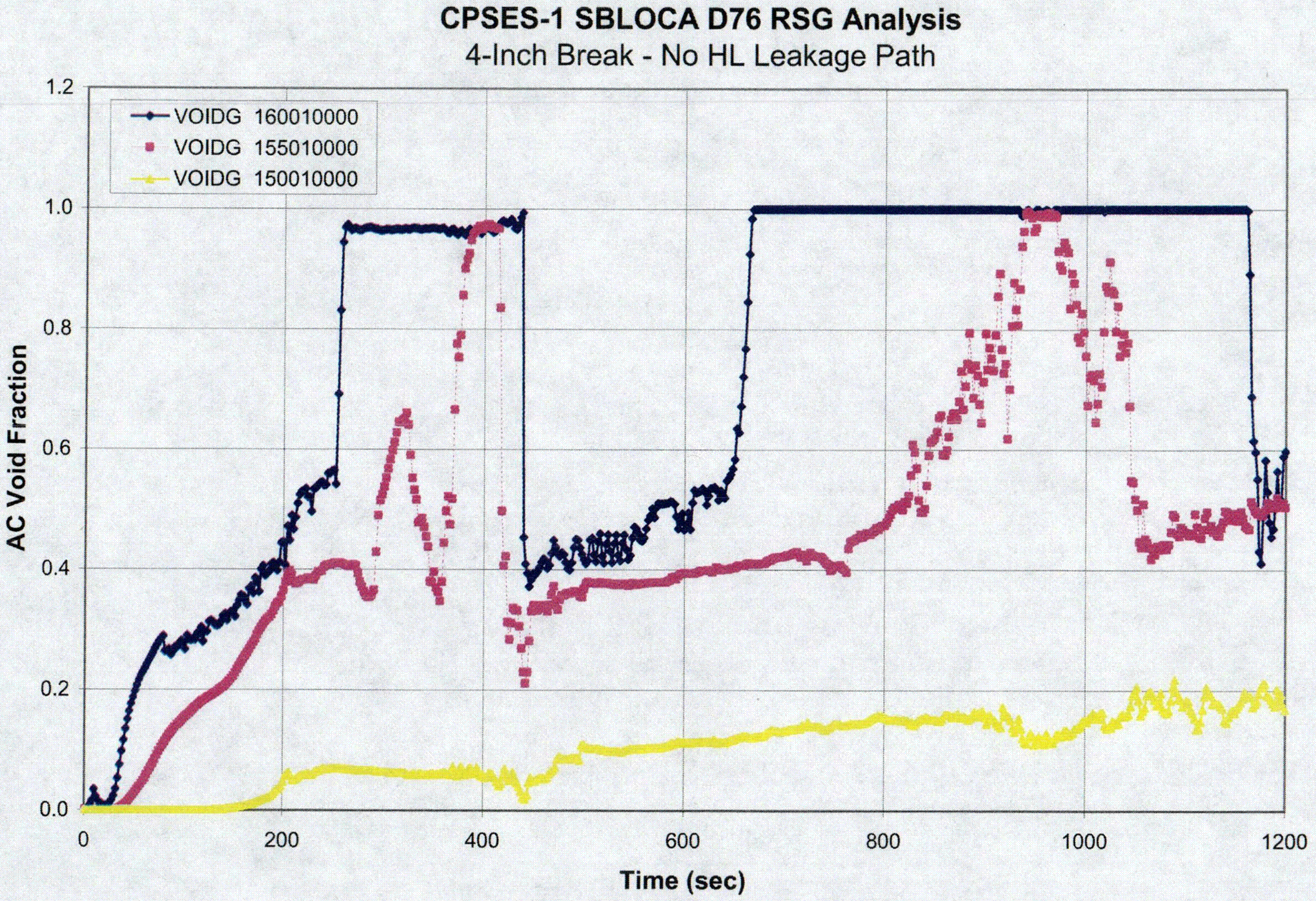

Figure I-4 Average Core Region Void Fractions – 4-in Break (No HL Leakage Path)

C92

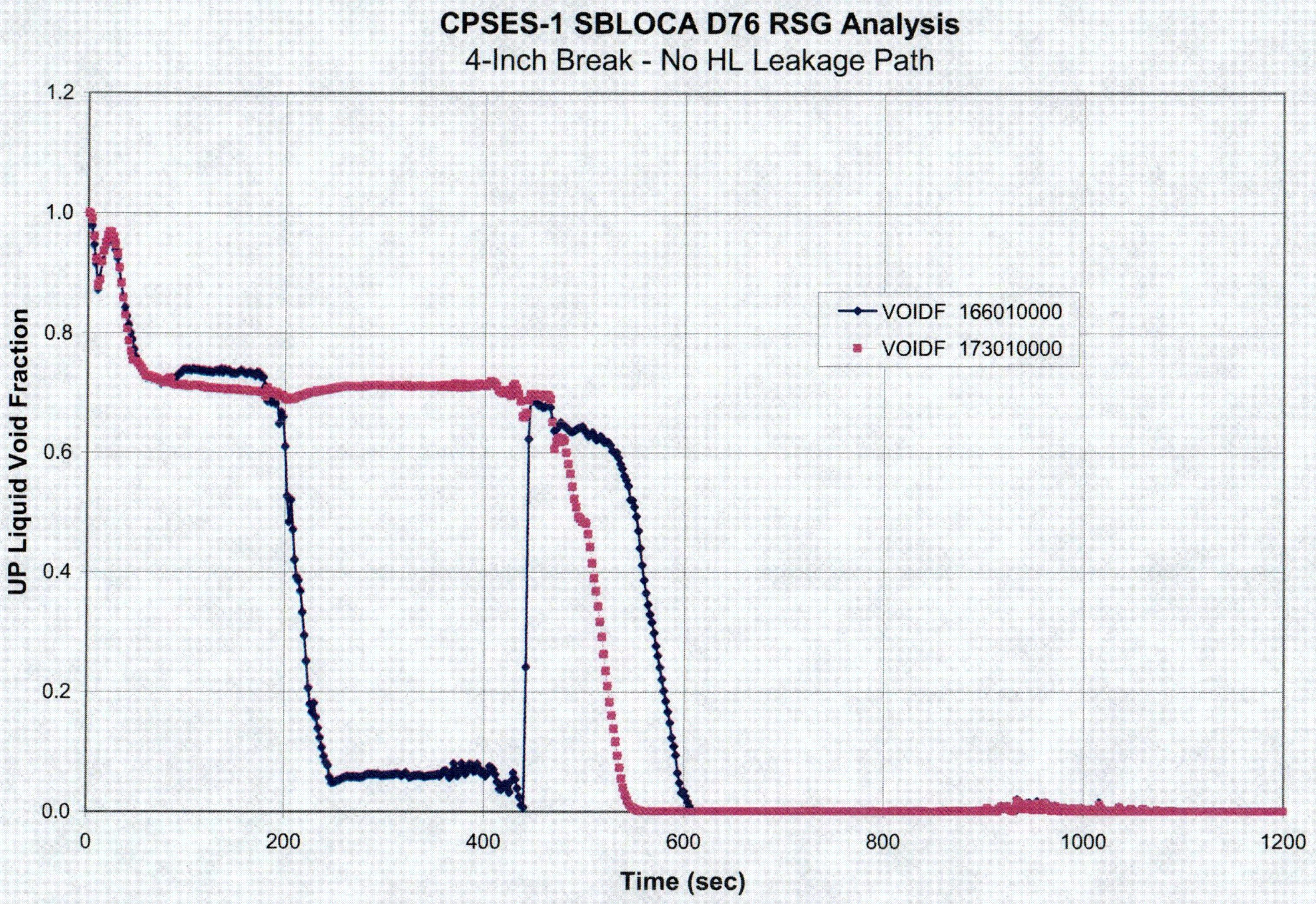

Figure I-5 Upper Plenum Liquid Fraction – 4-in Break (No HL Leakage Path)

C93

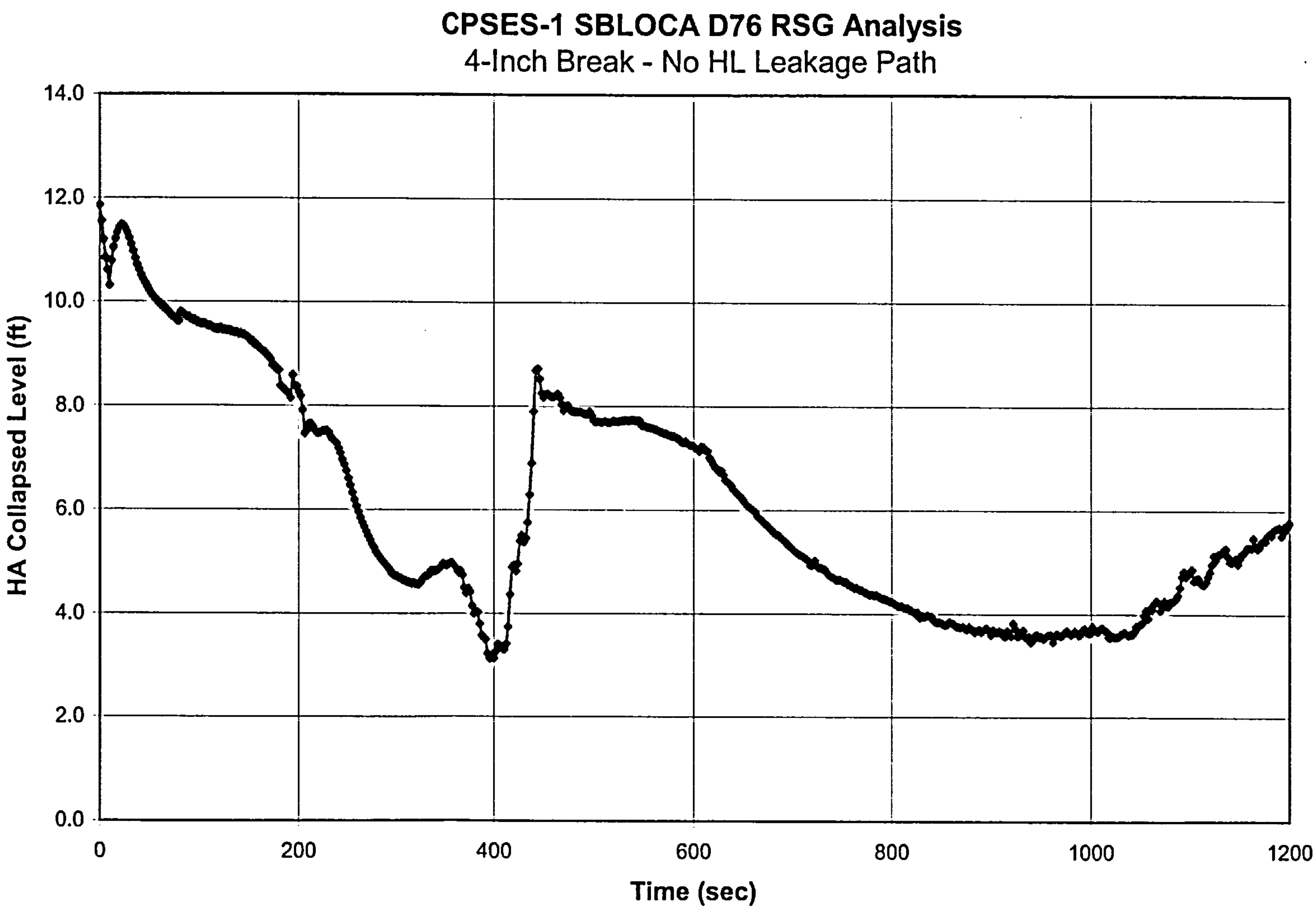

Figure I-6 Hot Assembly Collapsed Water Level – 4-in Break (No HL Leakage Path)

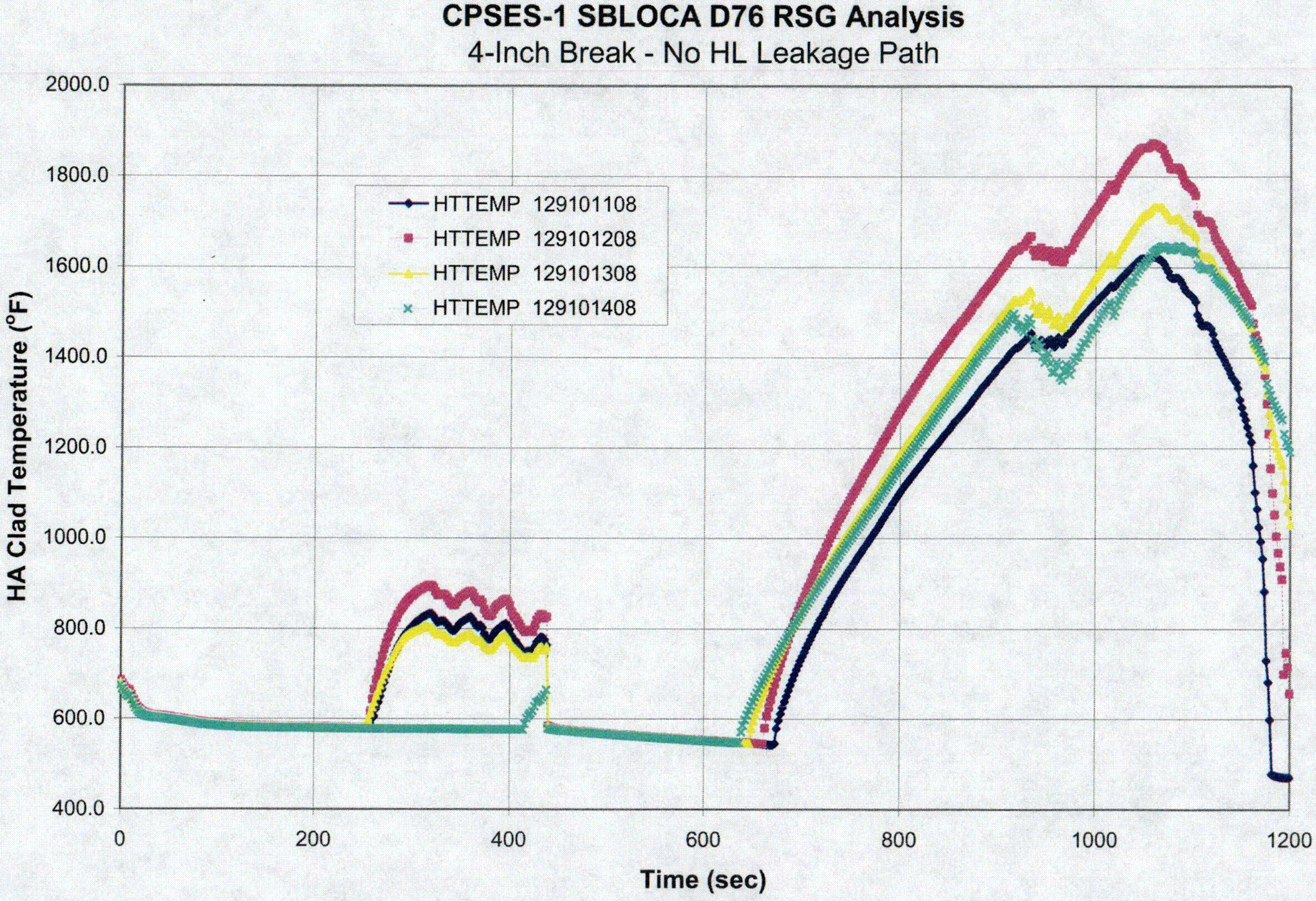

Figure I-7 Hot Assembly Clad Temperatures – 4-in Break (No HL Leakage Path)

C94

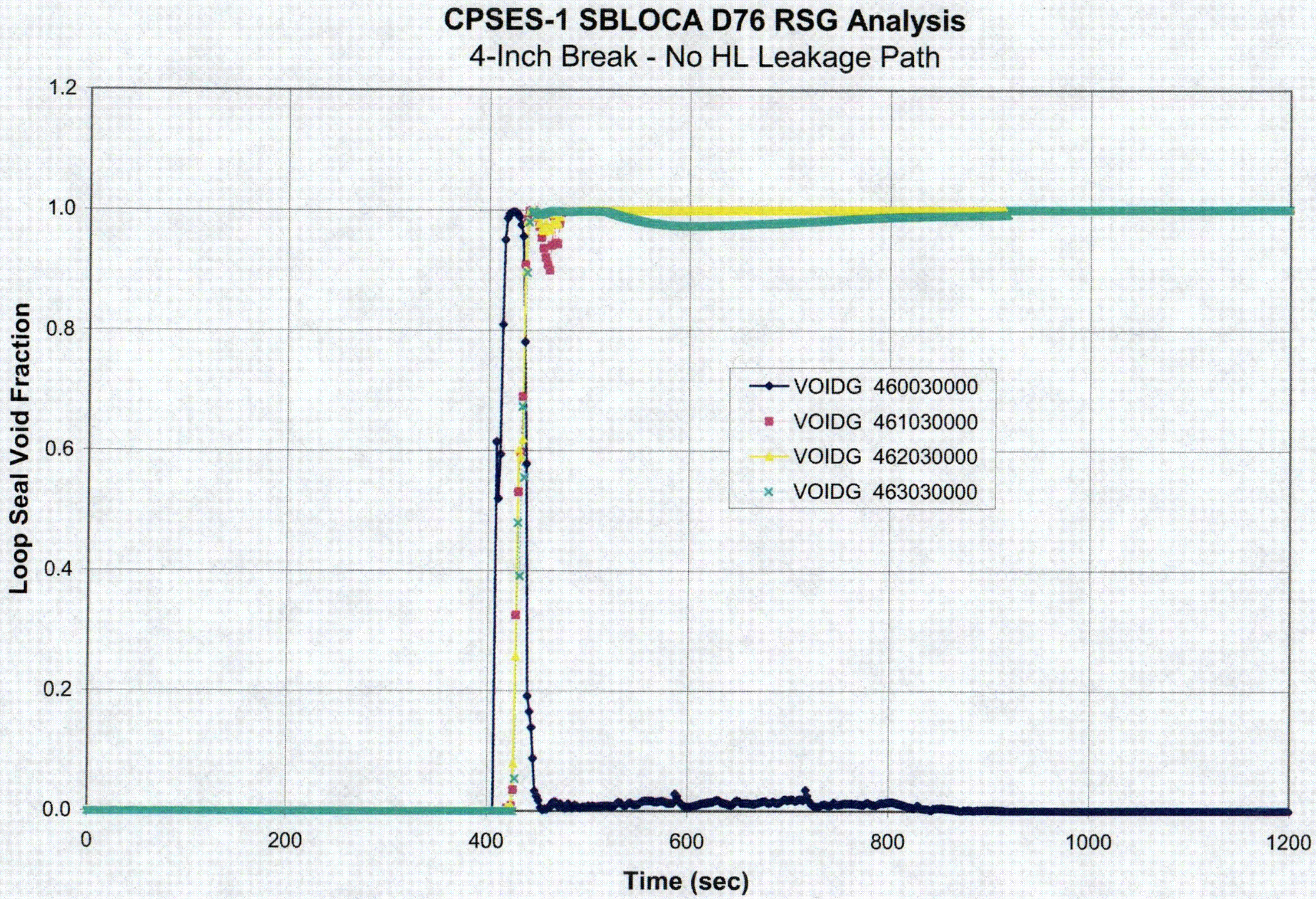

Figure I-8 Loop Seal Void Fractions – 4-in Break (No HL Leakage Path)

C95

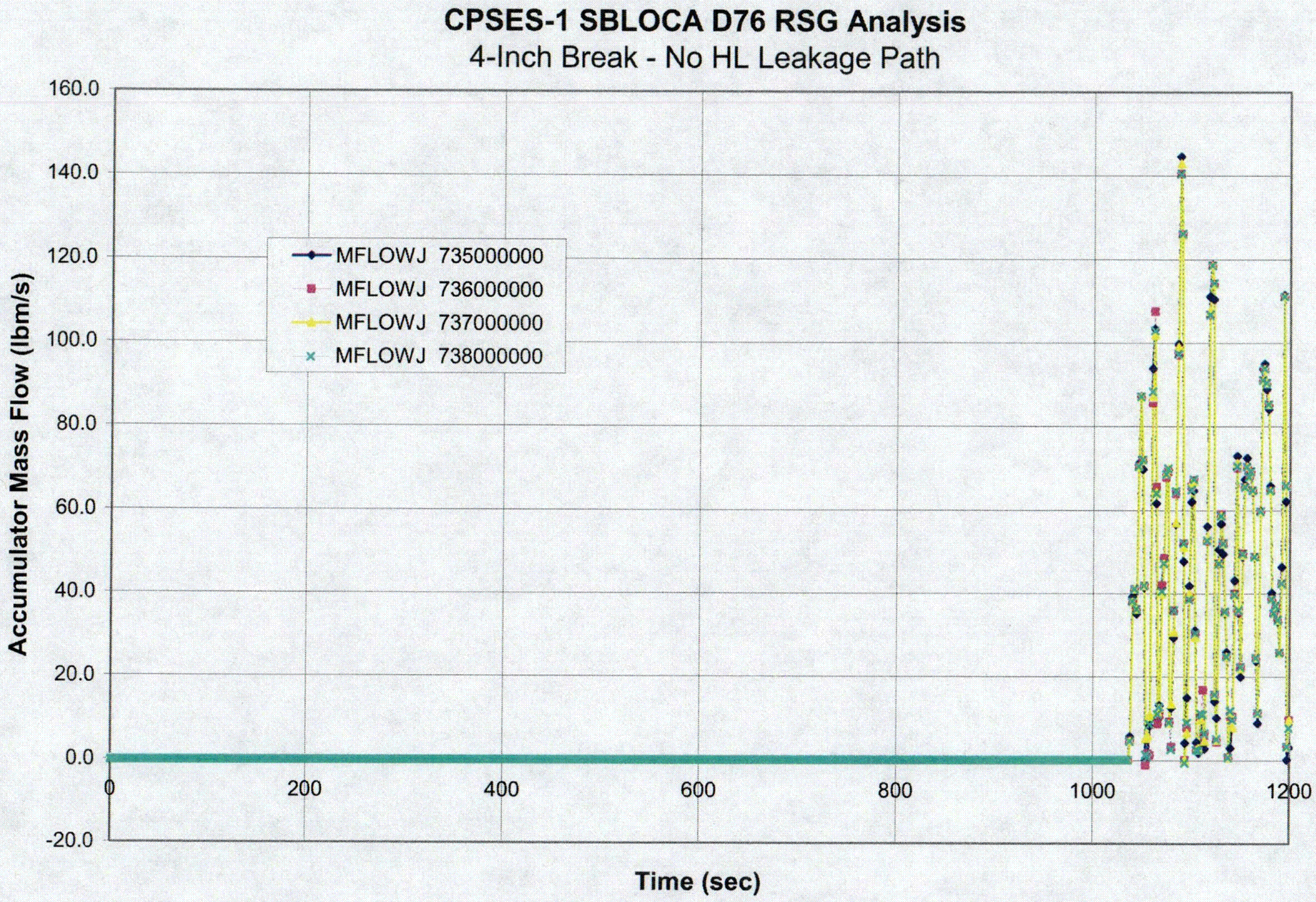

Figure I-9 Accumulator Mass Flow Rates – 4-in Break (No HL Leakage Path)

C96

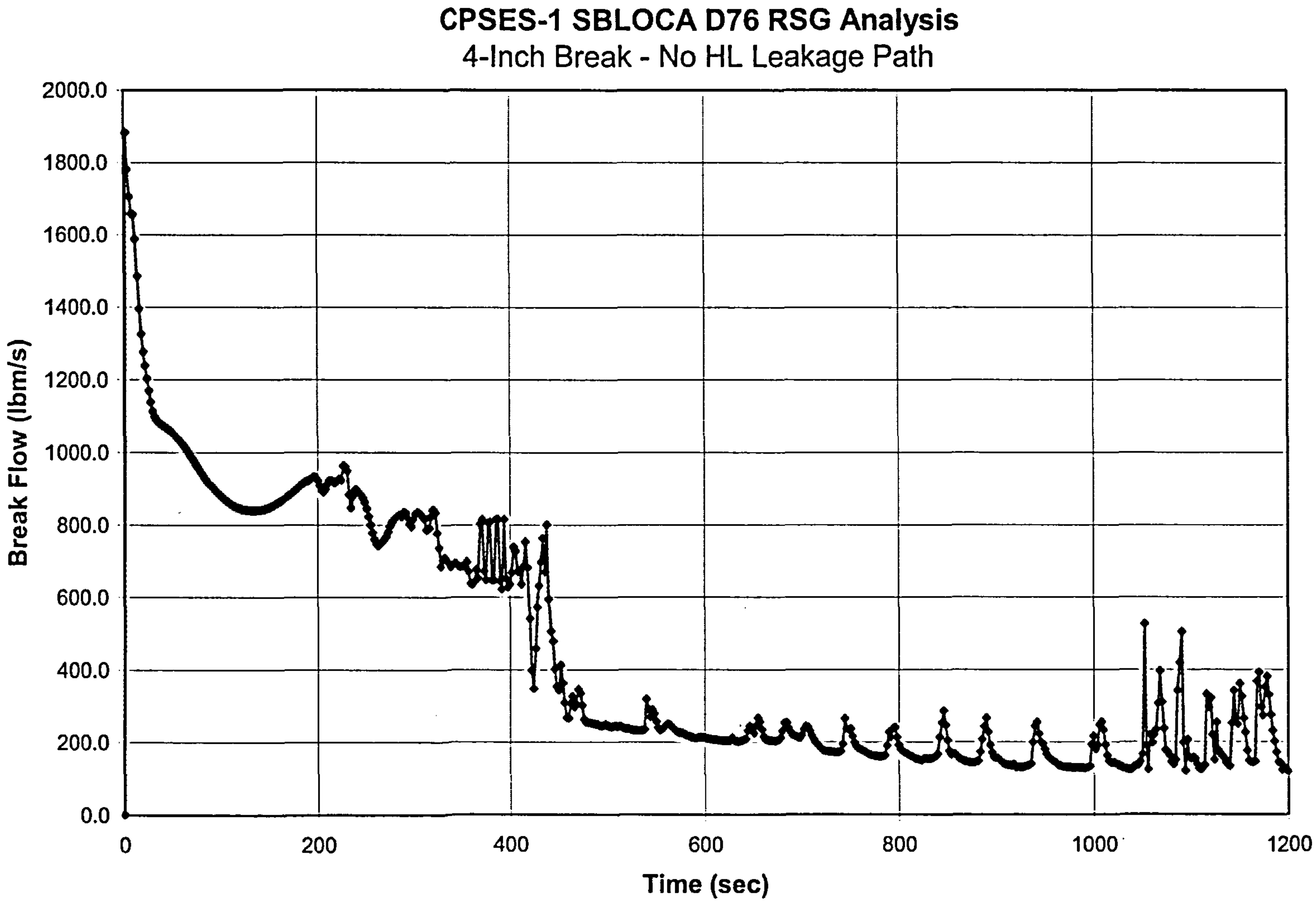

Figure I-10 Break Flow Rate – 4-in Break (No HL Leakage Path)

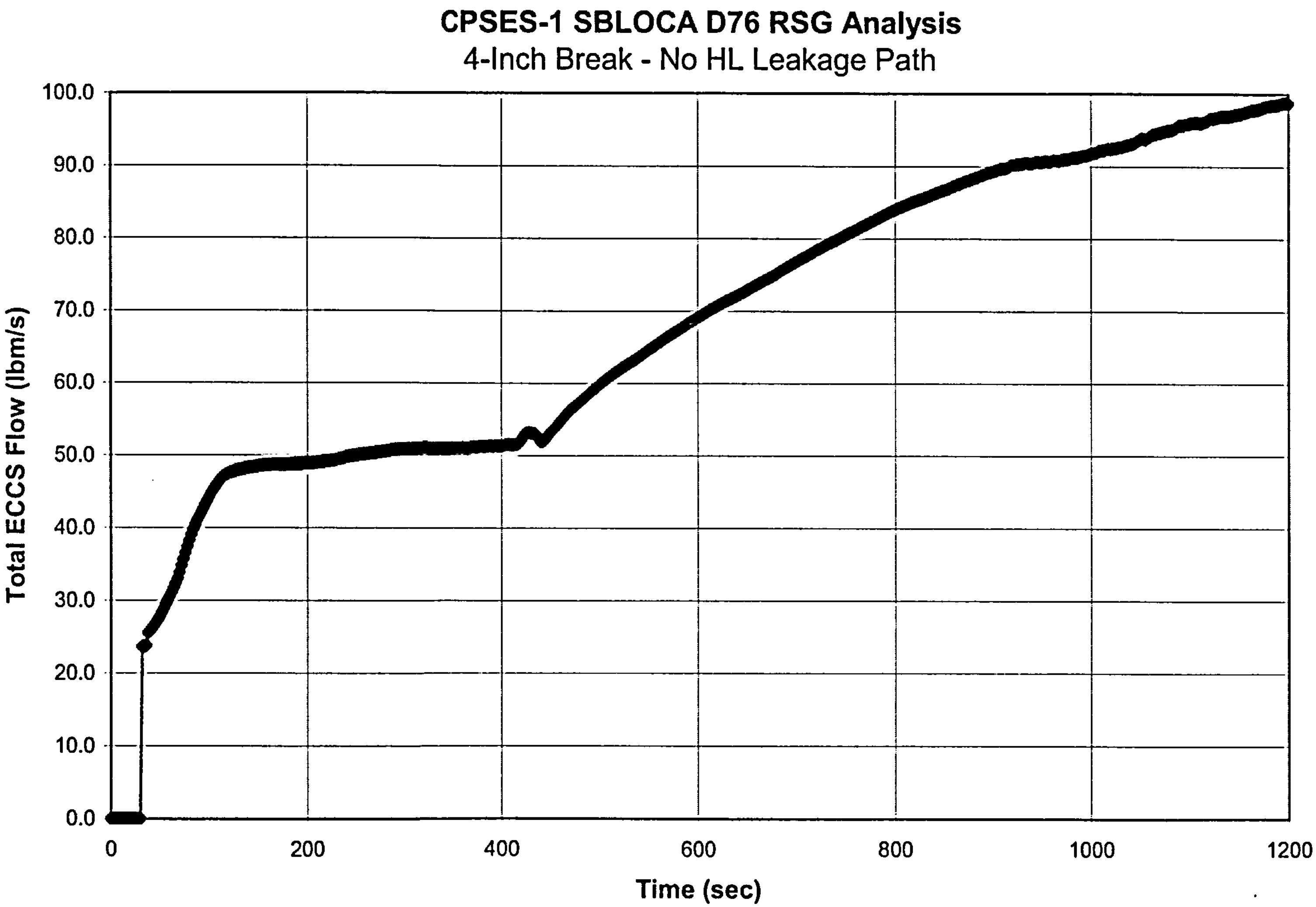

Figure I-11 Total Pumped ECCS Flow Rate – 4-in Break (No HL Leakage Path)

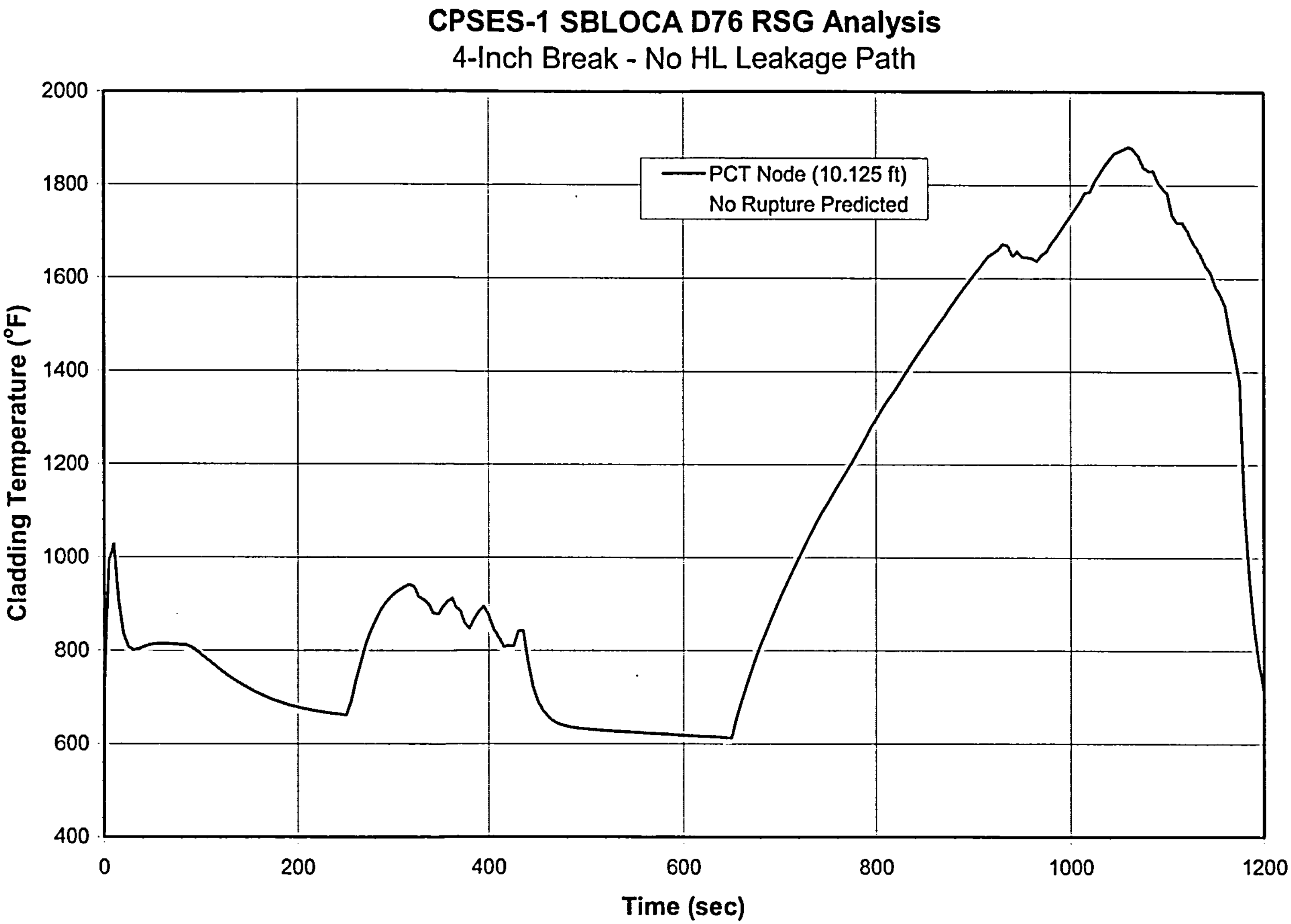

Figure I-12 TOODEE2 Clad Temperature – 4-in Break (No HL Leakage Path)

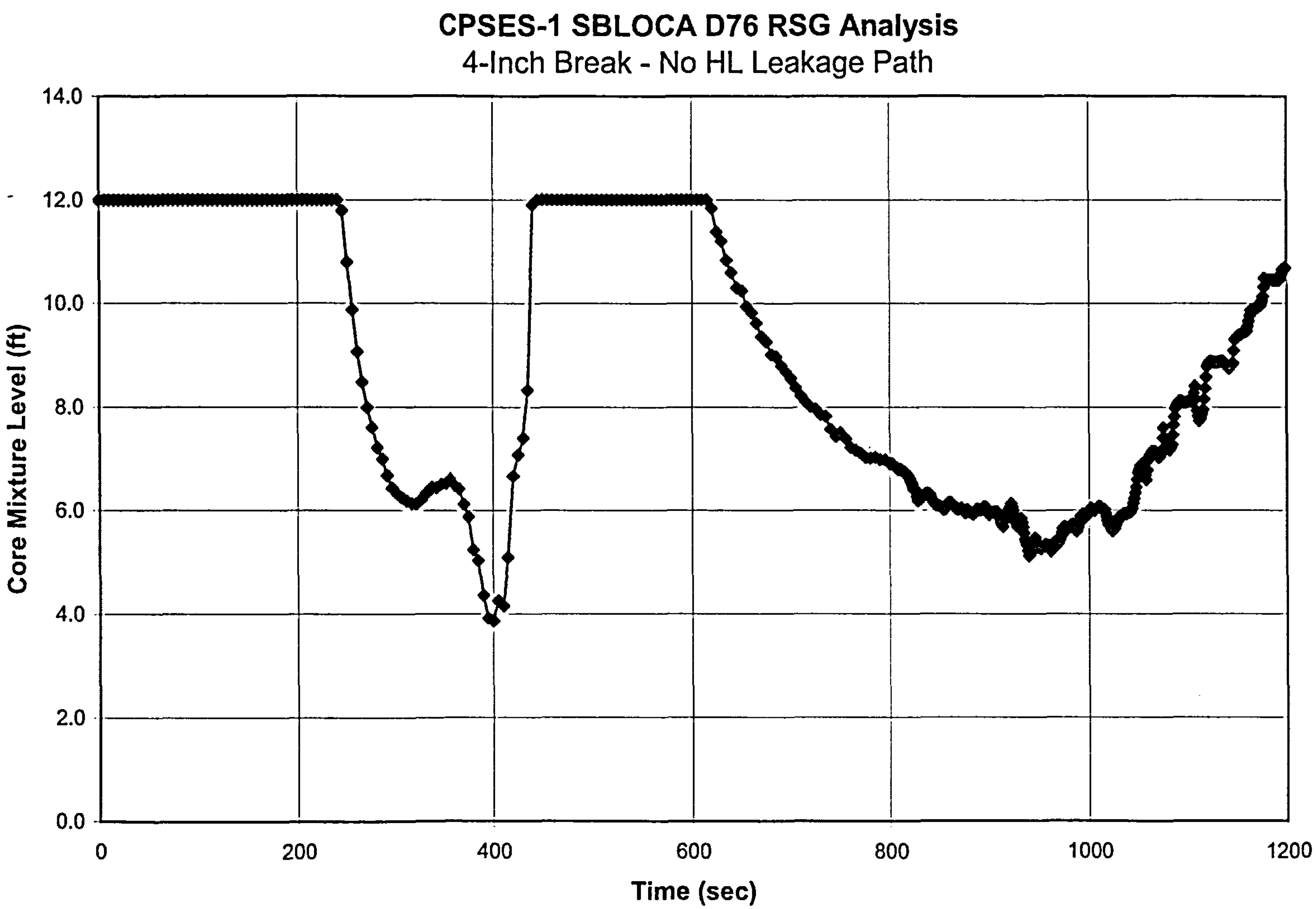

Figure I-13 Core Mixture Level – 4-in Break (No HL Leakage Path)

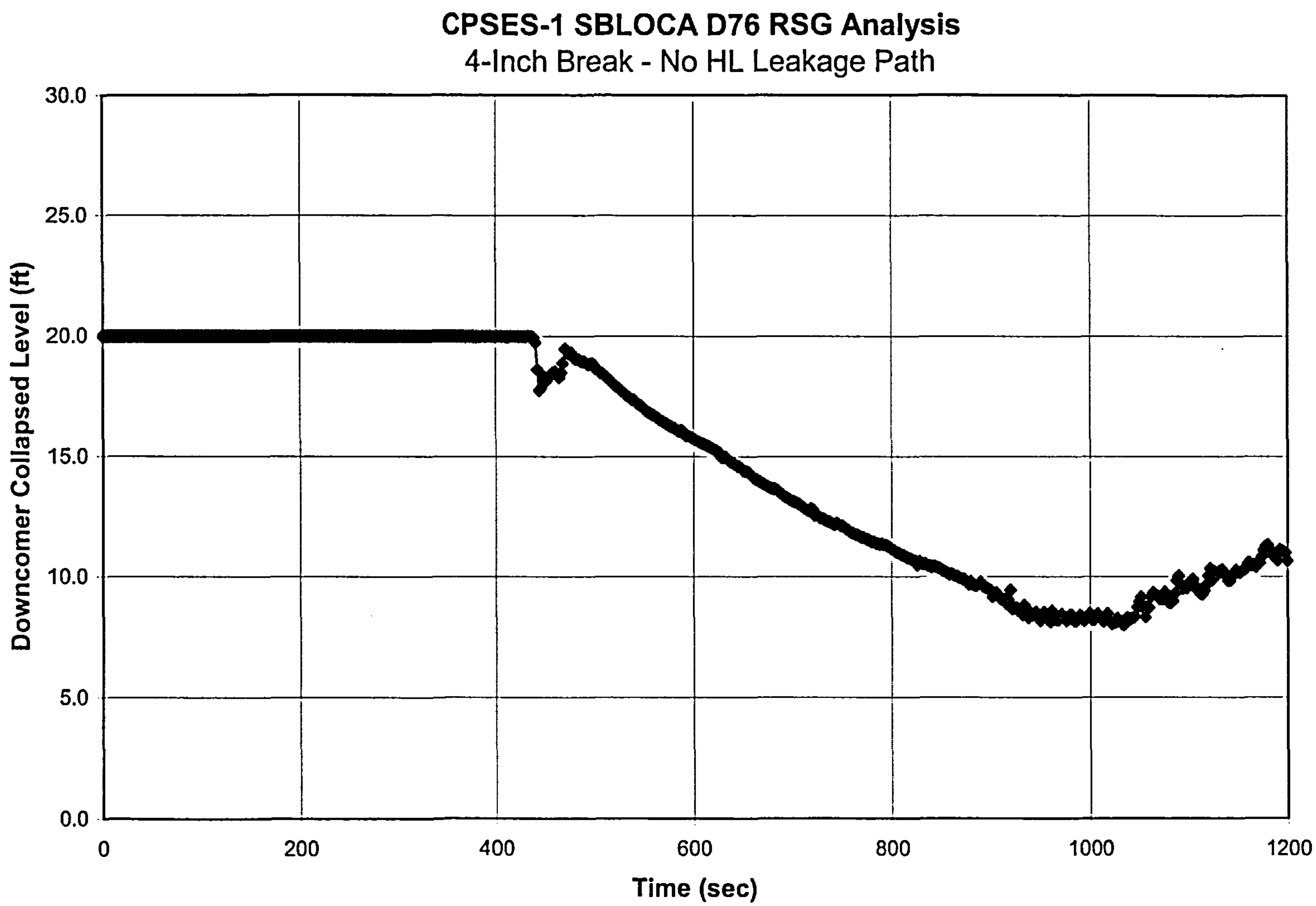

Figure I-14 Downcomer Liquid Level – 4-in Break (No HL Leakage Path)

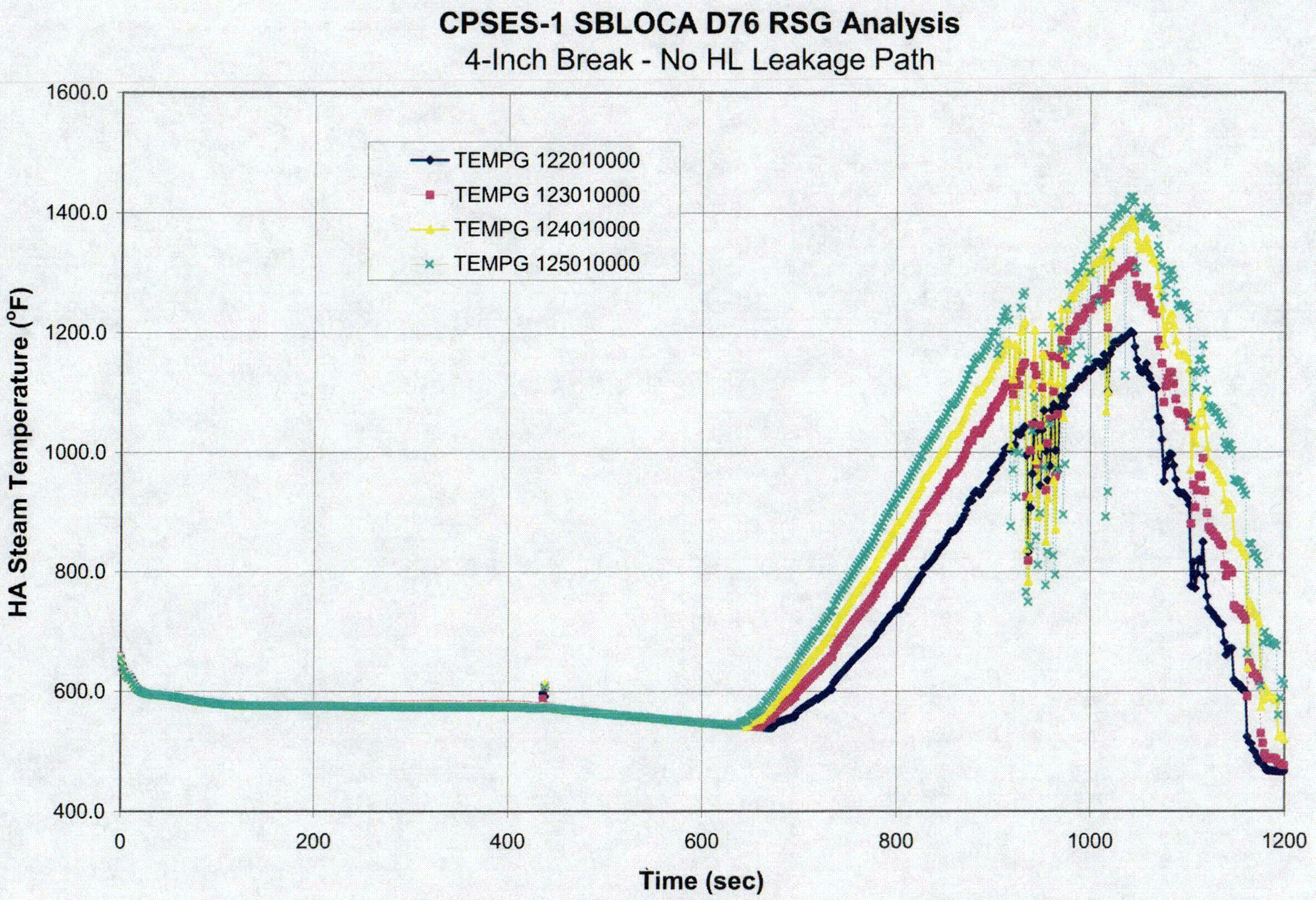

Figure I-15 Hot Assembly Steam Temperatures – 4-in Break (No HL Leakage Path)

C97

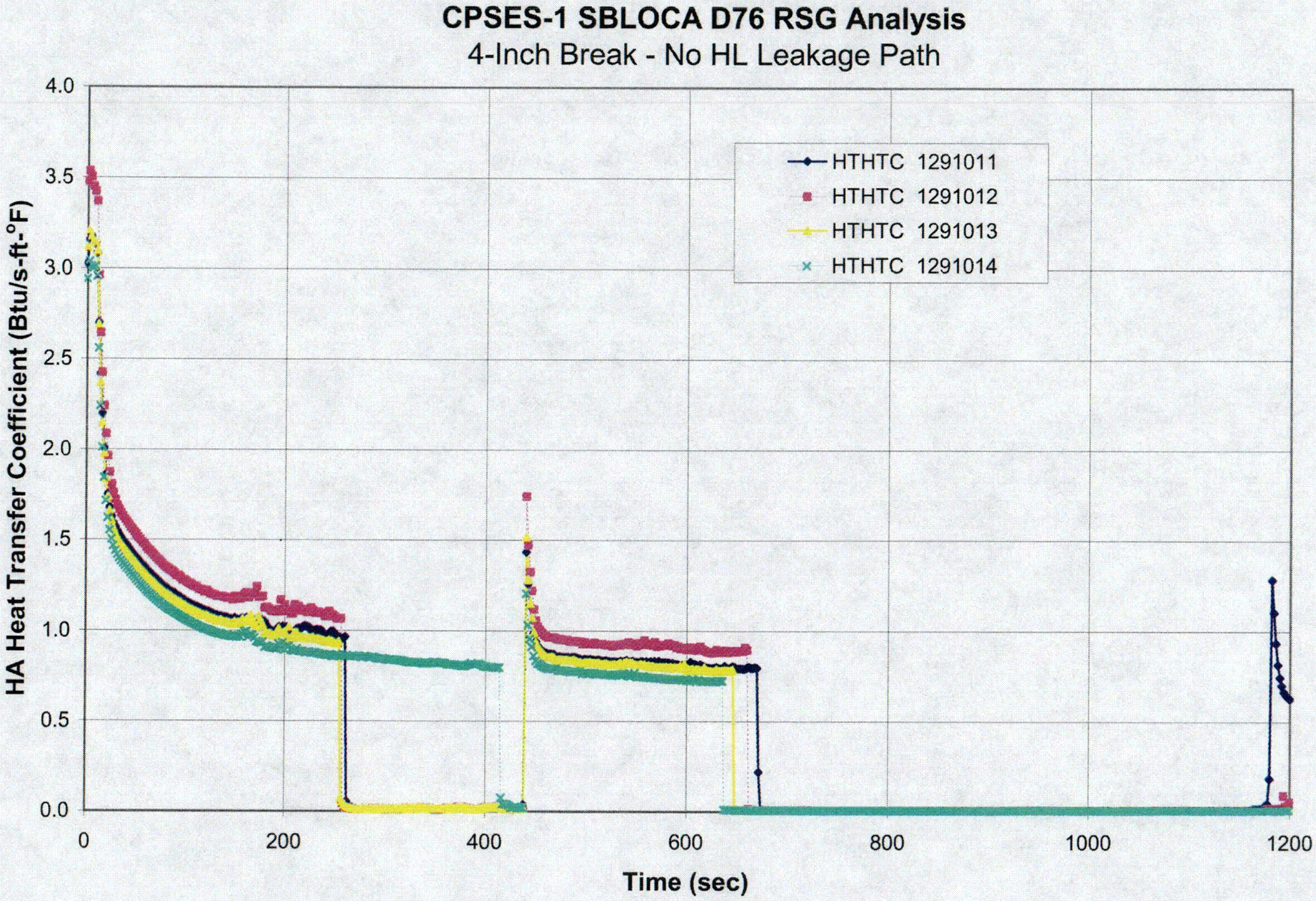

Figure I-16 Hot Assembly Heat Transfer Coefficients – 4-in Break (No HL Leakage Path)

C98

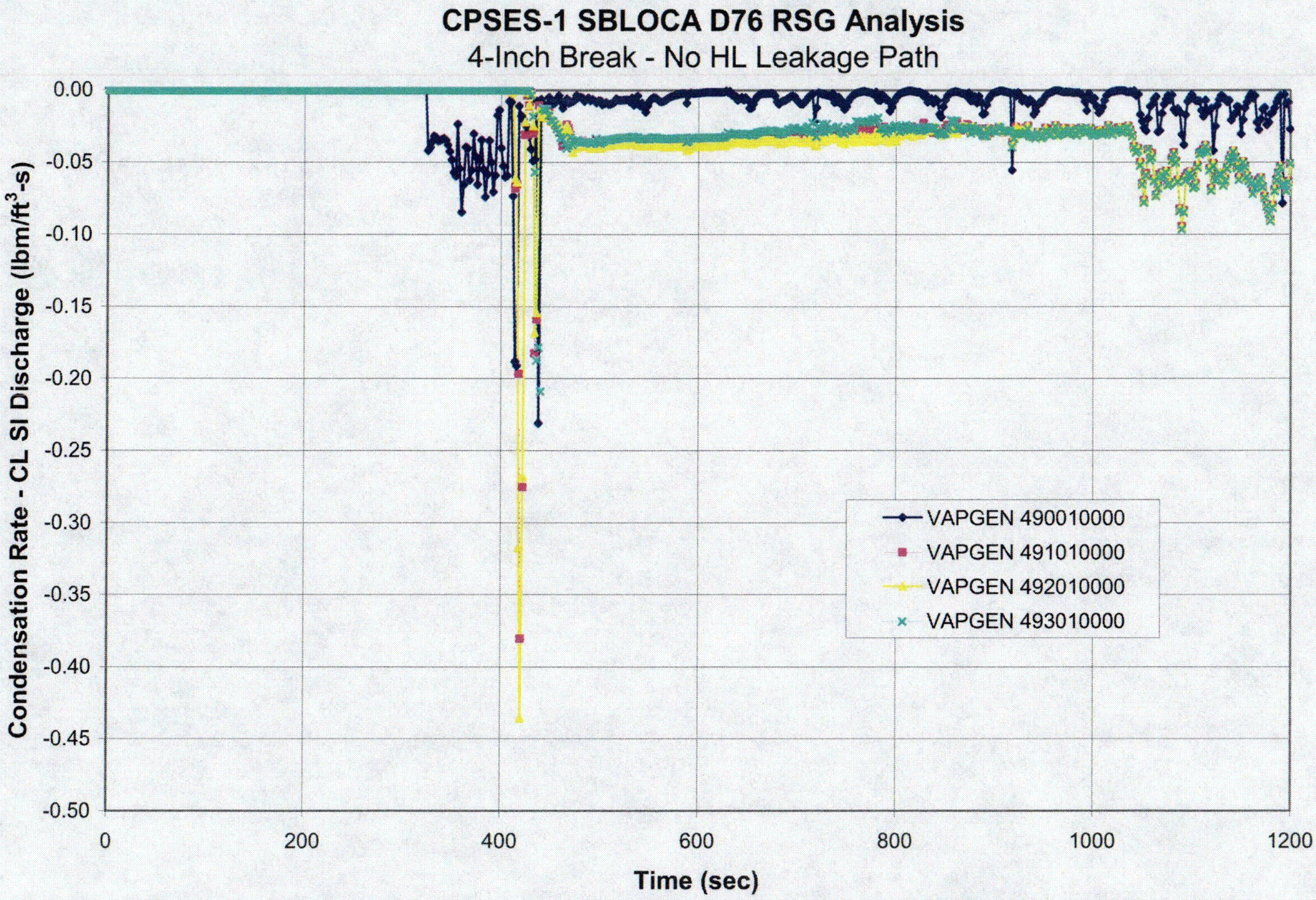

Figure I-17 Condensation Rate in Cold Leg Discharge – 4-in Break (No HL Leakage Path)

C99

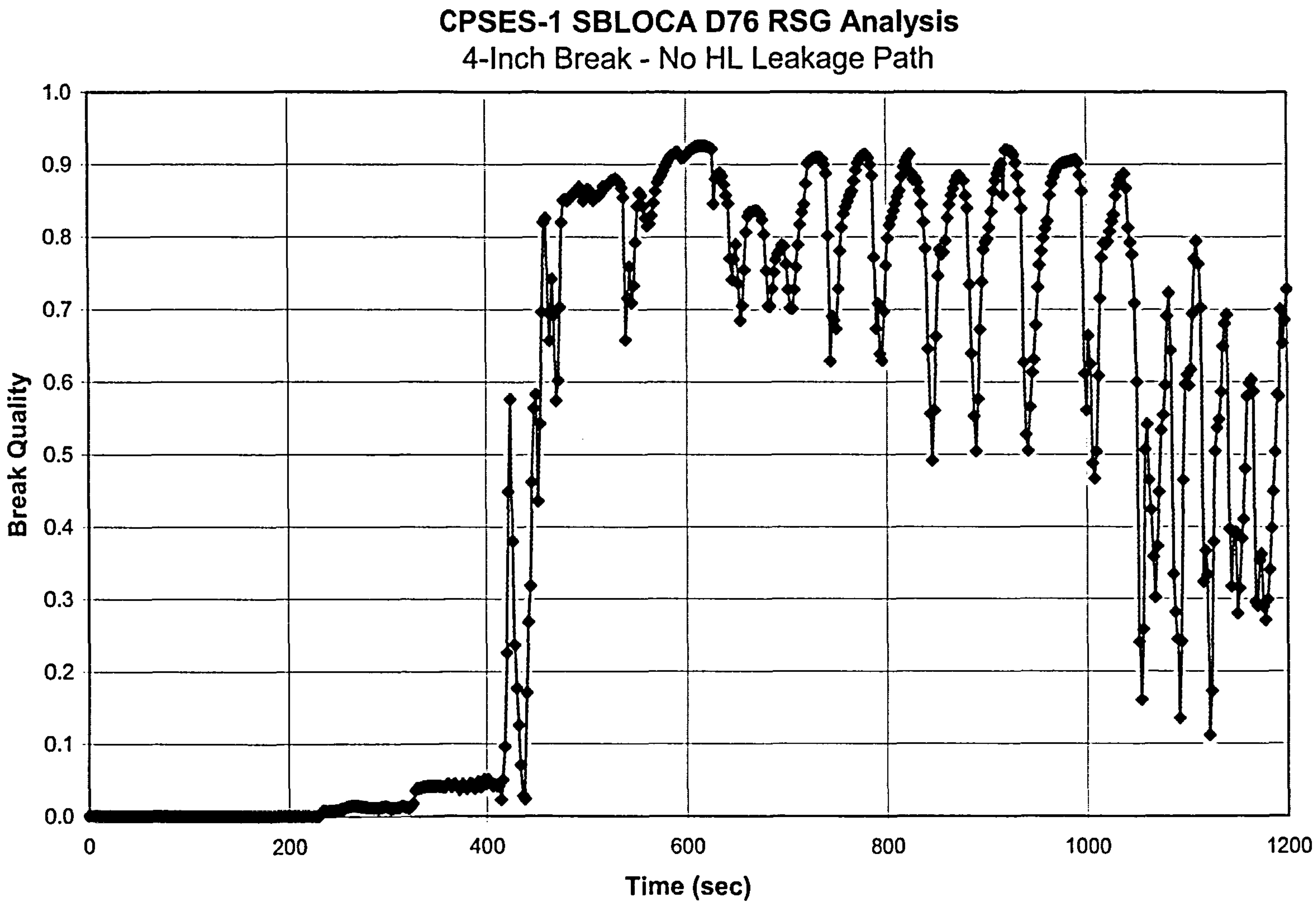

Figure I-18 Break Quality – 4-in Break (No HL Leakage Path)

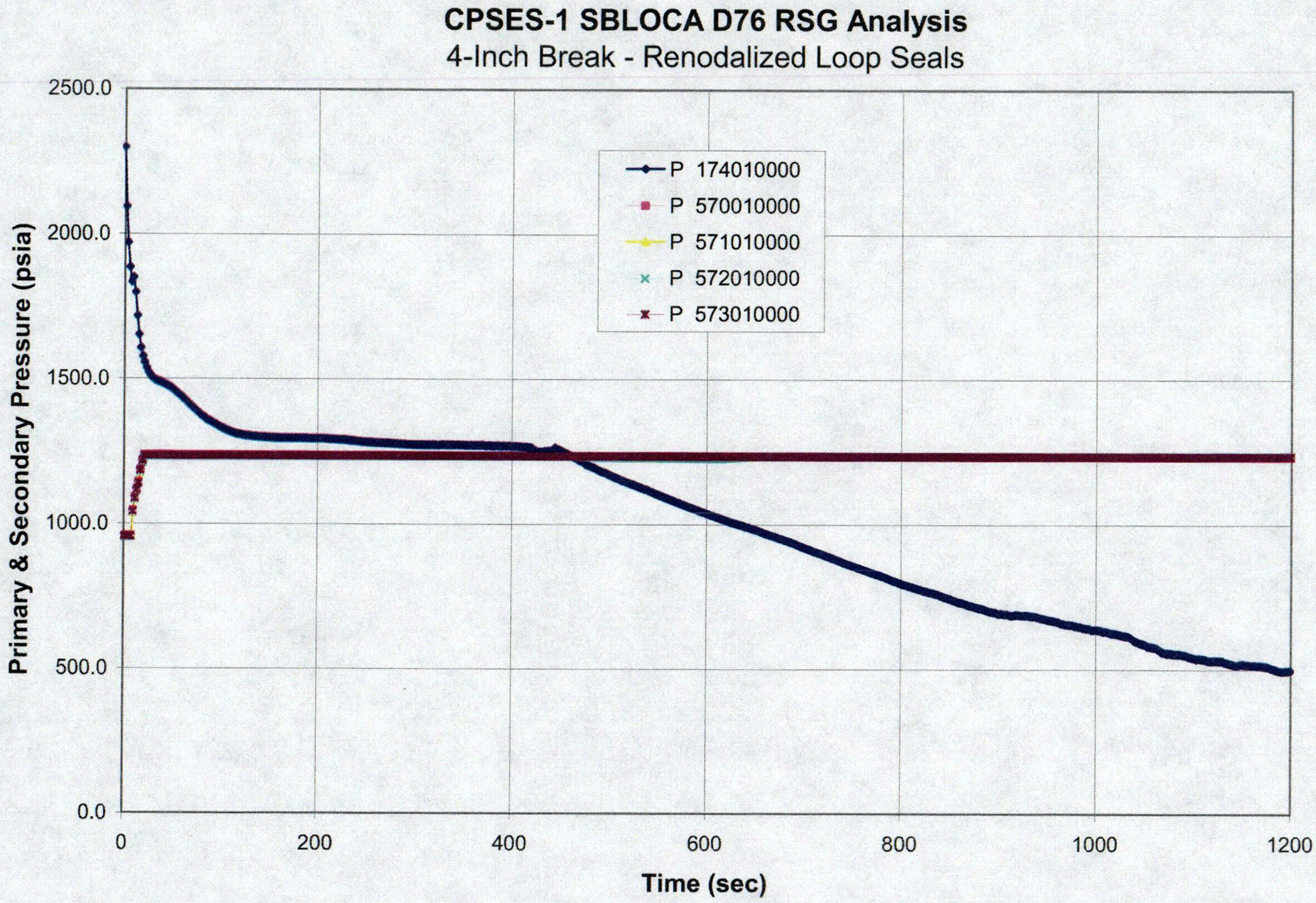

Figure J-1 Primary and Secondary System Pressures – 4-in Break (Renodalized Loop Seals)

C100

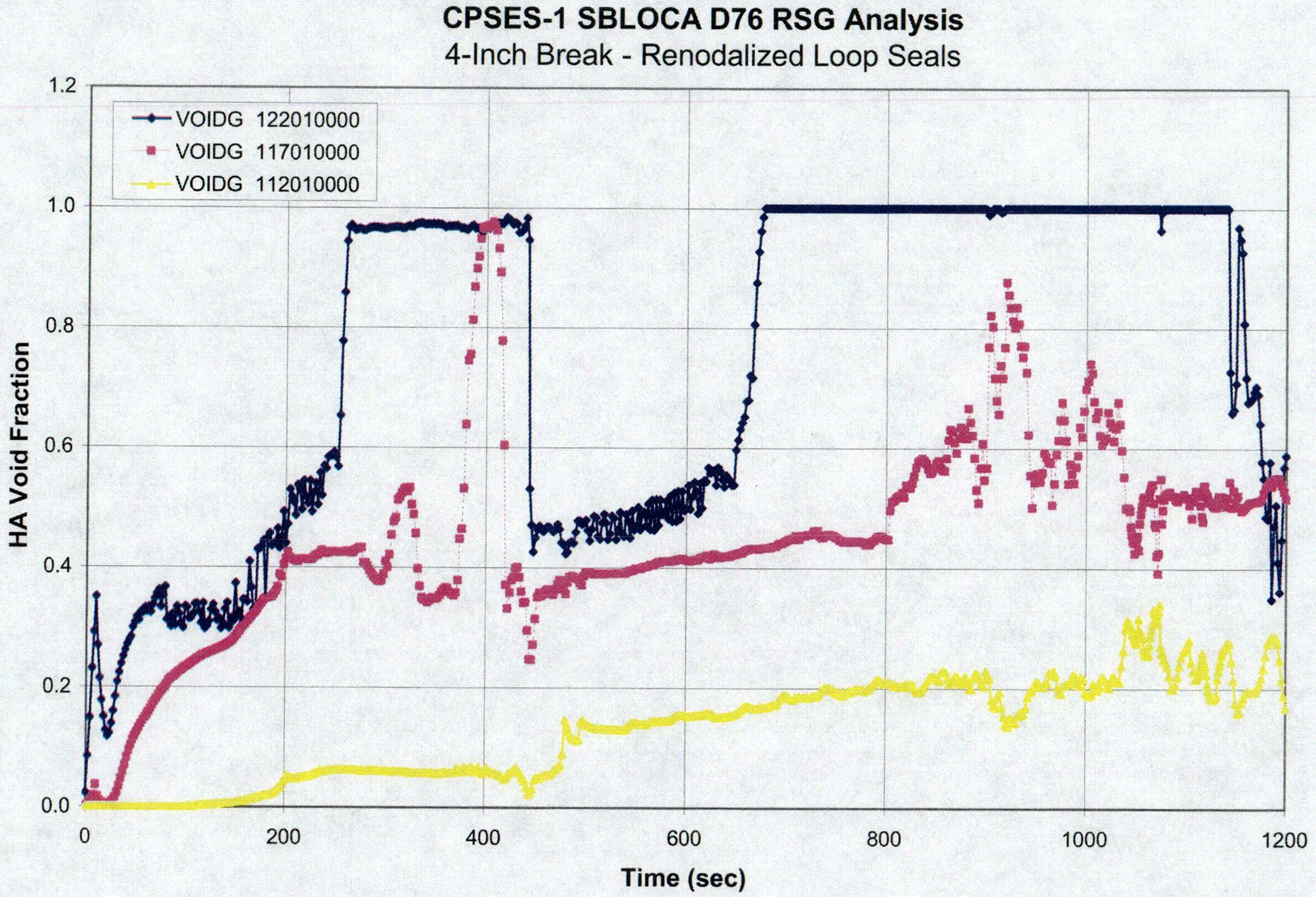

Figure J-2 Hot Assembly Region Void Fractions – 4-in Break (Renodalized Loop Seals)

C101

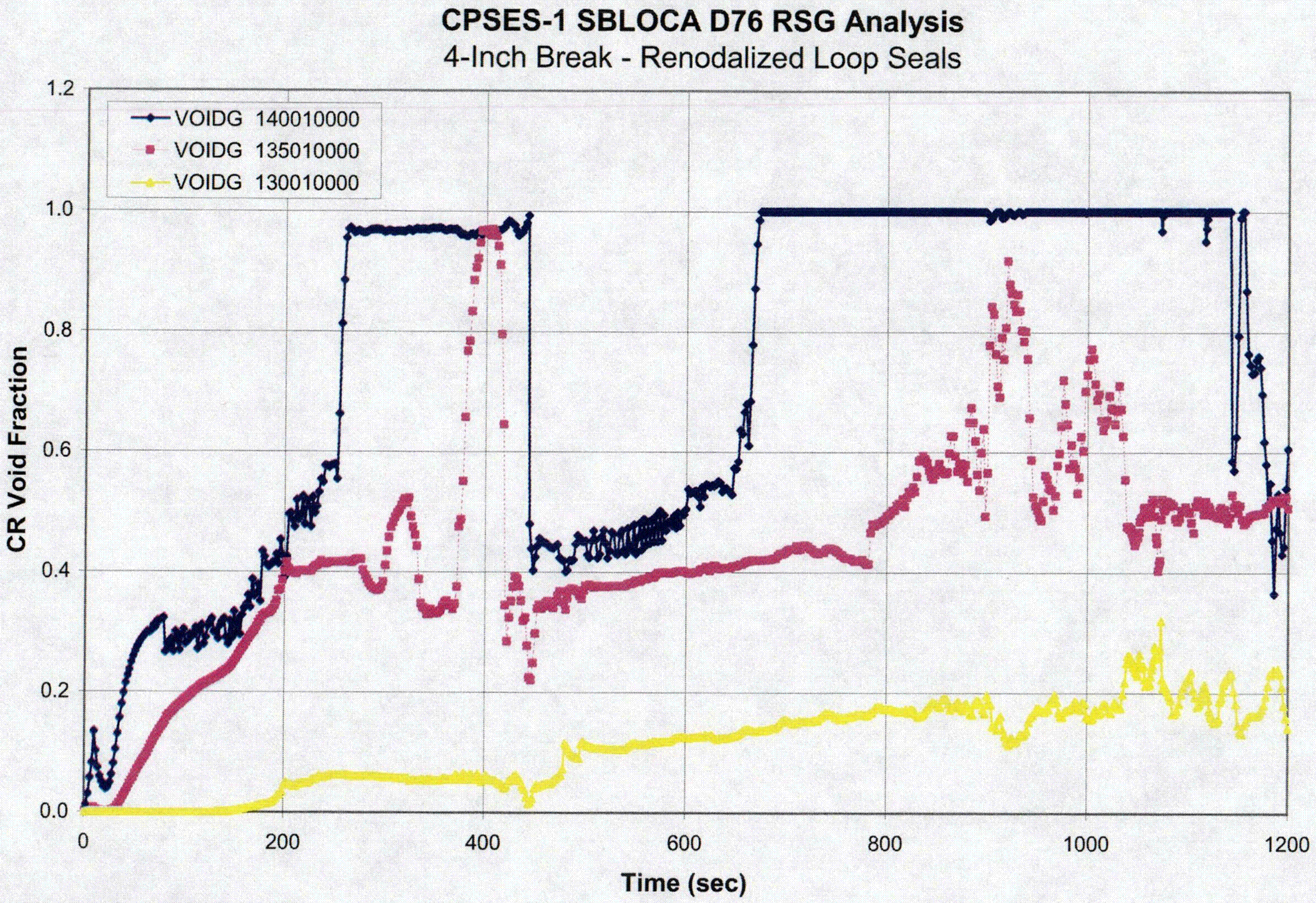

Figure J-3 Central Core Region Void Fractions – 4-in Break (Renodalized Loop Seals)

C102

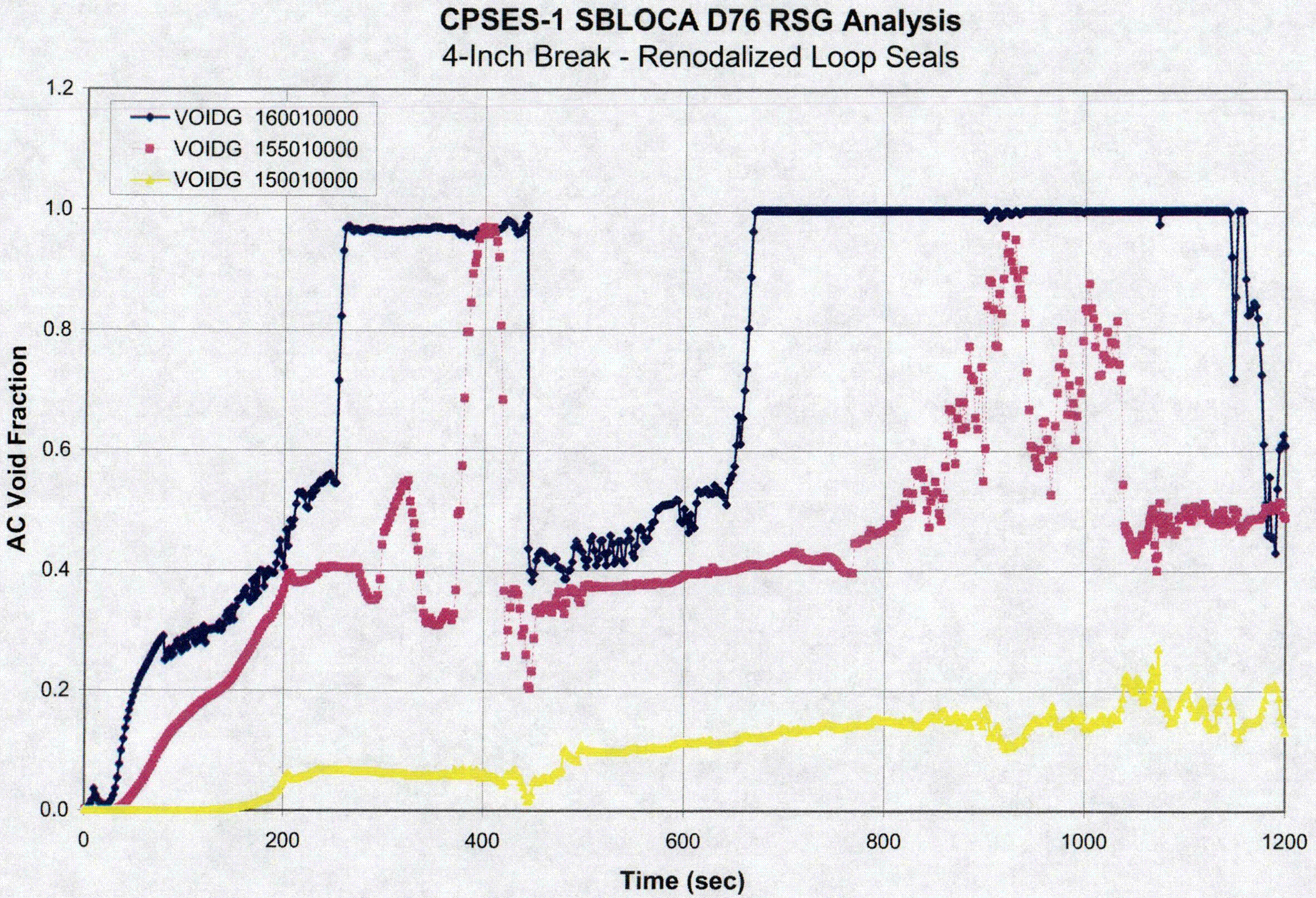

Figure J-4 Average Core Region Void Fractions – 4-in Break (Renodalized Loop Seals)

C103

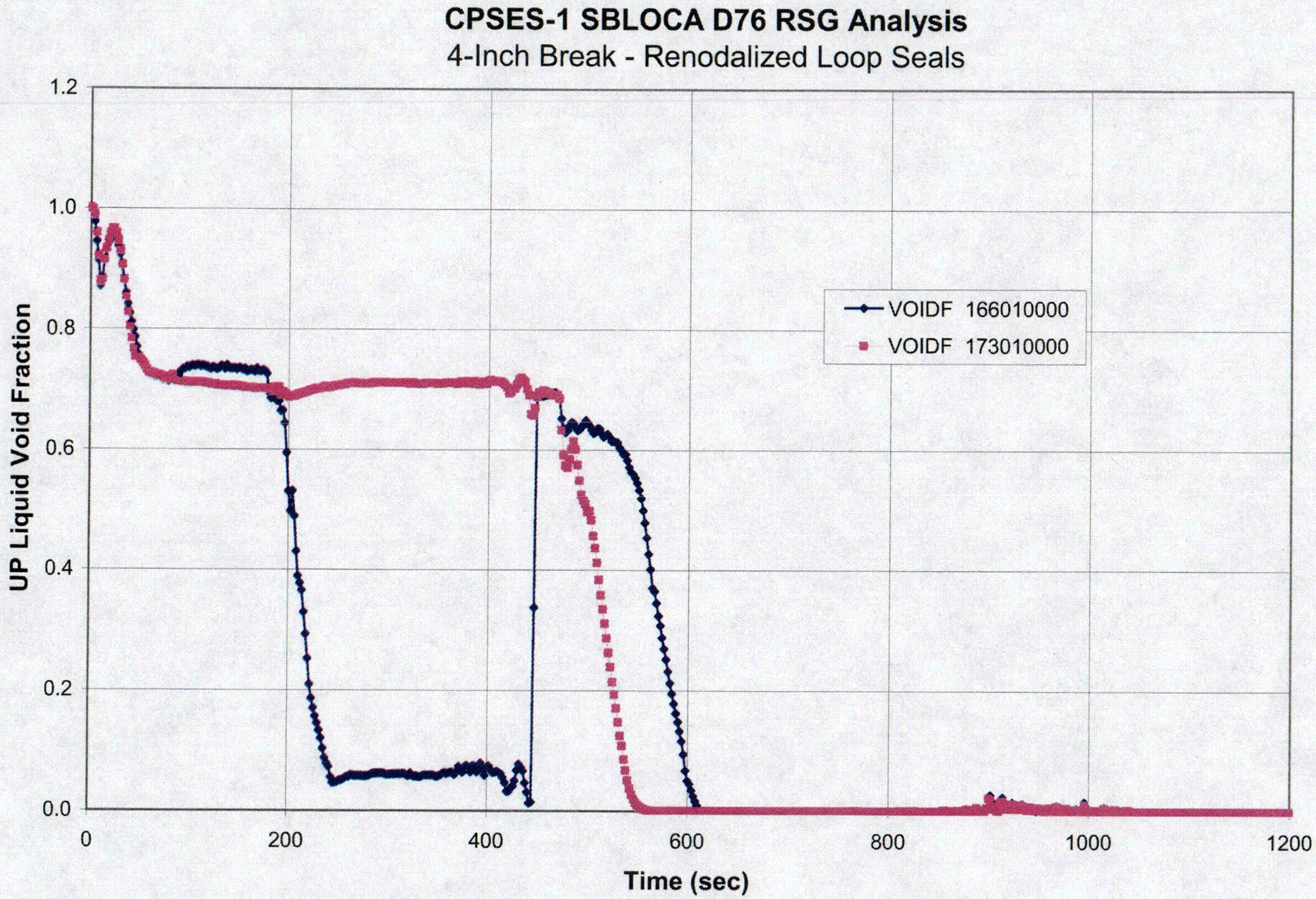

Figure J-5 Upper Plenum Liquid Fraction – 4-in Break (Renodalized Loop Seals)

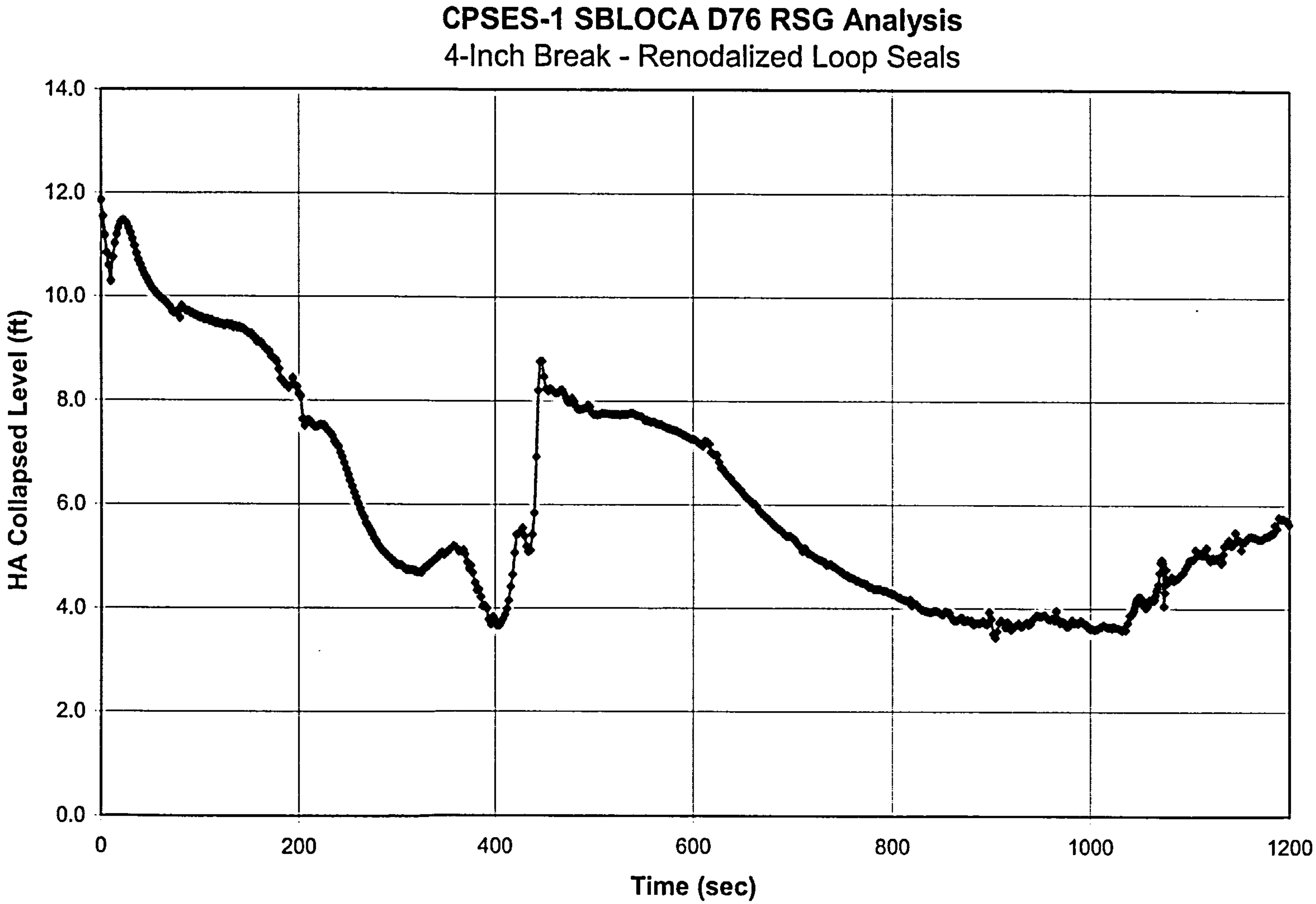

Figure J-6 Hot Assembly Collapsed Water Level – 4-in Break (Renodalized Loop Seals)

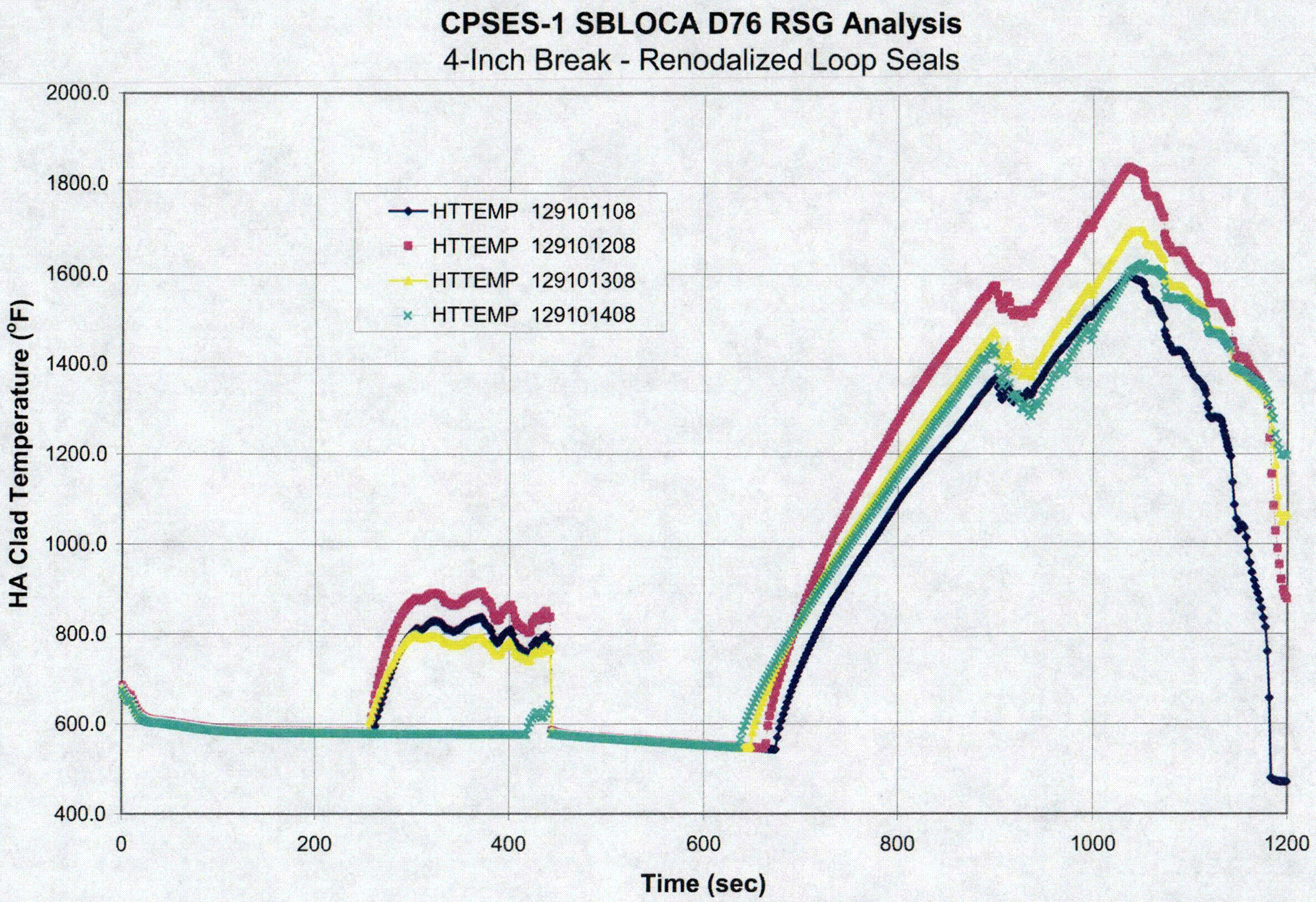

Figure J-7 Hot Assembly Clad Temperatures – 4-in Break (Renodalized Loop Seals)

C105

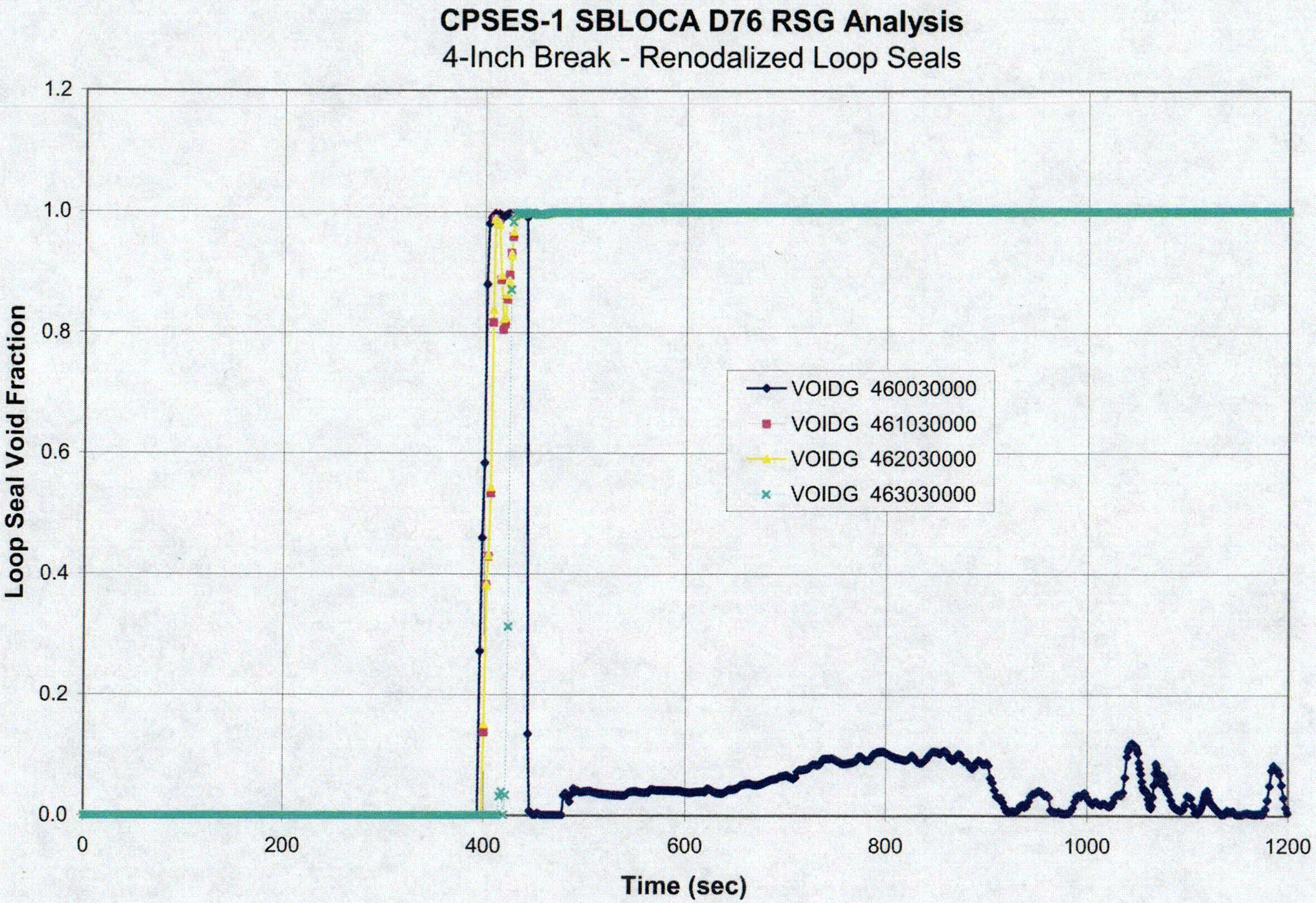

Figure J-8 Loop Seal Void Fractions – 4-in Break (Renodalized Loop Seals)

C106

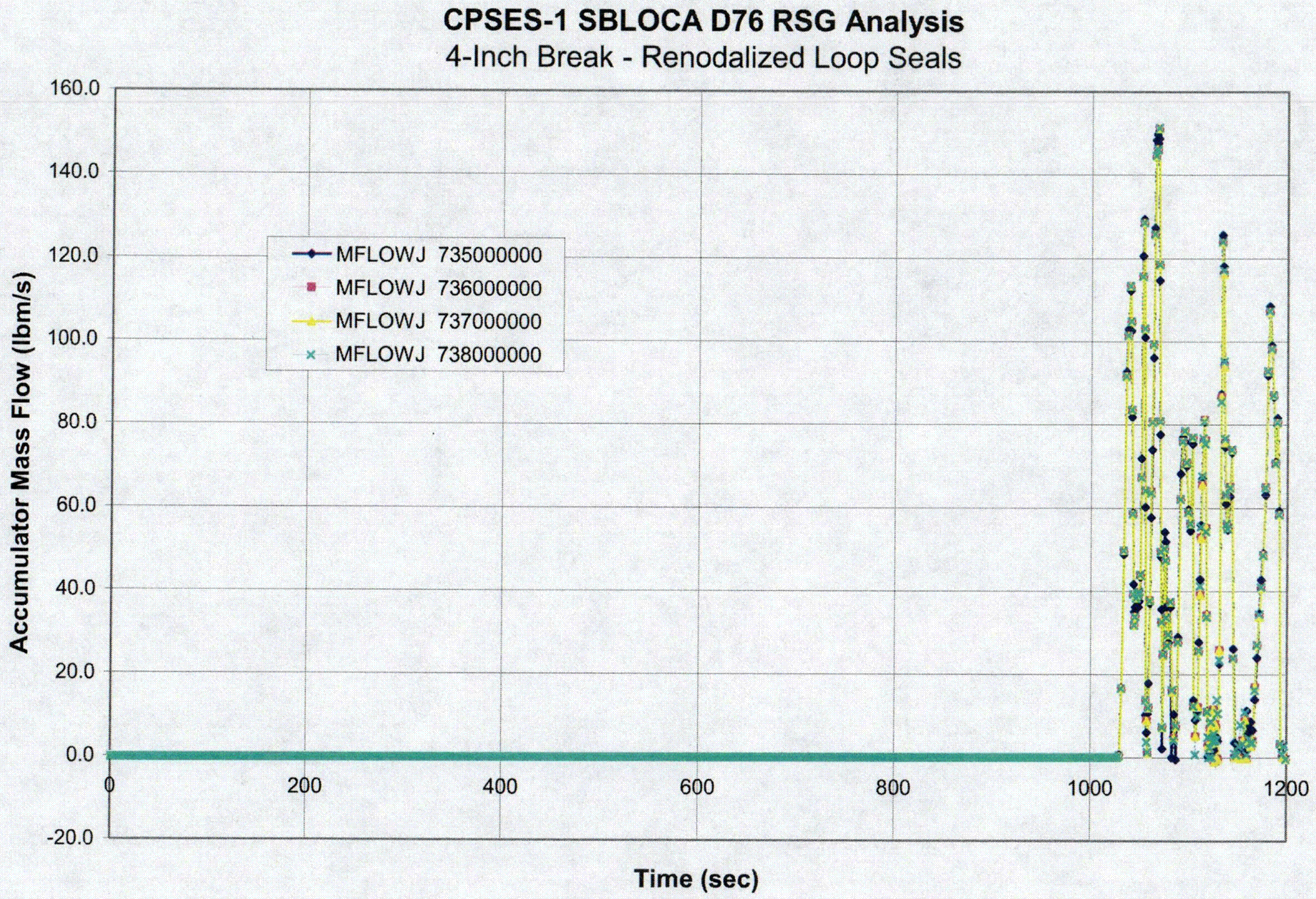

Figure J-9 Accumulator Mass Flow Rates – 4-in Break (Renodalized Loop Seals)

C107

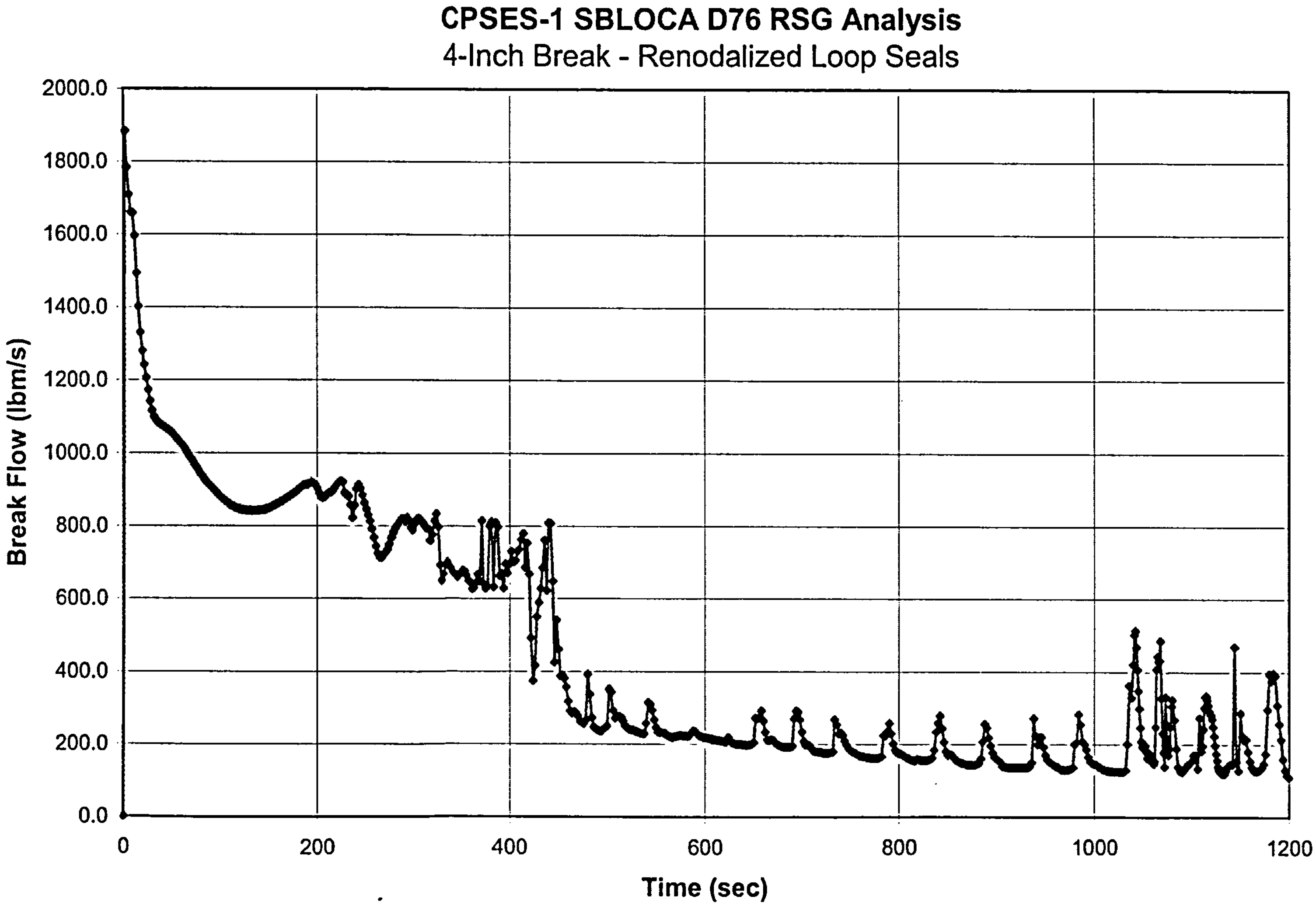

Figure J-10 Break Flow Rate – 4-in Break (Renodalized Loop Seals)

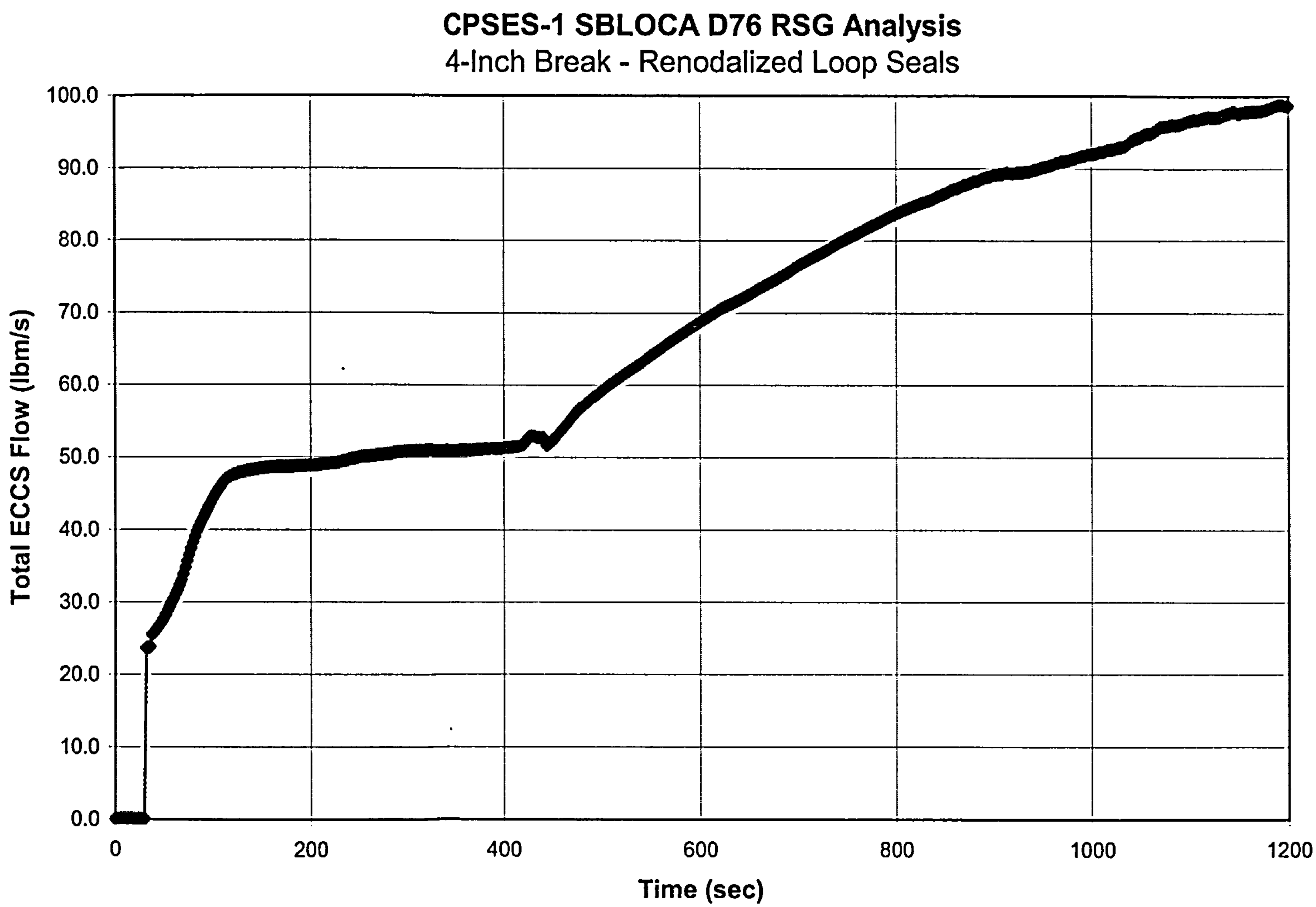

Figure J-11 Total Pumped ECCS Flow Rate – 4-in Break (Renodalized Loop Seals)

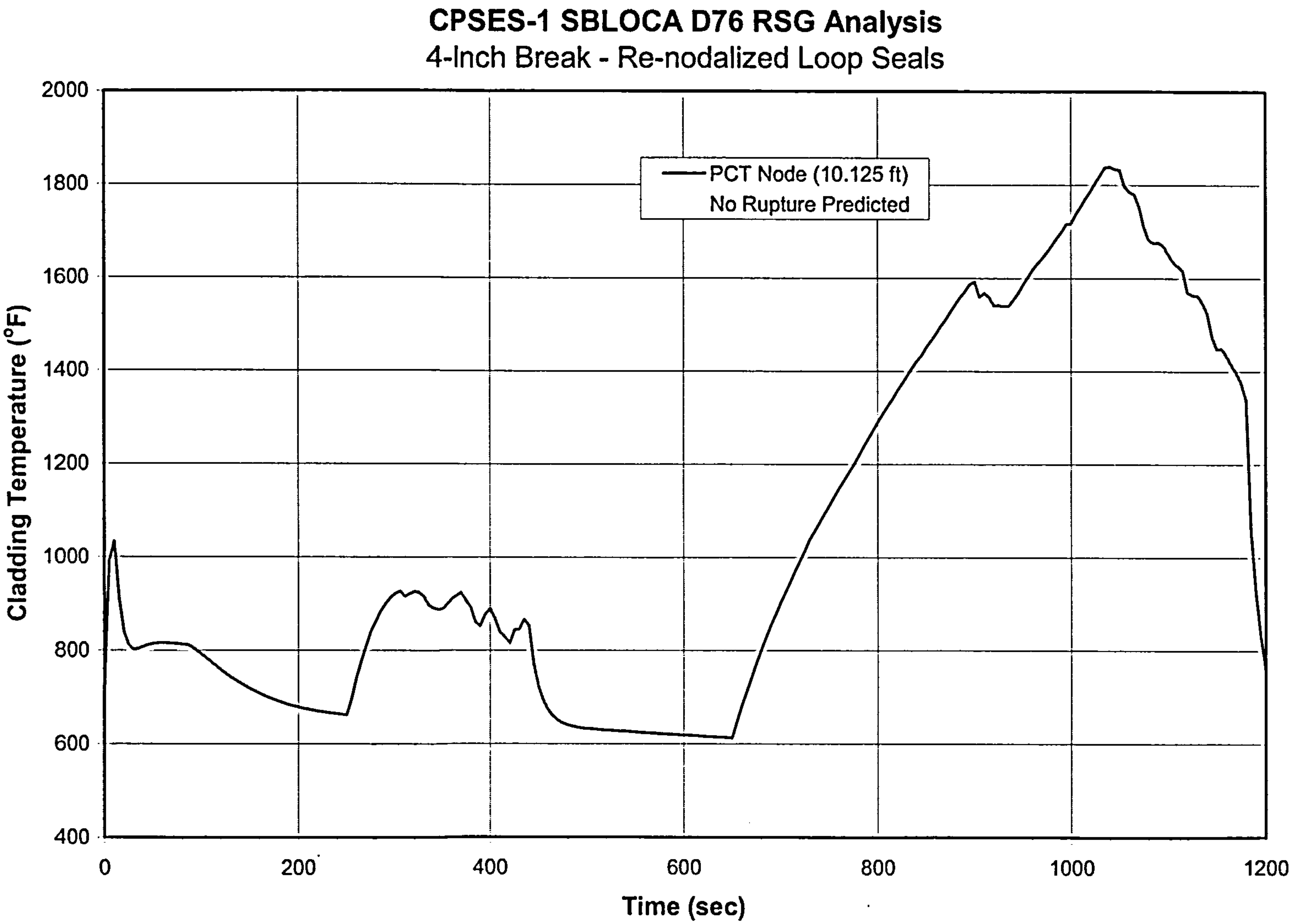

Figure J-12 TOODEE2 Clad Temperature – 4-in Break (Renodalized Loop Seals)

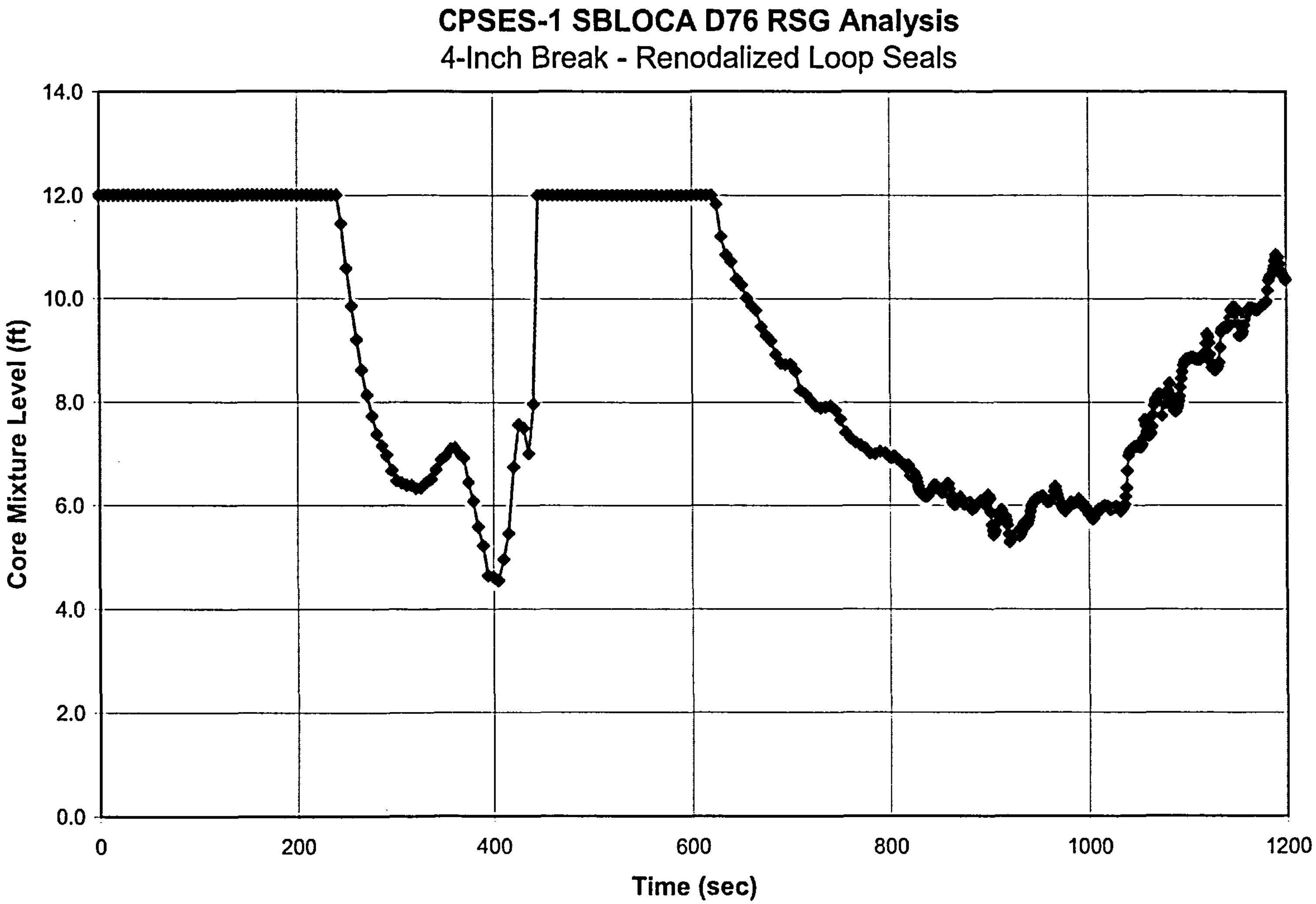

Figure J-13 Core Mixture Level – 4-in Break (Renodalized Loop Seals)

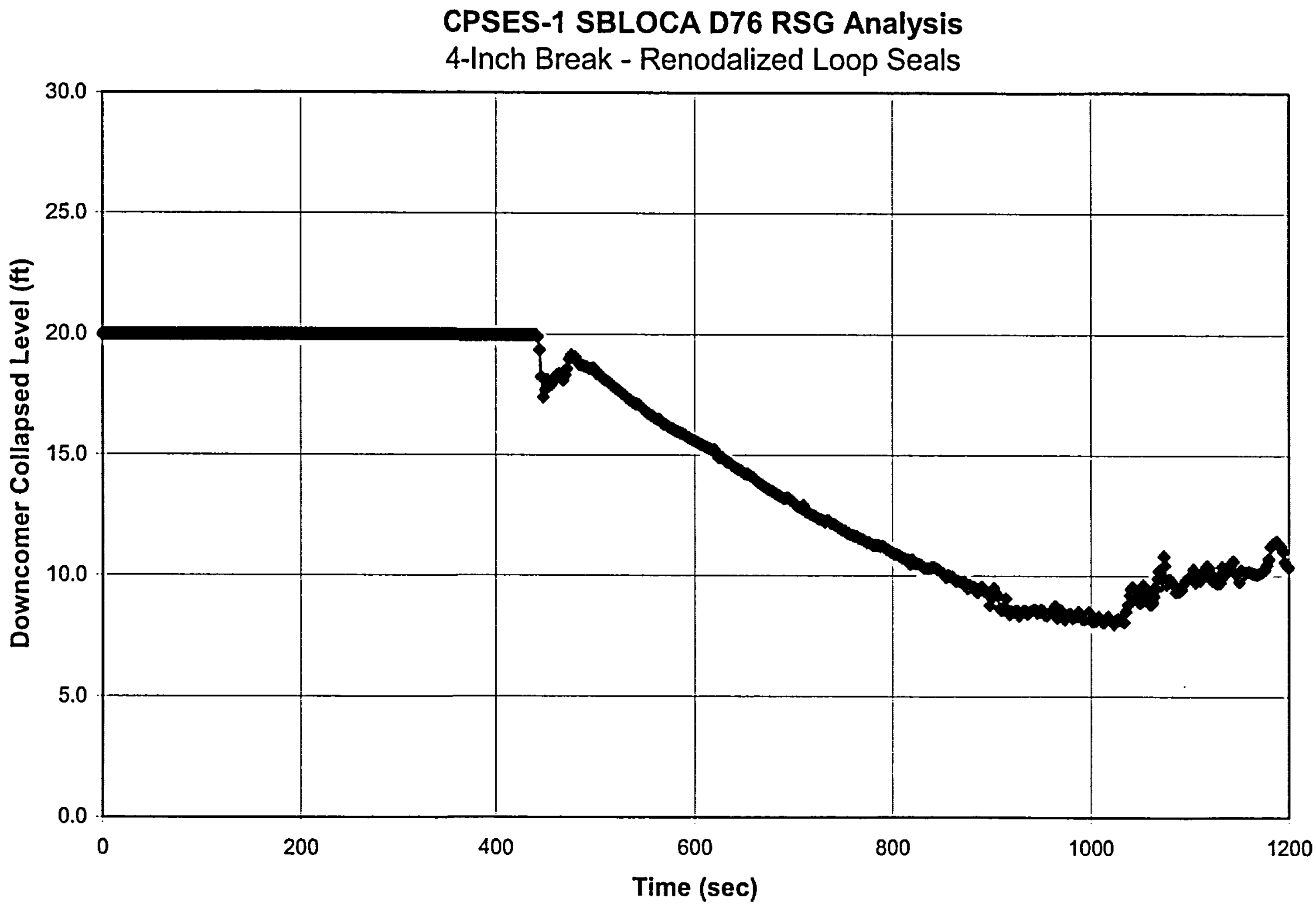

Figure J-14 Downcomer Liquid Level – 4-in Break (Renodalized Loop Seals)

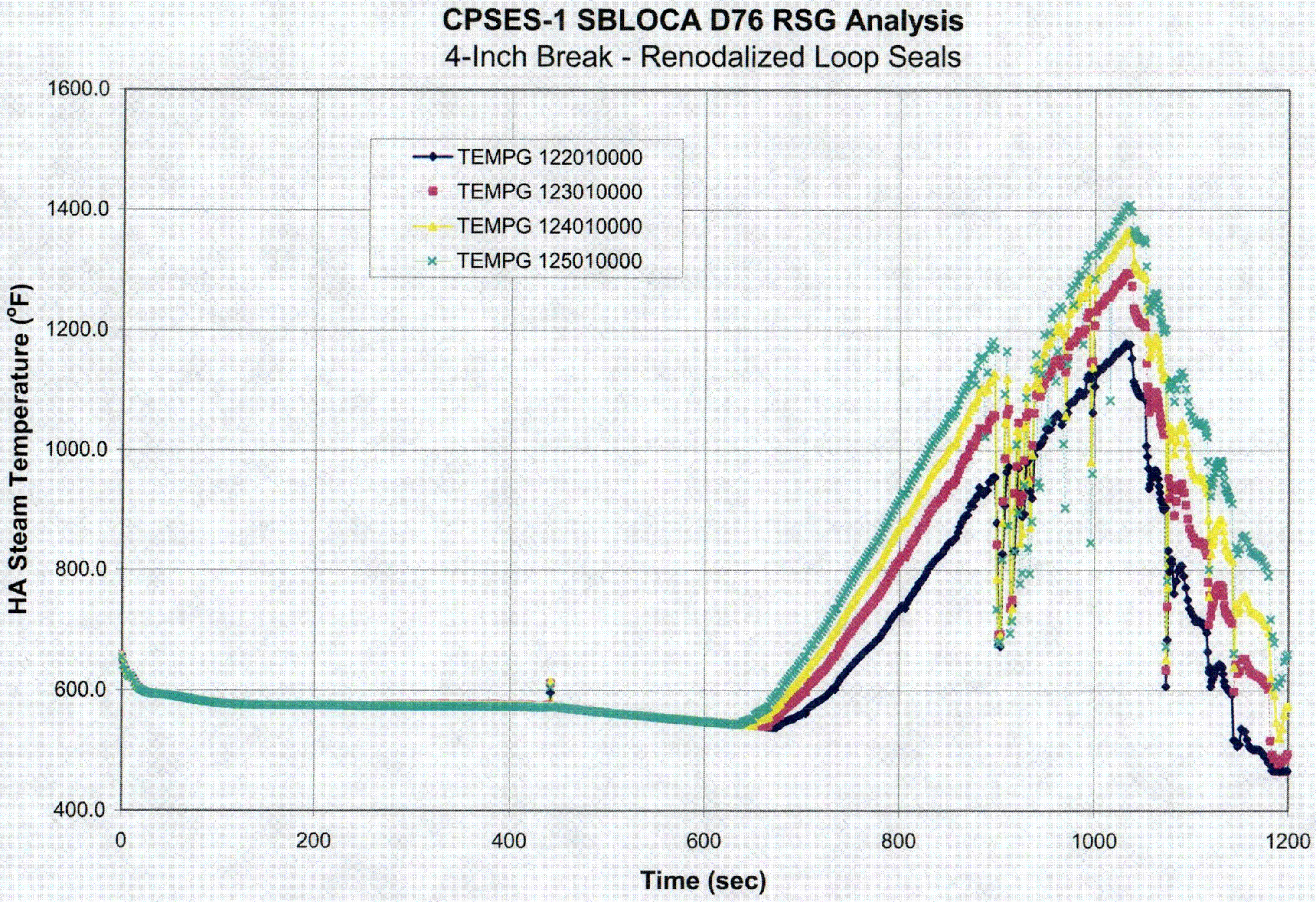

Figure J-15 Hot Assembly Steam Temperatures – 4-in Break (Renodalized Loop Seals)

C108

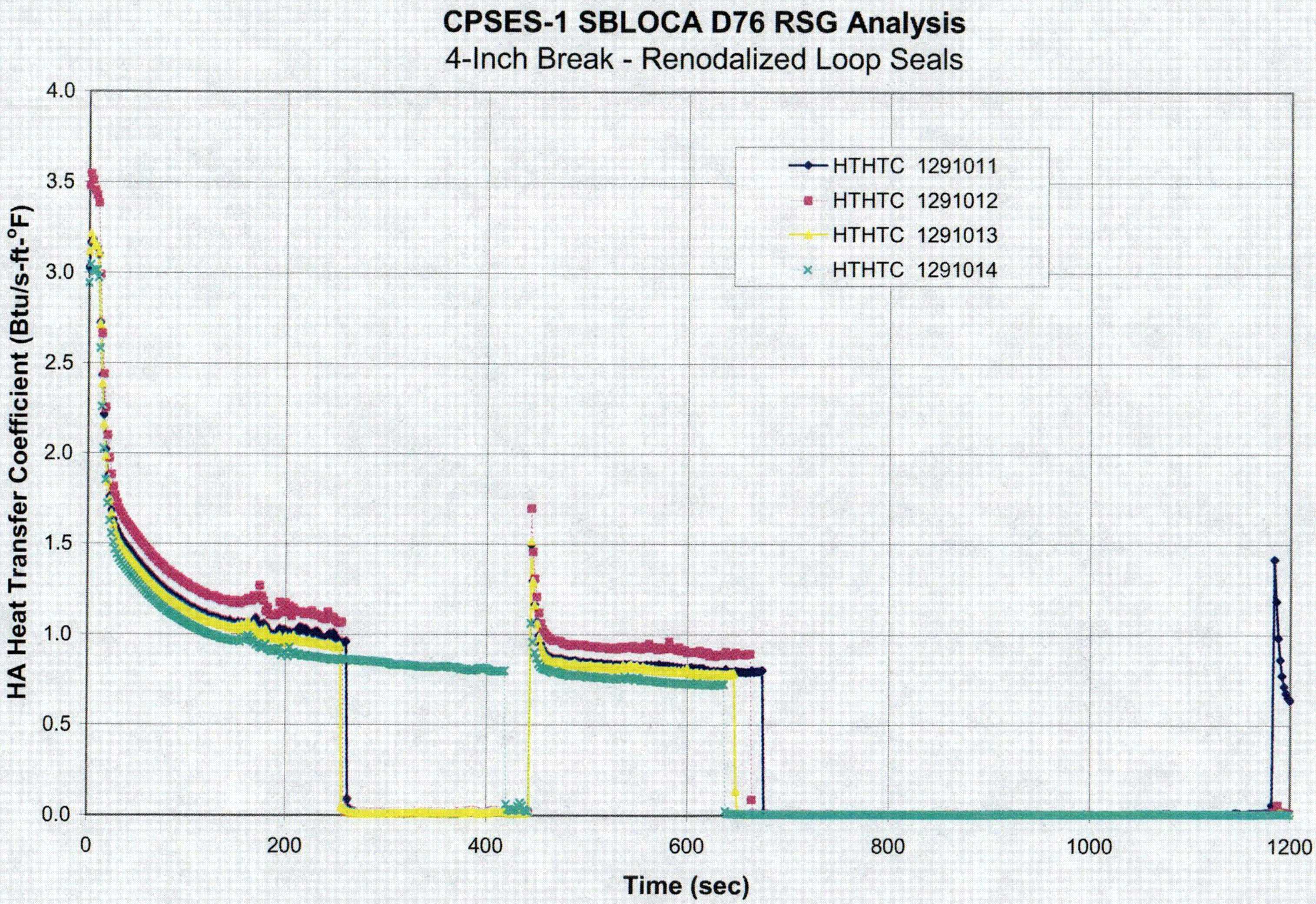

Figure J-16 Hot Assembly Heat Transfer Coefficients – 4-in Break (Renodalized Loop Seals)

C/09

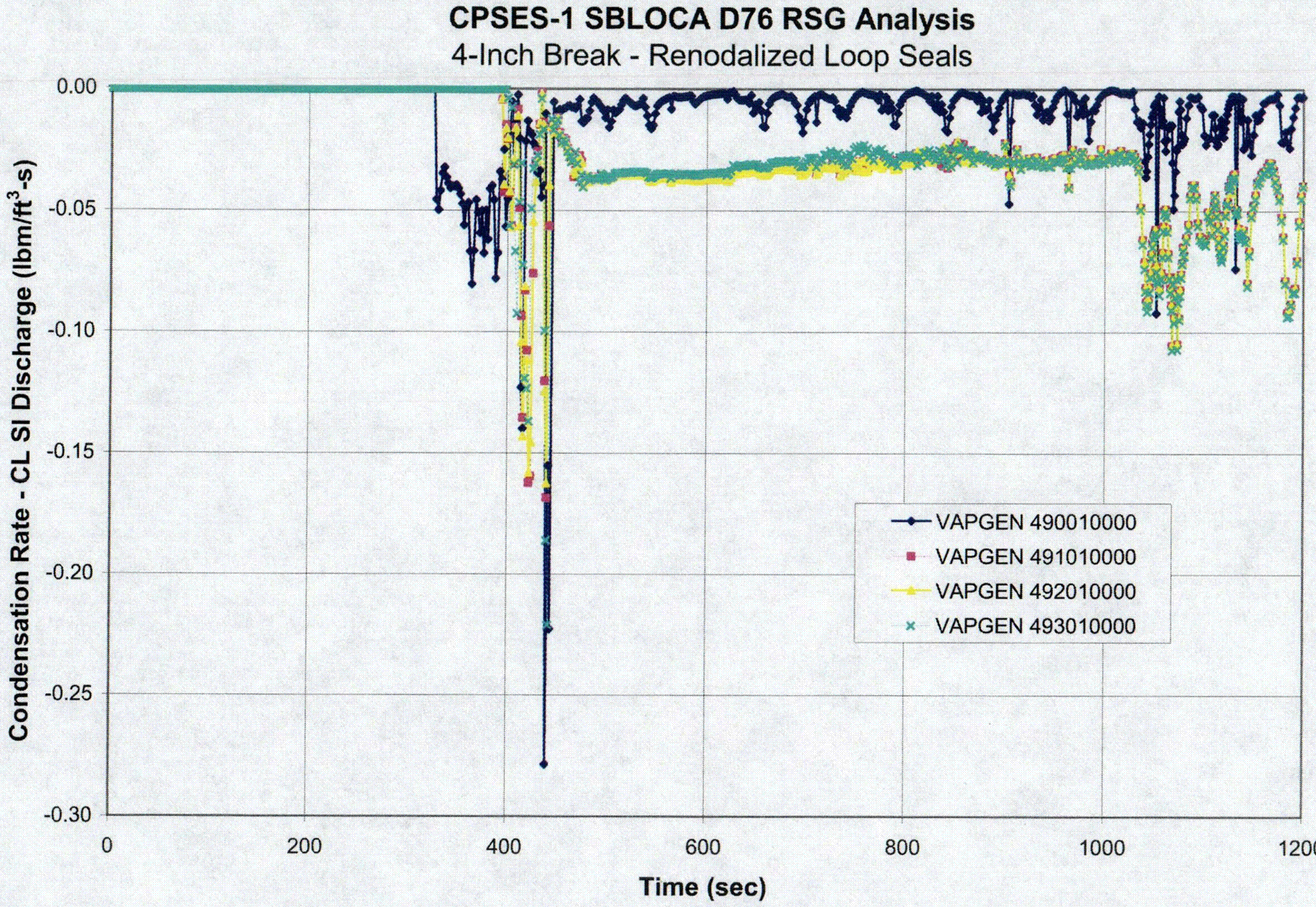

Figure J-17 Condensation Rate in Cold Leg Discharge – 4-in Break (Renodalized Loop Seals)

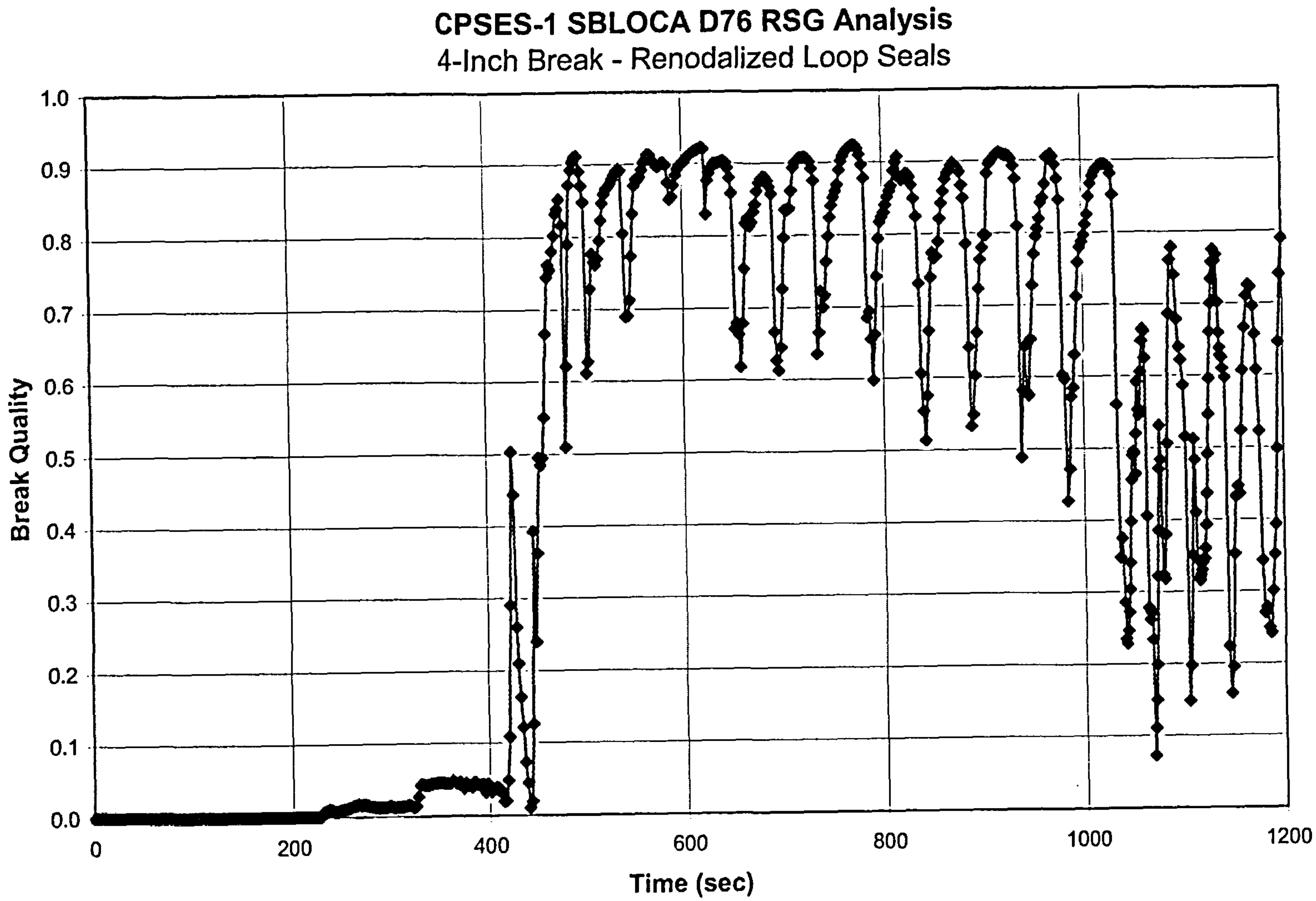

Figure J-18 Break Quality – 4-in Break (Renodalized Loop Seals)

V. MISCELLANEOUS FIGURES

This section contains miscellaneous figures which do not fit under the "complete set" of plots category provided in the previous section. These figures help to address issues within this RAI response package and are as follows:

Figure 1 Axial power profile used in the LOCA analyses

Figure 2 Flow regime history in the loop seals for the limiting base case 4" break (To interpret the flow regime history plot see the table below)

Flow regime	**Three Letter Code** (Major edits)	**Number** (Minor edits/plots)
High mixing bubbly	CTB	1
High mixing transition	CTT	2
High mixing mist	CTM	3
Bubbly	BBY	4
Slug	SLG	5
Annular-mist	ANM	6
Inverted annular	IAN	7
Inverted Slug	ISL	8
Mist	MST	9
Horizontal Stratified	STR	10

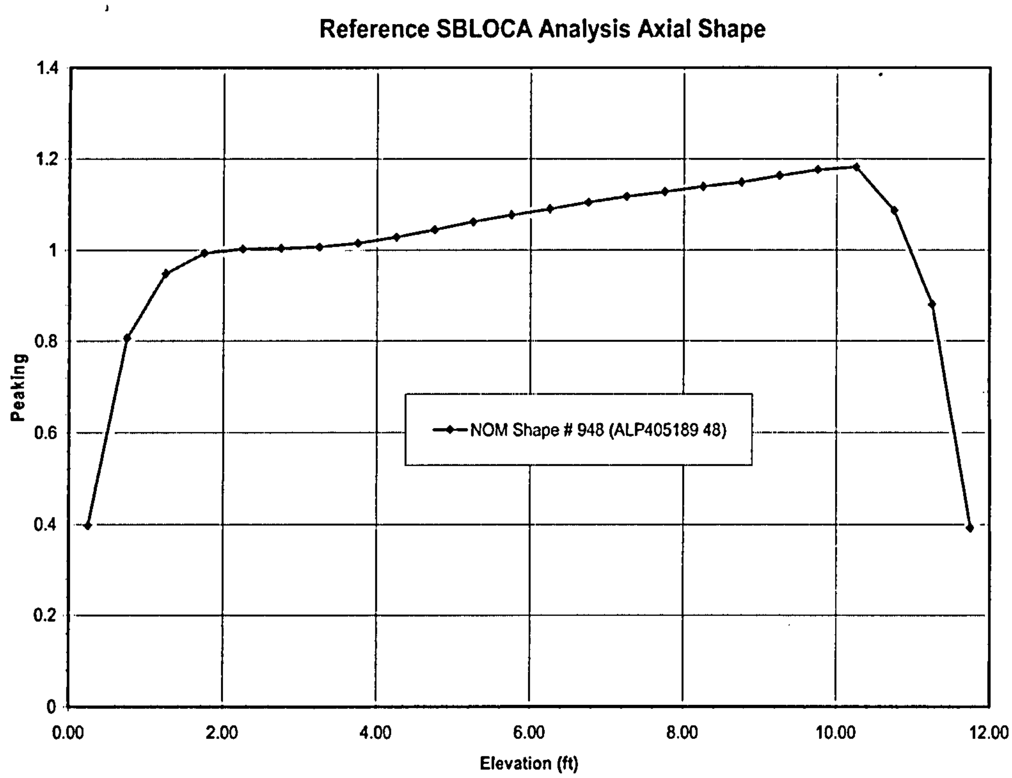

Figure 1 Axial Power Profile Used in the LOCA Analyses

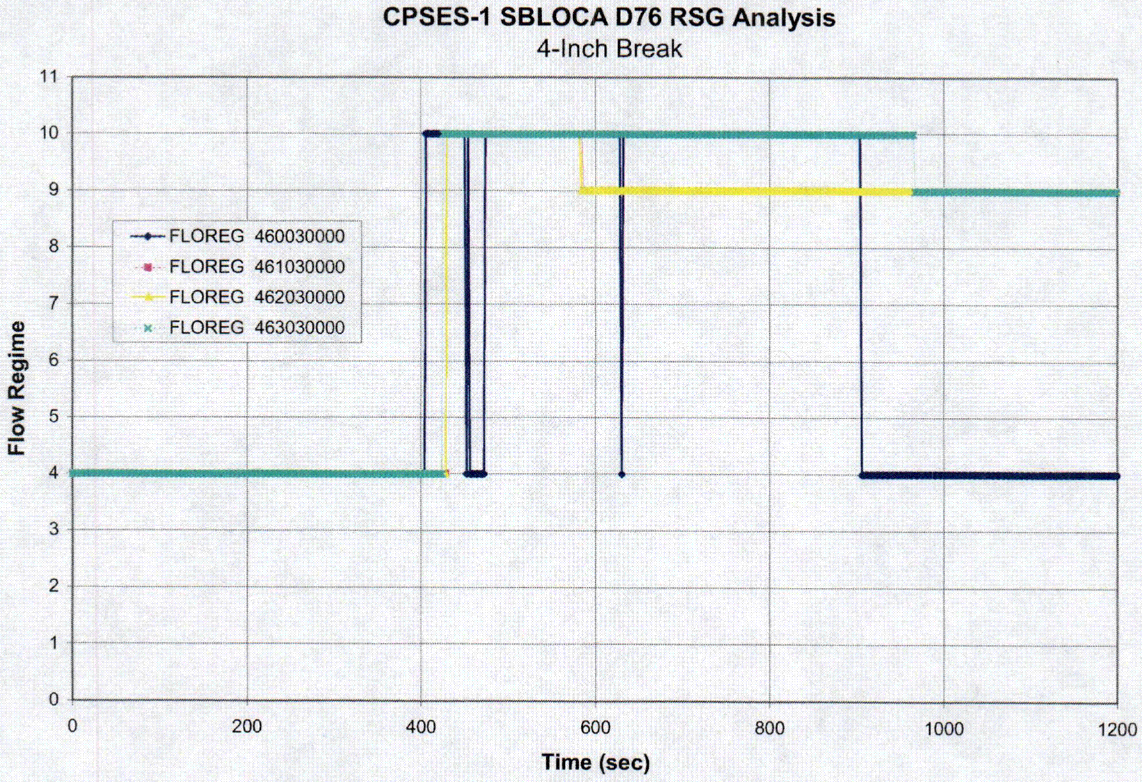

Figure 2 Flow Regime History in the Loop Seals for the Limiting Base Case 4" Break

C11)

ATTACHMENT 2 TO TXX-06125

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

FOR THE REPLACEMENT STEAM GENERATOR SUPPLEMENT TO TXU POWER'S OTHER THAN LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGIES

July, 2006

Question 1:

The licensee is replacing its current multi node steam generator model with the Feedring Steam Generator model from WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," (Ref. 9), figure 3.6-2. In Section 4.1 of ERX-04-005 (Ref. 2) it states, "The Δ76 steam generator model is used by CPSES accident analysis engineers in the same manner as described in Reference 9." Reference 9 of ERX-04-005 is WCAP-14882-P-A. WCAP-14882-P-A contains more than a description of a feedring steam generator model. Use of the feedring steam generator model is not called out separately from the rest of the model described in WCAP-14882-P-A. Therefore, it is difficult to separate the feedring steam generator model from the remainder of WCAP-14882-P-A, and the staff views this as a defacto request to add WCAP-14882-P-A to the licensee's list of approved methodologies. The staff requests the licensee:

a) **Provide information that shows how the Conditions of WCAP-14882-P-A are met.**
b) **With respect to WCAP-14882-P-A Condition 2, either confirm the Westinghouse Δ76 steam generator was part of the data used to develop the feedring steam generator model or provide justification for extending the WCAP-14882-P-A feedring steam generator model to the Westinghouse Δ76 steam generator.**
c) **WCAP-14882-P-A contains a description of how the RETRAN models contained therein are to be used with respect to Non-LOCA transients. Confirm that there are no conflicts with CPSES Unit 1 currently approved or proposed methodologies. If there are any conflicts, provide a resolution and justification for each.**

TXU Power Response:

As described in WCAP-14882-P-A, the Westinghouse accident analysis methods were previously based on the LOFTRAN system thermal-hydraulic computer code. WCAP-14882-P-A contains descriptions of the applicability of the LOFTRAN-based methods to RETRAN-02-based methods. Contained within that report is a description of how Westinghouse would apply RETRAN-02 modeling practices to develop both feed ring and preheat steam generator models. To demonstrate the adequacy of the entire model methodology, not just the steam generator model, Westinghouse developed a plant-specific model of a pilot plant and compared analytical results to actual plant transient results. The NRC approved the modeling methods described in the WCAP-14882-P-A. Although no further plant benchmarking was required, Westinghouse has since applied those same modeling practices to other steam generator feed ring designs (e.g., see Table 1-1). The modeling methodology is generally applicable to vertical U-tube steam generator designs.

The same statement is true of the previously approved CPSES modeling methods. The modeling methods, demonstrated to be valid through comparisons against actual CPSES plant transients, are equally applicable to a feed ring steam generator design as they are to a preheat design. In fact, in the preheat SG design, approximately 15% of the main feedwater flow is diverted through the auxiliary feedwater nozzle. This preheater bypass flow differs only in magnitude from a feed ring SG design.

CPSES engineers chose to use the feed ring steam generator modeling approach used by Westinghouse and described in WCAP-14882-P-A solely to facilitate the NRC review of the steam generator model. The effects of the feed ring and preheat SG designs on the accident analyses are not significantly different, and the CPSES modeling methods remain equally valid (and are, in fact, consistent with the Westinghouse approach). The statement that the model is used "in the same manner" as Reference 9 pertains to, and is limited to, the selection of the inputs (e.g., noding, loss coefficients, etc.) which describe the steam generator model.

CPSES is not requesting to add WCAP-14882-P-A to the list of approved methodologies for determining the Core Operating Limits. The methods by which the FSAR Chapter 15 accident analyses are performed are controlled by the analytical methods described in Technical Specification 5.6.5b, as supplemented by ERX-04-005. The use of the Δ76 steam generator model does not affect the methodology used to analyze the non-LOCA FSAR Chapter 15 accident analyses. The inputs, sensitivity studies, and selection of conservative assumptions are performed in the same manner. The differences are related to the geometric differences, such as location of the feedwater nozzle (important for the feedwater line break analysis), primary and secondary system volumes, heat transfer areas, and internal pressure drops.

With this basis for the relationship between ERX-04-005 and WCAP-14882-P-A and the methodology used to perform the FSAR Chapter 15 non-LOCA accident analyses, explicit responses to sub-Questions a, b, and c are not required.

Table 1.1 Recent applications of WCAP-14882-P-A Steam Generator Modeling Methodology

Plant	Steam Generator Model (All feed ring designs)	ADAMS Accession Number
South Texas Project	Delta 94	ML031400401
Ginna	Model 44	ML051960044
St. Lucie	Model 67 (CE design)	ML050120363
Prairie Island	Model 51	ML040680396
Indian Point 2	Model 44	ML042960007
Indian Point 3	Model 44	ML050600380
Seabrook	Model F	ML040860307
Callaway	73/19T (B&W design)	ML052570054

Question 2.

In Section 4.1 of ERX-04-005 (Ref. 2) it states, "This model was validated through steady-state comparisons to detailed design data provided by the vendor. In addition to the comparisons with the detailed vendor data, the integrated Δ76 system model was used to analyze selected Non-LOCA accident analyses; the results were compared to the results obtained using the currently approved models with the D-4 steam generator design. These comparisons demonstrate that the incorporation of the Δ76 model did not affect other parts of the qualified model."

a. **Provide the vendor design data and model results that were used to validate the model's ability to reproduce steady-state conditions. Charts and tables are acceptable.**
b. **Explain what evidence provides reasonable assurance the model will accurately reproduce transient conditions at CPSES.**
c. **Given the differences in design and modeling between the D-4 and Δ76 steam generators, the staff believes comparing the results of the two sets of analyses is problematic and may not be a true evaluation of the new modeling and methodology. Please provide comparisons of Δ76 steam generator analysis to those of similar design and size.**

TXU Power Response:

Three types of validation evaluations were performed to assess the adequacy of the Westinghouse-supplied model of the CPSES Δ76 SG model. A stand-alone model of the steam generator was first used to gain confidence in the model itself, as well as familiarity with the characteristics of the Δ76 SG design. The stand-alone model is essentially the same as described in the response to Question 1; the node and junction numbering scheme was modified for compatibility with the CPSES integrated model.

This stand-alone Δ76 SG model was then integrated with the RETRAN-02 model of CPSES. Null transients were run to ensure the Δ76 SG model was properly integrated. Finally, several transients were simulated, using the licensing methods described in RXE-91-001-A, and the results were compared against similar analyses using the existing D-4 SG model to ensure the differences in the results were consistent with the differences in the physical SG designs.

a. The data in the Table 2-1 is a summary of a comparison of relevant design data from the vendor's design tool and the RETRAN-02-based stand-alone Δ76 SG model for full power, steady-state operating conditions. As may be observed, the results from the full power RETRAN-02 stand-alone model compare quite well to the vendor-supplied design data.

Table 2-1. Steady-state data comparisons for CPSES Δ76 steam generator model.

	Δ76 SG for Unit 1	
	Design	*RETRAN*
Total Mass (lbm)	111303	110587
Liquid Mass (lbm)	104164	103572
Vapor Mass (lbm)	7139	7015
Recirculation Loop Pressure Drop (psi)	6.86	6.66
Circulation Ratio	4.1	4.1

Additional evaluations were then performed to assess the adequacy of the model for conditions other than full power operations. Forcing functions for the Reactor Coolant System flow rate and hot leg temperature and the main feedwater flow rate and temperature were used with the stand-alone steam model to simulate a gradual downpower transient. Design data at various power levels were extracted from design input data provided by the vendor specifically for this purpose. Comparisons of relevant parameters are presented in Figures 2-1 through 2-5. As may be observed, the stand-alone Δ76 steam generator model provides an adequate representation of vendor-supplied design data throughout the range of normal operating conditions.

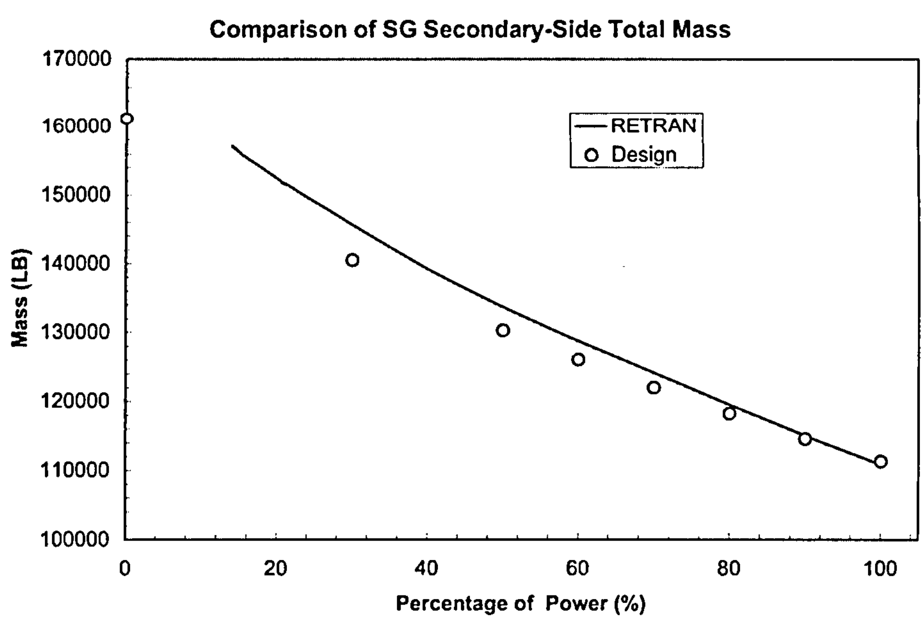

Figure 2-1. Comparison of Δ76 design parameters – Total Secondary Side Mass

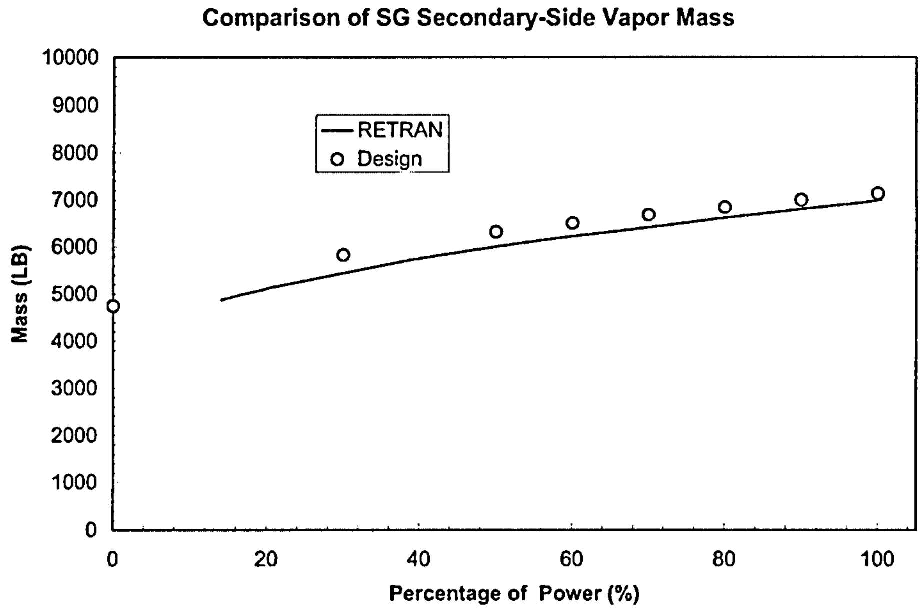

Figure 2-2. Comparison of Δ76 design parameters – Secondary Side Vapor Mass

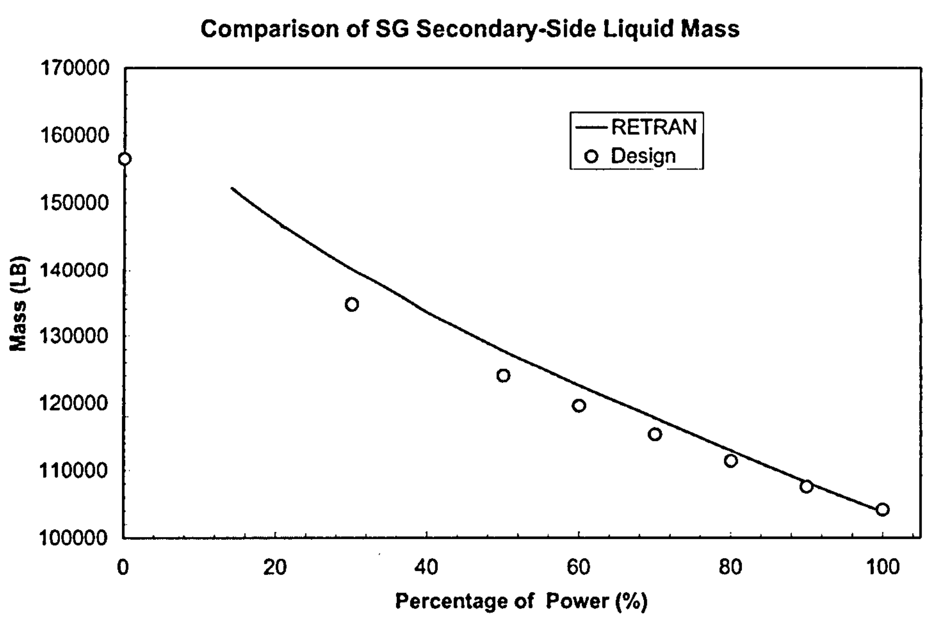

Figure 2-3. Comparison of Δ76 design parameters – Secondary Side Liquid Mass

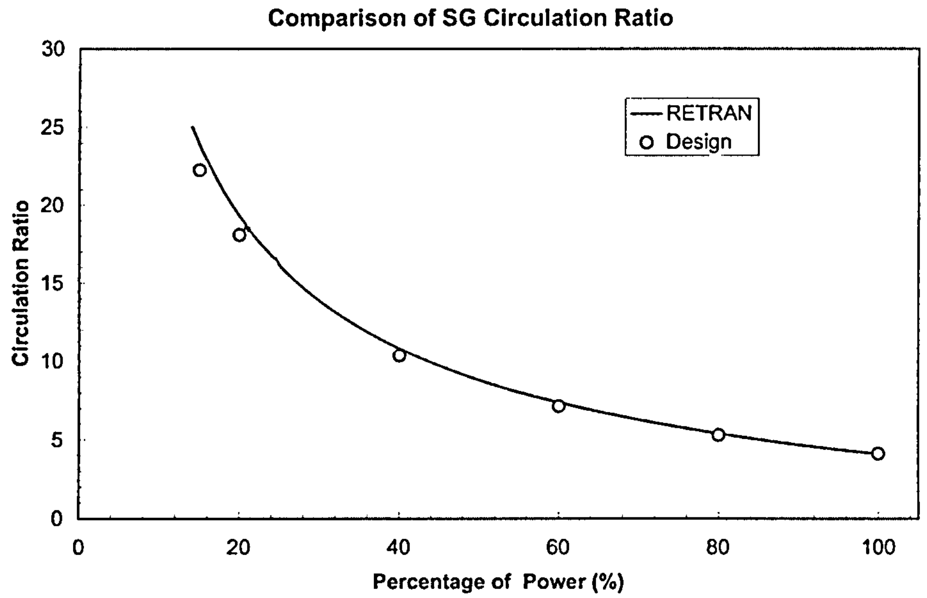

Figure 2-4. Comparison of Δ76 design parameters – Secondary Side Circulation Ratio

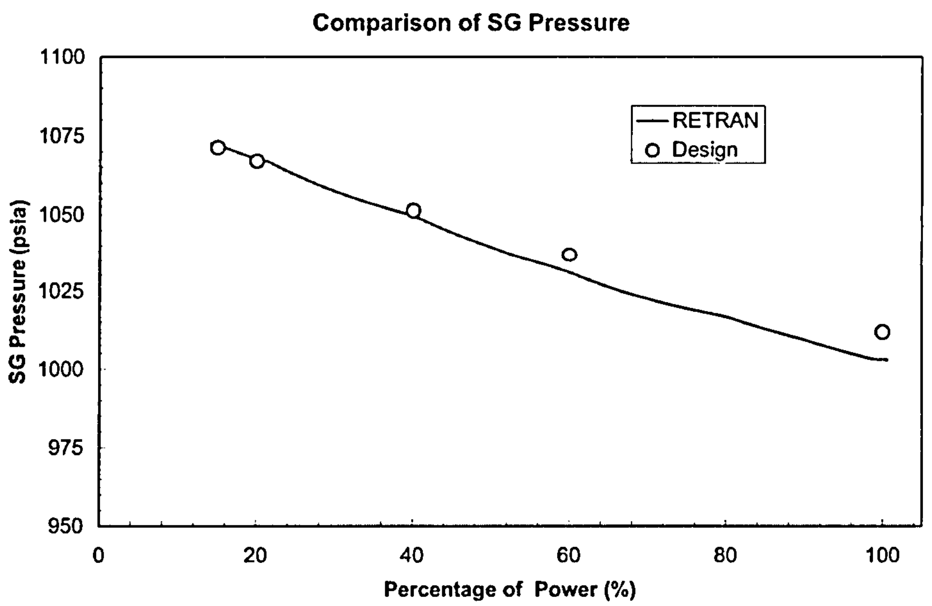

Figure 2-5. Comparison of Δ76 design parameters – Secondary Side Pressure

b. Reasonable assurance that the Δ76 SG model adequately reproduces transient conditions at CPSES is provided through the totality of the evaluations performed by CPSES engineers.

1. As shown in the referenced topical report RXE-91-001-A, CPSES engineers performed an extensive set of plant benchmarks to demonstrate the adequacy of the integrated CPSES model (including the D4 steam generator model) for use in performing the FSAR Chapter 15 transient and accident analyses. This model has been used extensively and continuously by CPSES engineers to support reload core configurations, evaluate plant design modifications, and other licensing and operations related activities n the years since the methods were first approved.
2. The D4 RETRAN-02 SG model was developed using approved methodologies (RXE-91-001-A).
3. The Δ76 RETRAN-02 SG model developed using approved methodologies (WCAP-14882-P-A).
4. CPSES engineers have developed a thorough understanding of the physical differences between the D-4 and Δ76 SG designs, as well as the model differences between the D-4 and Δ76 SG RETRAN-02 models, and of the effects of these differences on the FSAR Chapter 15 transient and accident analyses. The fundamental performance characteristics of the steam generator are not significantly different between the D4 and Δ76 designs. For example, if a main steam safety valve is assumed to open, or feedwater temperature is decreased, there will be no significant difference in the amount of energy removed from the Reactor Coolant System; the timing of the transient may change due to fluid inventories or heat transfer areas, but the magnitude of the transient is not significantly affected.
5. Comparisons and evaluations of FSAR Chapter 15 transient and accident analyses results, using the D-4 and Δ76 SG designs to ensure differences were consistent with understanding of physical and model differences. (Comparisons and evaluations of two typical transient analyses are presented in the response to Question 5.)

This approach provides a good basis for the evaluation of the adequacy of the Δ76 steam generator model under transient conditions. In the larger sense, only one parameter in the integrated RETRAN-02 model of CPSES is changing – the steam generator model and the CPSES engineers have a good understanding of how the results should be affected based on this single change.

In summary, the model qualification approach described in ERX-04-005 provides reasonable evidence of the quality of the model. The existing CPSES model, which includes the D-4 steam generator, has been thoroughly benchmarked to plant data, used extensively, and is well-understood by the CPSES engineers. The differences between the D-4 and Δ76 steam generator designs are also well-understood. In this situation, the comparison of the results from the D-4 and Δ76 analyses provides reasonable assurance that the model can be used to conservatively assess the transients and accidents described in FSAR Chapter 15.

c. Because there are no other Westinghouse four-loop plants with the Δ76 steam generator design, the data necessary to provide definitive comparisons of the CPSES model with the Δ76 steam generator design to plant transient data is not available. However, the CPSES Δ76 steam generator design is encompassed by the steam generator modeling methodology developed and benchmarked against plant data as described in WCAP-14882-P-A.

Recent applications of the WCAP-14882-P-A steam generator modeling methodology to various feed ring steam generator designs were listed in Table 1-1. Some of the geometrical and performance characteristics of selected steam generator designs, including the CPSES Δ76 steam generator design, are presented, for comparative purposes, in Table 2-2. As may be observed, the CPSES Δ76 steam generator design and performance characteristics are well within the database of steam generator designs and performance characteristics for which the steam generator modeling methodology has been approved. A relevant conclusion from this observation is that the steam generator modeling approach described in WCAP-14882-P-A, including such considerations as the selection of node boundaries and junction elevations are directly applicable to the CPSES Δ76 steam generator design. Therefore, there is reasonable assurance that the CPSES RETRAN-02 Δ76 steam generator model adequately represents the performance of the steam generator.

Table 2-2. Geometrical and Performance Characteristics of steam generator designs modeled with WCAP-14882-P-A methodology

	SG Model	Power (Mwt)	RCS Loop Flow (gpm)	Tavg (F)	Tsteam (F)	Plug (%)	# of Tubes	Area (ft2)	Circ Ratio
Beaver Valley	54F	970.0	87200	580	514.8	0	3592	54500	3.3
	54F	970.0	87200	580	505.7	22	3592	54500	3.3
	54F	970.0	87200	566.2	500.5	0	3592	54500	3.3
	54F	970.0	87200	566.2	490.2	22	3592	54500	3.3
Callaway	73/19T	894.8	93600	588.4	547.2	0	5626	78945	4.0
	73/19T	894.8	93600	588.4	546.5	5	5626	78945	4.0
	73/19T	894.8	93600	570.7	528.3	0	5626	78945	4.0
	73/19T	894.8	93600	570.7	527.5	5	5626	78945	4.0
Diablo Canyon	Delta 54	856.3	88500	577.6	521.7	0	4444	54240	3.9
	Delta 54	856.3	88500	577.6	518.2	10	4444	54240	3.9
South Texas	Delta 94	968.5	98000	592.6	551.4	0	7585	94500	3.9
	Delta 94	968.5	98000	592.6	549.9	10	7585	94500	3.9
	Delta 94	968.5	98000	582.7	540.7	0	7585	94500	3.9
	Delta 94	968.5	98000	582.7	539.2	10	7585	94500	3.9
St. Lucie	Model 67	1360	167500	576.5	526.6	30	8411	90232	4.0
Farley	Delta 54	928.3	86000	577.2	515.5	0	3592	54500	3.6
	Delta 54	928.3	86000	577.2	506.9	20	3592	54500	3.6
Summer	Delta 75	970.7	92600	587.4	540.4	0	6307	75185	3.3
	Delta 75	970.7	92600	587.4	538.4	10	6307	75185	3.3
	Delta 75	970.7	92600	572	523.7	0	6307	75185	3.3
	Delta 75	970.7	92600	572	521.7	10	6307	75185	3.3
CPSES-1	Delta 76	895.5	95700	589.2	545.8	0	5532	76000	4.1
	Delta 76	895.5	95700	589.2	543.8	10	5532	76000	4.1
	Delta 76	895.5	95700	574.2	529.6	0	5532	76000	4.1
	Delta 76	895.5	95700	574.2	527.5	10	5532	76000	4.1

Question 3.

In Section 4.1 of ERX-04-005 (Ref. 2) it states, "The specifics of the steam generator model to be used in each of the transient and accidents analyses is described in each of the cited methodology special reports, although, the use of alternate models is permitted if justified through sensitivity studies. In several of those transients and accident analyses, where only the gross steam generator performance is of importance, a single-node steam generator was justified as sufficient." However, the NRC's approval of RXE-91-001-A (Ref. 4) explicitly stated, ". . . it is acceptable for the licensee to use either the three-node or four-node steam generator model in using the TU Electric analysis methodology." Any discussion of the use of a 'sensitivity analysis' was limited to selecting either the three-node or four-node steam generator model described in RXE-91-001-A and not replacing them with a single node model.

a. **Please address this apparent discrepancy.**

b. **Identify which transients are being analyzed using a single node steam generator model.**

c. **Please provide the justification for the use of a single node steam generator model for those transients.**

TXU Power Response:

a. There is no discrepancy related to the use of a single-node steam generator model for selected events. As stated in the NRC's Safety Evaluation Report for RXE-91-001-A, "TER Sections 1.0 and 4.0 identify that application of the TU Electric transient analysis methodology to evaluation of inadvertent opening of a steam generator safety valve, steam system piping failure, steam generator tube rupture, and power distribution anomaly events are not within the scope of this review." Thus, the three-node or four-node steam generator model acceptance is not applicable to these transient and accident analyses.

b. A single node steam generator model is used for the inadvertent opening of a steam generator safety valve, the steam system piping failure, and the steam generator tube rupture analyses. Although a simpler model is allowed by the NRC approval of RXE-91-002-A, "Reactivity Anomaly Events Methodology," in practice, the multi-node steam generator model is used for the power distribution anomaly events.

c. The use of a single node steam generator model for the inadvertent opening of a steam generator safety valve and steam system piping failure analyses is specifically described, justified, and approved in topical report RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events." As stated in the NRC's Safety Evaluation Report for this topical report, "For the SLB analysis, the SG secondary side model is represented by a single saturated volume to allow for a constant steam exit quality of 100 percent throughout the transient." The rationale for the use of single node steam generator model is unaffected by the incorporation of the Δ76 steam generator model.

The use of a single node steam generator model for the steam generator tube rupture analysis is explicitly detailed in Section 5.3 of the topical report transmitted via TXX-88306. This model was initially approved along with the analysis in NUREG-0797, SSER 23.The rationale for the use of single node steam generator model is unaffected by the incorporation of the Δ76 steam generator model.

Question 4.

In Section 4.1 of ERX-04-005 (Ref. 2) it states, "A single-node RETRAN-02 representation of the Δ76 steam generator design can be easily adapted using the specific geometry of the Δ76 steam generator design. This model is not significantly different from the D-4 steam generator model; the Δ76 model volumes, junctions and heat conductors are simply representative of the Δ76 dimensions. No additional qualification of this model is considered necessary." The staff disagrees that 'no additional qualification' of the single node model is necessary. For those transients identified in response to question 3, provide sufficient information for the staff to reach a reasonable assurance conclusion that the single node RETRAN-02 representation of the Δ76 steam generator design is sufficiently accurate for the transients being analyzed.

TXU Power Response:

The use of a single-node steam generator model has been previously justified, for specific transients, and those justifications accepted by the NRC in the cited topical reports. The manner in which a single-node model is used is unaffected by the physical differences between the D-4 and Δ76 SG designs. See the response to Q3.

Question 5.

In Section 3.3 of ERX-04-005 (Ref. 2) the licensee provides a qualitative discussion of the impact of the Δ76 steam generator on the Non-LOCA transients. The licensee repeatedly uses the statement, "The incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses." Given the responses to RAI Questions 1 and 3 for RXE-91-001-A the staff would expect the potential for a significant difference in transients in which feedwater flow or temperature is affected. Given the differences in design and modeling between the D-4 and Δ76 steam generators, the staff believes comparing the results of the two sets of analyses is problematic and may not be a true evaluation of the new modeling and methodology. To determine the adequacy of existing methodologies, provide a quantitative discussion of the impact of the Δ76 steam generator on the Non-LOCA transients. Include how the transient progresses from start to finish.

TXU Power Response:

Two representative transients were selected to provide the requested evaluation. The "decrease in main feedwater temperature" transient results in an overcooling of the Reactor Coolant System (RCS) by the secondary system. The turbine trip transient is a rapid degradation of the heat removal capability of the secondary system, resulting in a rapid pressurization of both the RCS and secondary systems. These two transients were selected to support the observation that the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analysis. As noted in ERX-04-005, the exception to this observation is the analysis of the feedline break accident. In that analysis, the different location of the postulated break (at the main feedwater line nozzle), results in a transient response that resembles a steamline break response more so than the existing feedline break analytical results. (The feedline break accident analysis with the Δ76 steam generator model is shown in Chapter 4 of ERX-04-005.)

Decrease in Main Feedwater Temperature

The Decrease in Main Feedwater Temperature transient analysis is described in FSAR Section 15.1.1. The relevant acceptance criterion for this ANS Condition II event is compliance with the minimum Departure from Nucleate Boiling Ratio. This transient is assumed to be initiated by a failure which isolates a low pressure feedwater heater. Consequential system responses include the trip of the heater drain pumps and isolation of extraction steam, resulting in the loss of essentially all feedwater heating. In the current analyses with the D-4 steam generator design, the relatively cold main feedwater comes into direct contact with the U-tubes in the preheater region and thus, has an almost immediate effect on the RCS. Due to the moderator temperature feedback, the reactor power begins to increase until the overpower N-16 reactor trip setpoint is reached. In the analyses with the Δ76 steam generator design, the relatively cold main feedwater first mixes with the saturated recirculating fluid in the upper downcomer region of the steam generator. This fluid, now slightly subcooled, then circulates through the downcomer annulus to the entrance of the tube bundle. As such, the effect on the RCS is manifested more slowly than for the analysis performed with the D-4 steam generator design.

A comparison of the Sequence of Events for these two analyses is provided in Table 5-1. Comparative plots of relevant parameters are shown in Figures 5-1 through 5-5. The same trends are observed in each analysis. The effects of the larger heat transfer area of the Δ76 steam generator model are apparent in the initial conditions shown in Figure 5-5, where, for the same RCS average temperature, the steam generator secondary side pressure is higher.

As previously noted, the effects of the feedwater temperature decrease in the Δ76 steam generator design lag the effects seen with the D-4 steam generator design. The rate of power increase is faster in the D-4 analysis, resulting in a slightly higher overshoot in the reactor power between the time the overpower reactor trip setpoint is reached and control rods begin to drop into the reactor core.

<u>Turbine Trip</u>

The Turbine Trip transient analysis is described in FSAR Section 15.2.3. The relevant acceptance criteria for this ANS Condition II event are compliance with the minimum Departure from Nucleate Boiling Ratio, peak RCS pressure, and peak steam generator pressure. Separate analyses are performed with conservative assumptions and initial conditions selected to minimize the margin to a particular event acceptance criterion. For the purposes of this evaluation, the analysis that results in the most significant challenges to the RCS and steam generator pressures is presented. This transient is assumed to be initiated by a turbine trip signal; no credit is taken for the automatic reactor trip on turbine function. Following the turbine trip, and the assumed coincident loss of main feedwater flow, the heat removal rate from the RCS is quickly degraded. Both the RCS and secondary systems rapidly pressurize to and beyond the code safety valve set pressures. The transient is terminated following the insertion of the control rods upon receipt of a high pressurizer pressure reactor trip signal and the opening of the safety valves to relieve the overpressure condition. Due to the rate of RCS pressurization, and the finite time required to purge the loop seals on the pressurizer safety valve seats before significant pressure relief occurs, the peak RCS pressure (at the reactor coolant pump discharge) overshoots the pressurizer safety valve set pressure and is predicted to closely approach the event acceptance criterion of 110% of the RCS design pressure. The rate of the secondary system pressurization also results in a significant challenge to the event acceptance criterion; although, there is more margin between the peak pressure and the acceptance criterion (also 110% of the design pressure).

Because, during full power operation, the main feedwater flow makes direct contact with the U-tubes in the preheater region of the D-4 steam generator design, the interruption of the feedwater flow has an immediate impact on the RCS heat removal rate. In contrast, in the Δ76 steam generator design, the feedwater first mixes with the recirculating fluid in the upper downcomer region before flowing to the U-tubes, and the effects on the RCS of an interruption of the feedwater flow are seen at a slightly later time. Thus, the rate of RCS pressurization is slightly greater for the D-4 steam generator design, which is reflected in the slightly greater pressure overshoot.

A comparison of the Sequence of Events for the turbine trip – overpressurization analyses performed with the D-4 steam generator model and the Δ76 steam generator model is provided in Table 5-2. Comparative plots of relevant parameters are shown in Figures 5-6 through 5-10. The same trends are observed in each analysis. The effects of the larger heat transfer area of the Δ76 steam generator model are apparent in the initial conditions shown in Figure 5-10, where, for the same RCS average temperature, the steam generator secondary side pressure is higher.

(Note that sensitivity studies were performed, and it was concluded that for the turbine trip – overpressure analysis, the RCS pressure acceptance criterion is slightly more closely approached if the model is initialized at the nominal value of RCS average temperature (T_{avg}) less uncertainties. This observation is in contrast to the feedwater malfunction analysis, where it was shown that adding the T_{avg} temperature uncertainties was more limiting. This difference is reflected in the RCS temperatures shown in Figures 5-3 and 5-8, as well as in the steam generator pressures shown in Figures 5-5 and 5-10.)

Similar to the feedwater malfunction transient analysis, the effects of the turbine trip and the termination of the main feedwater flow in the Δ76 steam generator design lag the effects seen with the D-4 steam generator design. The rate of both the RCS and steam generator pressure increase is faster in the D-4 analysis, resulting in a slightly higher overshoot in the RCS pressure between the time the high pressurizer pressure reactor trip setpoint is reached, the safety valve set pressures are reached, and the transient is terminated by the control rods dropping into the reactor core and the safety valves open to relieve the overpressure condition.

<u>Summary</u>

Based on the forgoing analysis results, representative of secondary system – induced transients, it may be observed that the incorporation of the Δ76 steam generator design does not result in a transient response that is significantly different from the current CPSES analyses. The one exception to this observation, as noted in ERX-94-005, is the analysis of the main feed line break accident, wherein the location of the postulated break (at the main feedwater nozzle), leads to a very different transient.

Table 5-1 Time Sequence of Events for Feedwater Malfunction

Action	Time (Sec)	
	Δ76	D4
Start Problem/Steady State Initial Conditions	0.0	0.0
Initiate Feedwater Temperature Decrease	5.0	5.0
Overpower N-16 Setpoint Reached	20.8	14.7
Reactor Trip	22.8	16.7
Minimum DNBR Reached	23.1	17.1
End Problem	32.8	26.7

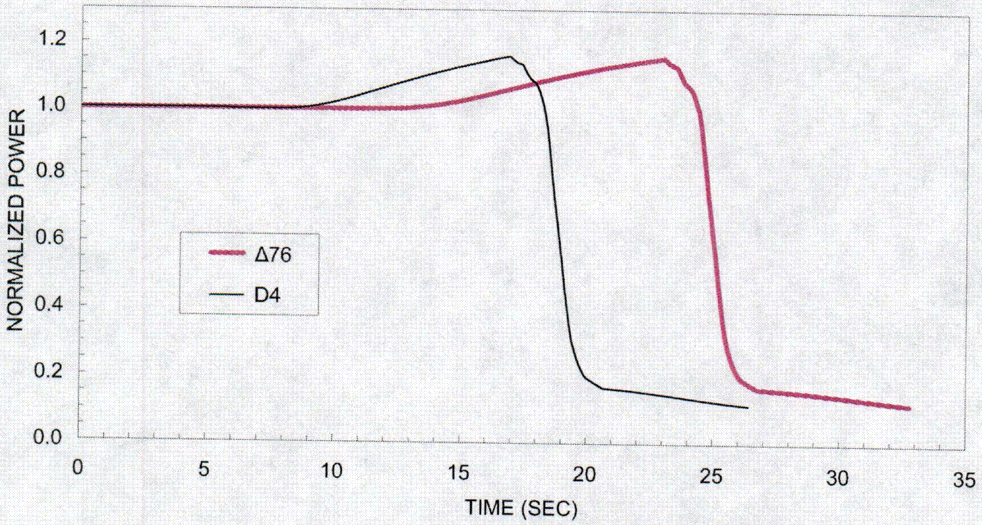

Figure 5-1. Comparison of results for an increase in feedwater flow transient with Δ76 and D4 steam generator designs – normalized reactor power

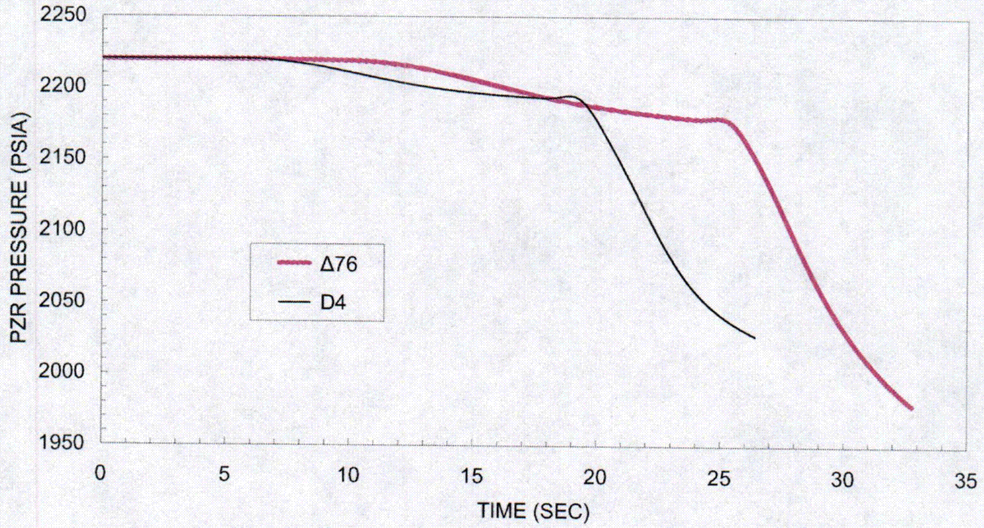

Figure 5-2. Comparison of results for an increase in feedwater flow transient with Δ76 and D4 steam generator designs – pressurizer pressure

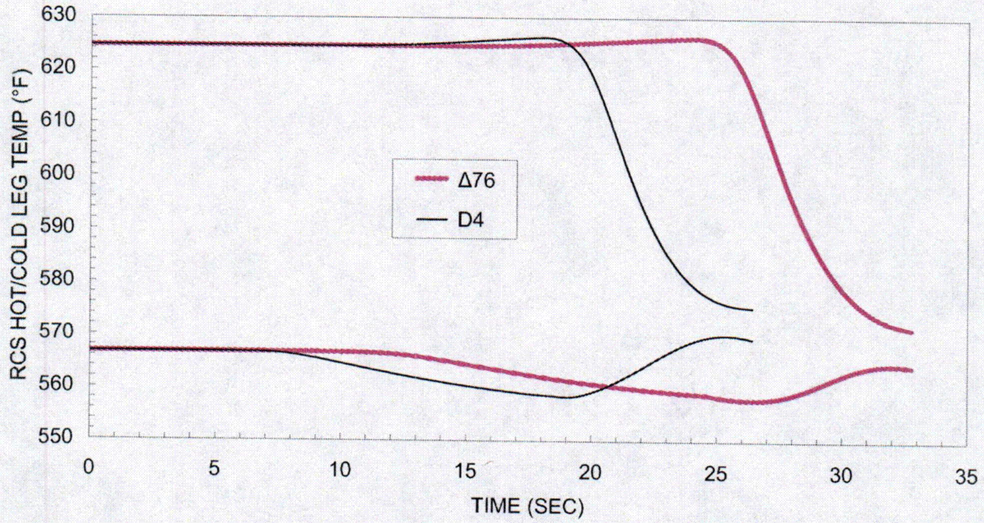

Figure 5-3. Comparison of results for an increase in feedwater flow transient with Δ76 and D4 steam generator designs – RCS hot and cold leg temperatures

C113

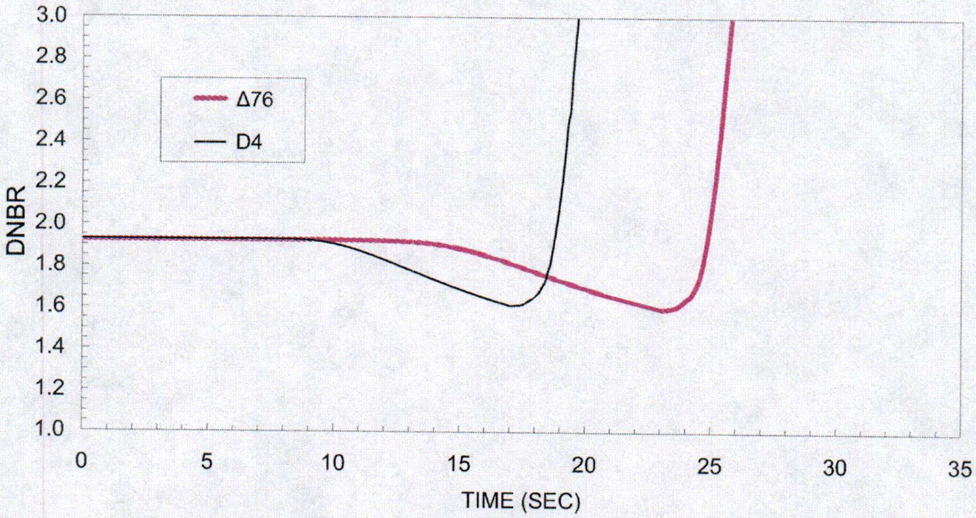

Figure 5-4. Comparison of results for an increase in feedwater flow transient with Δ76 and D4 steam generator designs – Departure from Nuclear Boiling Ratio

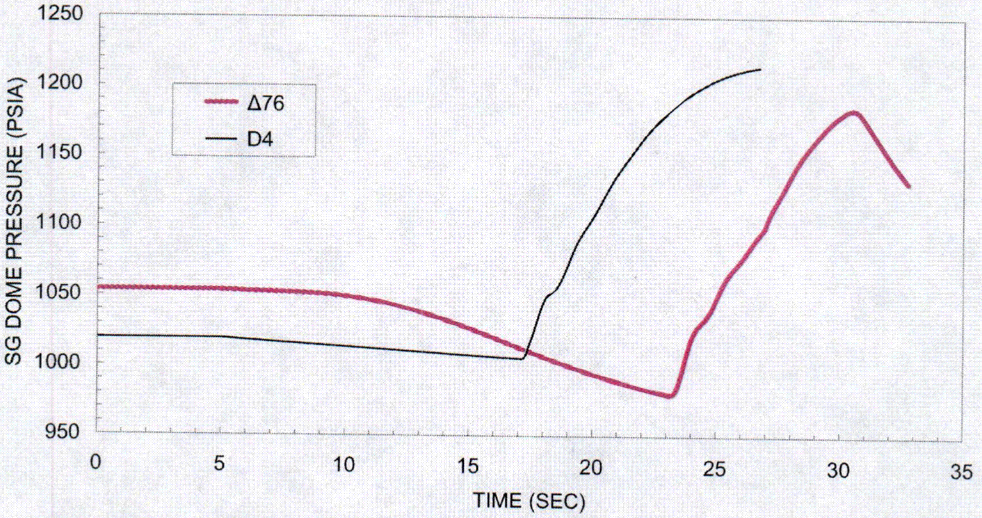

Figure 5-5. Comparison of results for an increase in feedwater flow transient with Δ76 and D4 steam generator designs – steam generator pressure

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Table 5-2. Time Sequence of Events for Turbine Trip

Action	Time (Sec)	
	Δ76	D4
Start Problem/Steady State Initial Conditions	0.0	0.0
Initiate Turbine Trip	1.0	1.0
All MFW Isolation Valves Close	1.0	1.0
High Pressurizer Pressure Reactor Trip Signal	7.4	7.0
PSVs Open Setpoint Reached	8.3	7.8
Reactor Trip	9.4	9.0
Maximum RCS Pressure Reached	9.8	9.3
Open MSSV-1	12.9	13.4
Open MSSV-2	13.8	14.2
Open MSSV-3	16.3	NA
End Problem	19.4	19.0

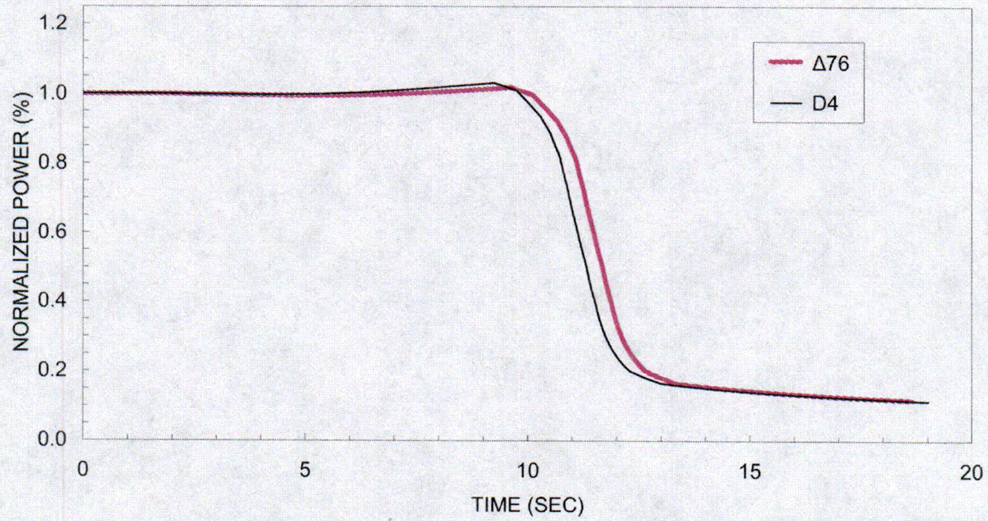

Figure 5-6. Comparison of results for a turbine trip transient with Δ76 and D4 steam generator designs – normalized reactor power

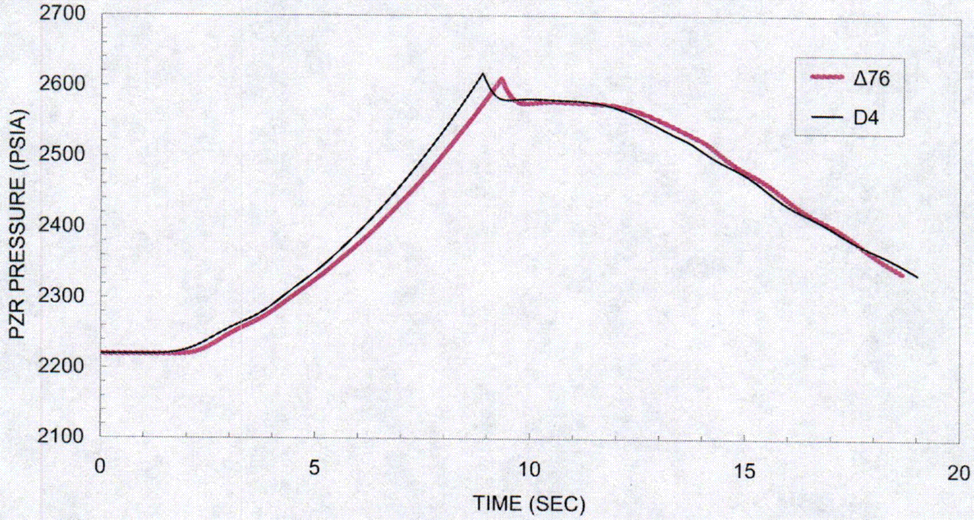

Figure 5-7. Comparison of results for a turbine trip transient with Δ76 and D4 steam generator designs – pressurizer pressure

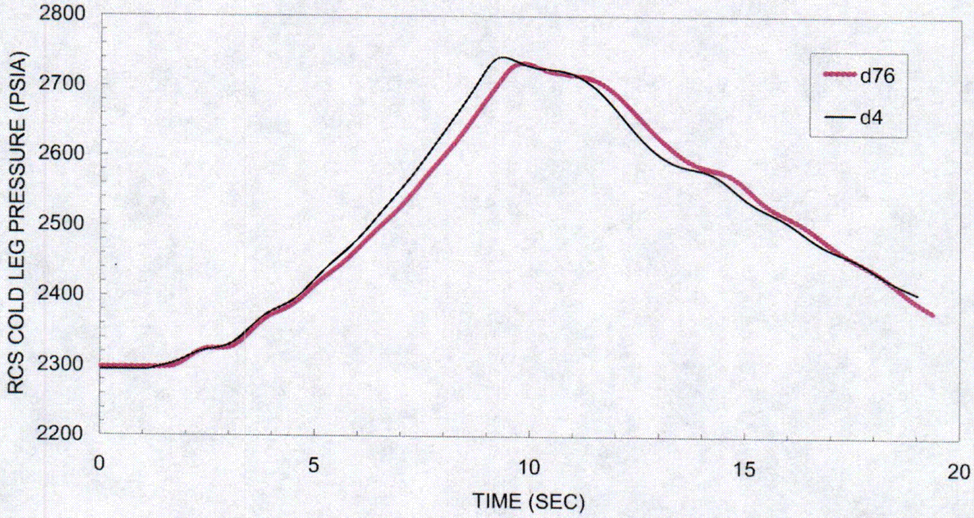

Figure 5-8. Comparison of results for a turbine trip transient with Δ76 and D4 steam generator designs – RCS cold leg (peak) pressure

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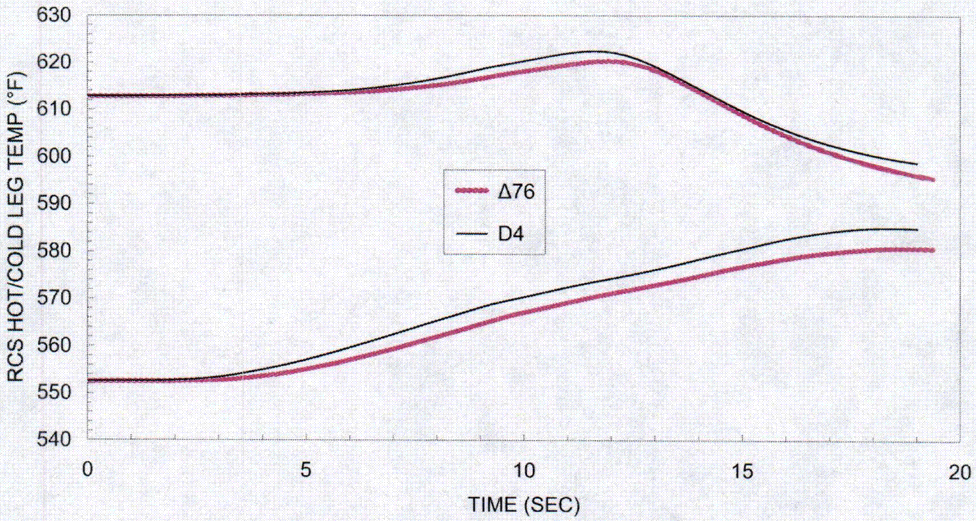

Figure 5-9. Comparison of results for a turbine trip transient with Δ76 and D4 steam generator designs – RCS hot and cold leg temperatures

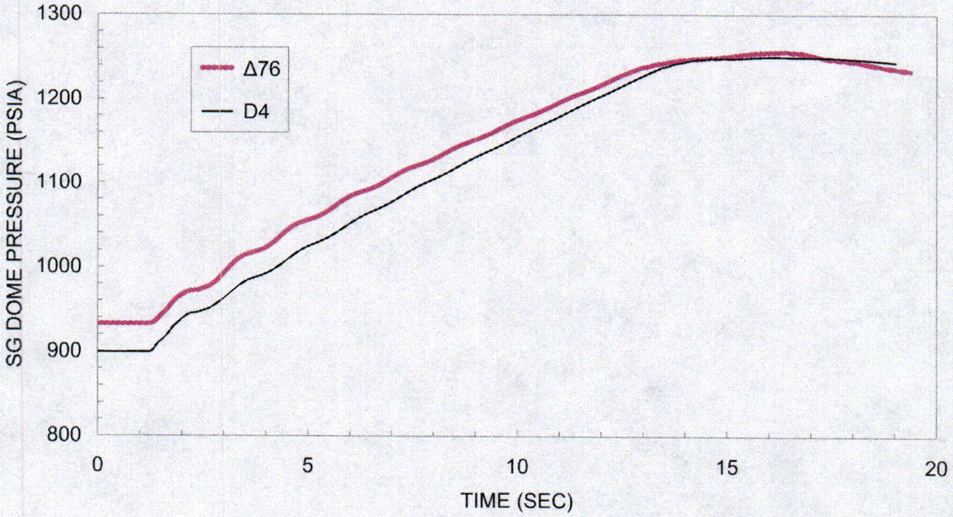

Figure 5-10. Comparison of results for a turbine trip transient with Δ76 and D4 steam generator designs – steam generator pressure

Question 6.

In Section 3.3 of ERX-04-005 (Ref. 2) it states, "As described in Reference 4 [RXE-91-001-A], a detailed steam generator model was developed by CPSES accident analysis engineers and used to determine a conservative mass equivalent of the low-low steam generator water level trip setpoint. For the Δ76 steam generator model, sufficient design data is available from the steam generator vendor to conservatively determine this mass equivalency without the need for the development of an additional detailed steam generator model."

a. **Please describe the design data and how it is used to determine steam generator water level.**
b. **In Section 4.2 of ERX-04-005 it states, ". . . a more detailed steam generator model is to be used to develop an appropriate mass-equivalent approximation of a particular steam generator water level setpoint." This appears to contradict the earlier statement that an additional detailed steam generator model was not needed to determine the Δ76 steam generator water level for transient analysis.**
   Please address this apparent discrepancy.
   1. **Please describe the 'detailed steam generator model' mentioned in Section 4.2 of ERX-04-005.**
   2. **Include the qualification/verification of the model.**

TXU Power Response:

The design data supplied by the vendor was calculated as described in Section 3.6 of WCAP-14882-P-A using detailed steam generator thermal-hydraulic codes which in turn have been extensively compared to actual plant data. This tool has been used extensively by Westinghouse design engineers since 1985 to provide steady-state thermal and hydraulic performance information. Because of the availability of the vendor-supplied data calculated using the detailed steam generator design tool, the development of a separate detailed steam generator model, as described in RXE-91-001-A, was not required for the replacement steam generator project.

During the development of the models and methods presented in RXE-91-001-A, a coarse-node steam generator model was incorporated into the integrated system model due to computer runtime considerations. This approach then required the development of a more detailed steam generator model, used in a stand-alone mode with forcing function boundary conditions, to translate the steam generator water level into a mass-equivalency that could be used with the coarse-node steam model in the integrated system model. An alternative would have been to use vendor design data; however, such information was not readily available to the utility at that time.

As part of the replacement steam generator activity, sufficient design data was supplied by the steam generator vendor to determine an appropriate mass-equivalency for use in the turbine trip, loss of non-emergency AC power, and loss of feedwater transient analyses. In these analyses, the normal fluid flow paths into, within, and out of the steam generator are maintained. As described in Section 4.2 of ERX-04-005, a different, transient-specific approach was used for the feedwater line break accident analysis due to the different steam flow paths out of the steam generator.

For the turbine trip, loss of non-emergency AC power, and loss of feedwater transient analyses, this vendor design data, based on the detailed steam generator design tool, was used to calculate an equivalent mass remaining the steam generator when the steam generator water level was predicted to be 0% of the narrow range span. Consistent with the application in RXE-91-001-A, this mass was then reduced by 10% for conservatism before use in the transient analyses.

For the feedwater line break analyses, a philosophically similar approach was used. Instead of using a detailed steam model to determine when the differential pressure between junctions representing the upper and lower steam generator water level taps would result in exceeding a setpoint, and then determining the equivalent mass remaining in the steam generator, the differential pressure was directly calculated in the actual transient simulation using the multi-node steam generator model from WCAP-14882-P-A. As noted in ERX-04-005, this slightly different approach was taken because the fluid flow paths in the steam generator are disrupted during a feedline break accident.

Q7 - LOCA related – see Attachment 1

Q8 - LOCA related – see Attachment 1

Question 9.

Generic Letter 83-11 (Ref. 8) requests each licensee to provide sufficient information to demonstrate their proficiency in using the code by submitting code verification performed by them. Please provide that information for each model affected ERX-04-005 and ERX-04-004.

TXU Power Response:

As noted in the NRC's Safety Evaluation Report approving RXE-91-001-A, "The licensee uses the safety processes to generate plant physical data input to its transient analysis methodology as it uses for loss of coolant (LOCA) analyses, including all data control provisions and consideration of differences between the two CPSES units. These processes were found acceptable in the staff safety evaluation report (SER) regarding the TU Electric LOCA evaluation model of April 26, 1993. We find the processes equally applicable and acceptable for use with the TU Electric transient analysis model."

The CPSES engineers continue to use work processes equivalent to those previously approved (the actual procedures have been revised; however, the fundamental requirements have not). All work is performed under an approved quality control program by qualified engineers. Thus, the expectations of GL 83-11 are met.

Question 10.

Absent any CPSES specific plant data available for the Δ76 steam generator, how will the licensee determine uncertainties for parameters associated with the RSG?

TXU Power Response:

The Δ76 steam generator model is an integrated part of the system model of CPSES. When performing analyses of specific transients and accidents, as described in Chapter 15 of the FSAR, uncertainties are applied to key parameters in the direction of conservatism for each analysis and each event acceptance criterion. The approach has been approved by the NRC in Section 3.4, "Licensing Analysis Approach" of the SER for RXE-91-001-A and continues to be used. The uncertainties are associated with instrument uncertainties and reactivity parameter calculation reliability factors.