PSE&G Public Service Electric and Gas Company

80 Park Plaza, Newark, NJ 07101 / 201 430-8217 MAILING ADDRESS / P.O. Box 570, Newark, NJ 07101

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

March 1, 1985

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

SAFETY EVALUATION REPORT
OPEN AND CONFIRMATORY ITEM STATUS
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354

Attachment 1 is a current list which provides a status of the open and confirmatory items identified in Sections 1.7 and 1.8 of the Safety Evaluation Report (SER). Items identified as "complete" are those for which PSE&G has provided responses and no confirmation of status has been received from the staff. We will consider these items closed unless notified otherwise. In order to permit timely resolution of items identified as "complete" which may not be resolved to the staff's satisfaction, please provide a specific description of the issue which remains to be resolved.

Enclosed for your review and approval (see Attachment 3) are the resolutions to the SER items listed in Attachment 2. This information will be incorporated, as required, into Amendment 10 of the HCGS FSAR.

Should you have any questions or require any additional information on these items, please contact us.

Very truly yours,

R L Mittl / R P Douglas

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Attachments

C D. H. Wagner
USNRC Licensing Project Manager (w/attach.)

A. R. Blough
USNRC Senior Resident Inspector (w/attach.)

M P84 154/04 1/2

Date: 3/1/85

ATTACHMENT 1

Item No.	Subject	Status	R. L. Mittl to A. Schwencer ltr. dated
OI-1	Riverborne Missiles	Partial Response	1/31/85 & 2/22/85
OI-2	Equipment Qualification	Partial Response	2/1/85, 2/20/85, & 2/28/85
OI-3	Preservice Inspection Program	Partial Response	2/14/85
OI-4	GDC 51 Compliance	Open	
OI-5	Solid-State Logic Modules	NRC Action	
OI-6	Postaccident Monitoring Instrumentation	NRC Action	
OI-7	Minimum Separation Between Non-Class IE Conduit and Class IE Cable Trays	Open	
OI-8	Control of Heavy Loads	Completed	1/18/85
OI-9	Alternate and Safe Shutdown	NRC Action	
OI-10	Delivery of Diesel Generator Fuel Oil and Lube Oil	Closed	Amendment 8
OI-11	Filling of Key Management Positions	Open	
OI-12	Training Program Items		
	(a) Initial Training Program	Completed	1/7/85
	(b) Requalification Training Program	Completed	12/28/84
	(c) Replacement Training Program	Completed	1/7/85
	(d) TMI Issues I.A.2.1, I.A.3.1, and II.B.4	Completed	1/7/85
	(e) Nonlicensed Training Program	Completed	1/7/85
OI-13	Emergency Dose Assessment Computer Model	Closed	1/7/85
OI-14	Procedures Generation Package	Closed	1/28/85
OI-15	Human Factors Engineering	Open	

Item No.	Subject	Status	R. L. Mittl to A. Schwencer ltr. dated
C-1	Feedwater Isolation Check Valve Analysis	Open	
C-2	Plant-unique Analysis Report	Completed	1/8/85 & 1/31/85
C-3	Inservice Testing of Pumps and Valves	Open	
C-4	Fuel Assembly Accelerations	Completed	Amendment 8
C-5	Fuel Assembly Liftoff	Completed	Amendment 8
C-6	Review of Stress Report	Open	
C-7	Use of Code Cases	Completed	12/17/84
C-8	Reactor Vessel Studs and Fastners	Completed	2/15/85
C-9	Containment Depressurization Analysis	NRC Review	
C-10	Reactor Pressure Vessel Shield Annulus Analysis	NRC Review	
C-11	Drywell Head Region Pressure Response Analysis	NRC Review	
C-12	Drywell-to-Wetwell Vacuum Breaker Loads	NRC Review	
C-13	Short-Term Feedwater System Analysis	Open	
C-14	Loss-of-Coolant-Accident Analysis	Completed	3/1/85
C-15	Balance-of-Plant Testability Analysis	Completed	Amendment 8
C-16	Instrumentation Setpoints	Completed	2/15/85
C-17	Isolation Devices	Open	
C-18	Regulatory Guide 1.75	NRC Review	
C-19	Reactor Mode Switch	NRC Review	
C-20	Engineered Safety Features Reset Controls	Open	

Item No.	Subject	Status	R. L. Mittl to A. Schwencer ltr. dated
C-21	High Pressure Coolant Injection Initiation	Open	
C-22	IE Bulletin 79-27	Completed	Amendment 8
C-23	Bypassed and Inoperable Status Indication	NRC Review	
C-24	Logic for Low Pressure Coolant Injection Interlock Circuitry	Open	
C-25	End-of-Cycle Recirculation Pump Trip	Completed	3/1/85
C-26	Multiple Control System Failures	NRC Review	
C-27	Relief Function of Safety/Relief Valves	Completed	2/15/85
C-28	Main Steam Tunnel Flooding Analysis	Open	
C-29	Cable Tray Separation Testing	Open	
C-30	Use of Inverter as Isolation Device	Open	
C-31	Core Damage Estimate Procedure	Open	
C-32	Continuous Airborne Particulate Monitors	Open	
C-33	Qualifications of Senior Radiation Protection Engineer	Open	
C-34	Onsite Instrument Information	Open	
C-35	Airborne Iodine Concentration Instruments	Open	
C-36	Emergency Plan Items	Partial Response	11/9/84, 1/16/85, & 2/7/85
C-37	TMI Item II.K.3.18	Partial Response	3/1/85

ATTACHMENT 2

ITEM NO.	SER SECTION	SUBJECT
C-14	6.3.5 and 15.9.3	Loss-of-Coolant-Accident Analysis
C-25	7.6.2.4	End-of-Cycle Recirculation Pump Trip
C-37	15.9.3	TMI Item II.K.3.18

JES:mr

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ATTACHMENT 3

SER Confirmatory Item No. 14 (SER Section 6.3.5 and 15.9.3)

Loss-of-Coolant-Accident Analysis

The LOCA analysis reported in the FSAR were for a lead plant representative of Hope Creek. The applicant has committed to supply plant-specific LOCA analyses in a later amendment to the FSAR before fuel loading. The NRC staff will report the results of its review of the plant-specific analyses in a supplement to this SER. This is a confirmatory item.

Response:

HCGS FSAR Sections 1.3, 1.10, 1.14, 6.2, 6.3, and 15.6 and Questions Responses 440.0, 440.27, and 440.28 have been revised to reflect the results of the HCGS plant-specific ECCS analysis.

JS:vw

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TABLE 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS(a)

	Hope Creek BWR 4/5 251-764	Hatch 1 BWR 4 218-560	Limerick BWR 4/5 251-764	Susquehanna BWR 4 251-764
Thermal and Hydraulic Design (Section 4.4)				
Rated power, MWt	3293	2436	3293	3293
Design power, MWt (ECCS design basis)	~~3435~~ 3430	2550	3435	3439
Steam flow rate, lb/h	14.156 E6	10.03 E6	14.156 E6	13.48 E6
Core coolant flow rate, lb/h	100.0 E6	78.5 E6	100.0 E6	100.0 E6
Feedwater flow rate, lb/h	14.117 E6	10.445 E6	14.117 E6	13.574 E6
System pressure, nominal in steam dome, psia	1020	1020	1020	1020
Average power density, kW/liter	48.7	51.2	48.7	48.7
Maximum linear heat generation rate, kW/ft	13.4	13.4	13.4	13.4
Average linear heat generation rate, kW/ft	5.34	7.11	5.3	5.34
Maximum heat flux, Btu/h-ft^2	361,600	428,300	361,600	361,000
Average heat flux, Btu/h-ft^2	144,100	164,700	143,700	144,100
Maximum UO_2 temperature, °F	3412	4380	3435	3330
Average volumetric fuel temperature, °F	2149	2781	2130	2130
Average cladding surface temperature, °F	566	558	566	558
Minimum critical power ratio	1.20	(a)	1.24	1.23
Coolant enthalpy at core inlet, Btu/lb	526.1	526.2	526.1	521.8
Core maximum exit voids within assemblies	77.1	79	77.1	76.00
Core average exit quality, % steam	14.1	12.7	14.1	13.2
Feedwater temperature, °F	419.9	387.4	420	383

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performing the necessary work and submitted this information for staff review and approval.

Response

General Electric provided information concerning the NRC's small-break-model concerns in a meeting between GE and the NRC staff held on June 18, 1981 and subsequent documentation included in a letter from R.H. Bucholz (GE) to D.G. Eisenhut (NRC) dated June 26, 1981. Based on its review of this information, the NRC staff has prepared a draft safety evaluation report (SER) that concludes the test data, comparisons, and other information submitted by GE acceptably demonstrate that the existing GE small-break model is in compliance with 10 CFR 50, Appendix K and, therefore, no model changes are required. ~~Should the NRC management review of the draft SER raise any further concerns, they will be resolved prior to the initiation of the HCGS-specific ECCS analysis in late 1984.~~

- II.K.3.31 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR 50.46

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents as described in II.K.3 item 30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

Calculations to be submitted by January 1, 1983 or 1 year after staff approval of loss-of-coolant accident analysis models, whichever is later (required only if model changes have been made).

Response

Small-break LOCA calculations are described in Section 6.3.3.7, and the results are summarized in Table 6.3-4. The references in Section 6.3.6 describe the currently approved Appendix K methodology used. Compliance with 10 CFR 50.46 has been previously established by the NRC. No model changes are necessary (see response to item II.K.3.30).

1.14.1.26.2 Response

This issue is not applicable to the HCGS because it does not have a HPCS.

1.14.1.27 Adequate Core Cooling Maintained with LPCI Diversion, LRG I/RSB-18

1.14.1.27.1 Issue

The NRC staff asked for a demonstration that adequate core cooling would be maintained if the flow of the low-pressure coolant injection were diverted to the wetwell and drywell sprays and to suppression pool cooling.

1.14.1.27.2 Response

This situation is addressed in Section 6.3. Sufficient margin exists in the peak cladding temperature to accommodate the diversion of low-pressure coolant injection at 600 seconds into the transient. This demonstrates adequate core cooling. ~~Further confirmation will be provided in the plant-unique ECCS analysis that will be completed in July 1985.~~ e e e

1.14.1.28 Temperature Drop with Feedwater Heater Failure, LRG I/RSB-19

1.14.1.28.1 Issue

The analysis of the feedwater heater failure event is based on a temperature drop no greater than 100°F. However, an actual failure demonstrated a 150°F drop. The NRC staff has requested a justification for the smaller temperature drop or a reanalysis with a justified temperature decrease.

1.14.1.28.2 Response

The design specification for the feedwater heating system requires that the maximum temperature decrease due to a single failure be no greater than 100°F. Sufficient analyses have been

1.14.1.107.2 Response

See response to LRG Issue No. 106, Section 1.14.106.

1.14.1.108 Nonconservatism in the Models For Fuel Cladding Swelling and Rupture LRG I/CPB-2 and LRG II/1-CPB

1.14.1.108.1 Issue

The procedures proposed in NUREG-0630 introduce additional conservatism in the models for fuel cladding swelling and rupture during a loss-of-coolant accident. To assure the degree of swelling and incidence of rupture are not underestimated as required by Appendix K of 10 CFR 50.46, supplemental calculations to the current ECCS analyses should be performed. If the swelling is underestimated, the bundle cooling may be overestimated, and the peak cladding temperature may be nonconservative.

1.14.1.108.2 Response

ADD INSERT D

~~The current understanding with the NRC staff is that the ECCS analyses have adequate overall conservatism although they may underestimate the effects of cladding swelling and rupture. When the HCGS-unique ECCS calculations are prepared, in July of 1985, the curve for perforation stress versus temperature will be modified for temperatures below 1600°F, and the then-current model technology will be utilized.~~

1.14.1.109 Fuel Rod Cladding Ballooning and Rupture

1.14.1.109.1 Issue

The procedures proposed in NUREG-0630 introduce additional conservatism in the models for fuel cladding swelling and rupture during a loss-of-coolant accident. To assure the degree of swelling and incidence of rupture are not underestimated as required by Appendix K of 10 CFR 5046, supplemental calculations to the current ECCS analyses should be performed. If the swelling is underestimated, the bundle cooling may be overestimated, and the peak cladding temperature may be nonconservative.

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INSERT D

The HCGS-unique ECCS calculations were prepared utilizing a cladding rupture temperature model modified for temperatures less than 1600°F The NRC Staff found this model acceptable with respect to the criteria in NUREG-0630 as evidenced by a supplementary safety evaluation report accepting General Electric's fuel cladding balloning and rupture model (see Reference in Section 1.14.108.2.1).

1.14.108.2.1 Reference

Letter from H. Bernard (NRC) to G.G. Sherwood (GE), "Supplementary Acceptance of Licensing Topical Report NEDE-20566-P," MFN 067-82, May 11, 1982.

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CHAPTER 6

ENGINEERED SAFETY FEATURES

FIGURES (cont)

Figure	Title
6.3-11	Head vs Low Pressure Coolant Injection Flow Used in LOCA Analyses
6.3-12	Process Diagram, Residual Heat Removal System
6.3-13	RHR (LPCI) Pump Characteristics
6.3-14	Peak Cladding Temperature and Maximum ~~Minimum~~ Local Oxidation vs Break Area
6.3-15	Normalized Core Power vs Time
6.3-16	Core Average Pressure vs. Time After Break (DBA, Recirculation Suction Break, Failure of Channel A DC Source)
6.3-17	Normalized Core Average Inlet Flow vs Time After Break (DBA, Recirculation Suction Break, Failure of Channel A DC Source)
6.3-18	Core Inlet Enthalpy vs. Time After Break (DBA, Recirculation Suction Break, Failure of Channel A DC Source)
6.3-19	Minimum Critical Power Ratio vs. Time After Break (DBA, Recirculation Suction Break, Failure of Channel A DC Source)
6.3-20	Water Level Inside Shroud vs Time After Break (DBA, Recirculation Suction Break, Failure, Channel A DC Source)
6.3-21	Reactor Vessel Pressure vs. Time After Break (DBA, Recirculation Suction Break, Failure of Channel A DC Source)
6.3-22	Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Large Break Model) (DBA, Recirculation Suction Break, Failure of Channel A DC Source)

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CHAPTER 6

ENGINEERED SAFETY FEATURES

FIGURES (cont)

Figure	Title
	(0.09 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)
6.3-42	Water Level Inside Shroud vs. Time After Break (Small Break Model) (~~0.2~~ 0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)
6.3-43	Reactor Vessel Pressure vs. Time After Break (Small Break Model) (~~0.2~~ 0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)
6.3-44	Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Small Break Model) (~~0.2~~ 0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)
6.3-45	Peak Cladding Temperature vs. Time After Break (Small Break Model) (~~0.2~~ 0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)
6.3-46	Water Level Inside Shroud vs. Time After Break (Small Break Model) (Maximum Core Spray Line Break, Failure of Channel A DC Source)
6.3-47	Reactor Vessel Pressure vs. Time After Break (Small Break Model) (Maximum Core Spray Line Break, Failure of Channel A DC Source)
6.3-48	Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Small Break Model) (Maximum Core Spray Line Break, Failure of Channel A DC Source)
6.3-49	Peak Cladding Temperature vs. Time After Break (Small Break Model) (Maximum Core Spray

CHAPTER 6

ENGINEERED SAFETY FEATURES

FIGURES (cont)

Figure	Title
	Line Break, Failure of Channel A DC Source)
6.3-50	Water Level Inside Shroud vs. Time After Break (Small Break Model) (Maximum Feedwater Line Break, Failure of Channel A DC Source)
6.3-51	Reactor Vessel Pressure vs. Time After Break (Small Break Model) (Maximum Feedwater Line Break, Failure of Channel A DC Source)
6.3-52	Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Small Break Model) (Maximum Feedwater Line Break, Failure of Channel A DC Source)
6.3-53	Peak Cladding Temperature vs. Time After Break (Small Break Model) (Maximum Feedwater Line Break, Failure of Channel A DC Source)
6.3-54	Water Level Inside Shroud vs. Time After Break (Maximum Main Steam Line Inside ~~Primary~~ e Containment, Failure of Channel A DC Source)
6.3-55	Reactor Vessel Pressure vs. Time After Break (Maximum Main Steam Line Break Inside e ~~Primary~~ Containment, Failure of Channel A DC Source)
	ADD INSERT A →
6.3-~~56~~ 58	Water Level Inside Shroud vs. Time After Break (Maximum Main Steam Line Break Outside e ~~Primary~~ Containment, Failure of Channel A DC Source)
6.3-~~57~~ 59	Reactor Vessel Pressure vs. Time After Break (Maximum Main Steam Line Break Outside e ~~Primary~~ Containment, Failure of Channel A DC Source)
6.3-~~58~~ 60	Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Small Break Model) (~~Main~~ Maximum

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6.3-56 Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Maximum Main Steam Line Break Inside Containment, Failure of Channel A DC Source)

6.3-57 Peak Cladding Temperature vs. Time After Break (Maximum Main Steam Line Break Inside Containment, Failure of Channel A DC source)

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CHAPTER 6

ENGINEERED SAFETY FEATURES

FIGURES (cont)

Figure	Title
	Steam Line Break Outside ~~Primary~~ Containment, Failure of Channel A DC Source)
6.3-~~59~~ 61	Peak Cladding Temperature vs. Time After Break (Small Break Model) (Maximum Main Steam Line Break Outside ~~Primary~~ Containment, Failure of Channel A DC Source)
6.3-~~60~~ 62	Total Time Highest Powered Node Remains Uncovered vs Break Area (Failure of Channel A DC Source)
6.4-1	Control Room Arrangement
6.4-2	Plant Layout with Respect to Control Room Intake
6.7-1	Main Steam Isolation Valve Sealing System, P&ID
6A-1	Model Schematic for Inadvertent Spray Actuation
6A-2	Thermal Heat Removal Efficiency of Containment Atmosphere Spray
6A-3	Containment Pressure Response - Inadvertent Spray Actuation - 2 Spray Loops, and 1 PV Fails
6A-4	Containment Temperature Response - Inadvertent Spray Actuation - 2 Spray Loops, and 1 PV Fails
6A-5	Differential Pressure Between Drywell and Suppression Chamber - Inadvertent Spray Actuation - 2 Spray Loops, and 1 VB Fails
6A-6	Containment Temperature Response - Inadvertent Spray Actuation - 2 Spray Loops, and 1 VB Fails
6B-1	Flow Diverter
6B-2	Reactor Shield Annulus Arrangement
6B-3a	Schematic of the RPV Shield Annulus Model

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conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2

e. Criterion 5, Long-Term Cooling - "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated generically for General Electric BWRs in Section III.A of Reference 6.3-1. Briefly summarized, the core remains covered to at least the jet pump suction elevation, and the uncovered region is cooled by spray cooling and/or by steam generated in the covered part of the core.

6.3.3.3 Single Failure Considerations

The functional consequences of single failures, including operator errors that might cause any manually controlled, electrically operated valve in the ECCS to move to a position that could adversely affect the ECCS, and the potential for submergence of valve motors in the ECCS, are discussed in Sections 6.3.1.1.2 and 6.3.1.1.4. The most severe single failures are identified in Table 6.3-6. Therefore, only these single failures are considered in the ECCS performance analyses. ~~For large breaks, failure of one of the SDGs is, in general, the most severe failure. For small breaks, loss of HPCI is the most severe failure.~~

ADD Insert A

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as the following:

a. An initiation signal is received.

b. A small lag time (to open all valves and run the pumps up to rated speed) occurs.

c. The ECCS flow enters the reactor vessel.

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For both large and small breaks, failure of the channel A dc source is the most severe failure.

¶

A single failure in the ADS (one ADS valve) has no effect in large breaks. Therefore, as a matter of calculational convenience, it is assumed in all calculations that one ADS valve fails to operate in addition to the identified single failure. This assumption reduces the number of calculations required in the performance analysis and bounds the effects of one ADS valve failure and the channel A dc source failure by themselves. The only effect of the assumed ADS valve failure on the calculations is a small increase (on the order of 100°F) in the calculated temperatures following small breaks.

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Immediately following a LOCA, the RHR system is aligned to the LPCI mode.

6.3.3.6 Limits on ECCS System Parameters

Refer to Sections A.6.3.3.6 through A.6.3.3.7.2 of Appendix A of Reference 6.3-3.

Compliance with Regulatory Guide 1.47 is identified in Section 1.8.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

Refer to Section A.6.3.3.7.1, of Appendix A, of Reference 6.3-3. The significant input variables used by the LOCA codes are given in Table 6.3-2 and on Figure 6.3-15.

6.3.3.7.2 Accident Description

Reference to a detailed description of the LOCA calculation is provided in Section A.6.3.3.7.2, of Appendix A, of Reference 6.3-3.

6.3.3.7.3 Break Spectrum Calculations

~~The analysis results presented in this section were obtained from a typical LOCA analysis, which is representative of this plant size and product line. A plant-specific LOCA analysis will be submitted later as an FSAR amendment.~~

A complete spectrum of postulated break sizes and locations is considered in the evaluation of ECCS performance. For ease of reference, a summary of all figures and tables in Section 6.3.3 is shown in Table 6.3-4.

A summary of the results of the break spectrum calculations is shown in tabular form in Table 6.3-3 and graphically on

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Figure 6.3-14. Conformance to the acceptance criteria (peak cladding temperature ≤ 2200°F, local oxidation ≤ 17%, and core-wide metal-water reaction ≤ 1%) is demonstrated. Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Recirculation Line Break Calculations

The characteristics that determine which is the most limiting large break are:

a. The calculated time for reflooding the ~~the~~ hot node

b. The calculated time for uncovering the hot node

c. The calculated time of boiling transition.

The calculated time of boiling transition increases with decreasing break size, since the time of uncovering of the jet pump suction inlet, which leads to boiling transition, is determined primarily by the break size. The calculated time for uncovering the hot node also generally increases with decreasing break size, since it is determined primarily by the reactor coolant inventory lost during the blowdown.

The hot node reflooding time is determined by a number of interacting phenomena, such as depressurization rate, countercurrent flow limiting, and a combination of available ECCS.

The period between the uncovering of the hot node and its reflooding is the period when the hot node has the lowest heat transfer. Hence, the break that results in the longest period during which the hot node remains uncovered results in the highest calculated peak cladding temperature. If two breaks have similar times during which the hot node remains uncovered, then the larger of the two breaks will be limiting, as it would have an earlier boiling transition time (i.e., the larger break would have a more severe result from a blowdown heat transfer analysis).

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Figure 6.3-~~60~~ 62 shows the variation with break size of the calculated time the hot node remains uncovered. Based on these calculations, the design basis accident (DBA) was determined to be the break that results in the highest calculated peak cladding temperature in the 1.0 ft^2 to 4.1 ft^2 region (the largest possible area of a recirculation system line break is 4.1 ft^2). Confirmation that this is the most limiting break over the entire break spectrum is shown in Figure 6.3-14.

Important variables from the analysis of the DBA are shown on Figures 6.3-16 through 6.3-25. These variables are:

a. Core average pressure as a function of time

b. Core flow as a function of time

c. Core inlet enthalpy as a function of time

d. Minimum critical power ratio as a function of time

e. Water level as a function of time

f. Pressure as a function of time

g. Fuel rod convective heat transfer coefficient as a function of time

h. Peak cladding temperature as a function of time

i. Hot pin (the rod with the highest cladding temperature at a particular time) average fuel temperature as a function of time

j. Hot pin fuel internal pressure as a function of time

The maximum average planar linear heat generation rate (MAPLHGR), maximum local oxidation, and peak cladding temperature as functions of exposure (from the analysis of the DBA), are shown in Table 6.3-5.

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6.3.3.7.6 Small Recirculation Line Break Calculations

Important variables from the analysis of the small break yielding the highest cladding temperature are shown on Figures 6.3-38 through 6.3-41. These variables are:

a. Water level as a function of time

b. Pressure as a function of time

c. Fuel rod convective heat transfer coefficient as a function of time

d. Peak cladding temperature as a function of time

The same variables resulting from the analysis of a less limiting small break are shown on Figures 6.3-42 through 6.3-45.

6.3.3.7.7 Calculations for Other Break Locations

Reactor vessel water level and pressure, ~~and~~ fuel rod convective heat transfer coefficient and the peak cladding temperature are shown on Figures 6.3-46 through 6.3-49 for the core spray line break, ~~and~~ on Figures 6.3-50 through 6.3-53 for the feedwater line break, and on Figures 6.3-54 ~~and~~ Through 6.3-~~55~~ 57 ~~show the reactor vessel water level and pressure~~ for ~~a~~ the main steam line break inside ~~the primary~~ containment.

An analysis was also done for a main steam line break outside ~~the primary~~ containment. Reactor vessel water level and pressure, fuel rod convective heat transfer coefficient and peak cladding temperature are shown on Figures 6.3-~~56~~ 58 through 6.3-~~59~~ 61.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Section 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10 CFR 50.46 acceptance criteria, given operation at or below the

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automatically realign from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The core spray and LPCI systems begin injection into the reactor pressure vessel (RPV) when reactor vessel pressure decreases to system discharge shutoff pressure. HPCI injection begins as soon as the HPCI turbine-pump is up to speed. The injection valve is open, since the HPCI system is capable of injecting water at full flow into the RPV over a pressure range from 200 psig to reactor pressure specified in mode A of Figure 6.3-3.

6.3.6 REFERENCES

6.3-1 General Electric, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NED~~O~~-20566-P, November 1975.
E

6.3-2 H. M. Hirsch, Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems, NEDO-10739, General Electric, January 1973.

6.3-3 General Electric, "General Electric Standard Application for Reactor Fuel," including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

Approx. 22 Reactor low pressure is reached. Core spray valves receive pressure permissive signal to open.

TABLE 6.3-1

OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEM FOR DESIGN BASIS LOSS-OF-COOLANT ACCIDENT(1)

Time (s)	Events
0	Design basis loss-of-coolant accident is assumed to start; offsite power is assumed to be lost.
Approx. 0	Drywell high pressure(2) and reactor vessel low water level (level 3) are reached. All SDGs are signaled to start, reactor scram is initiated, and HPCI, core spray, and LPCI receive the first signal to start on drywell high pressure.
Approx. ~~3~~ 2	Reactor vessel low-low water level (level 2) is reached. HPCI receives the second signal to start. HPCI injection valve is signaled to open
Approx. ~~7~~ 5	Reactor vessel low-low-low water level (level 1) is reached. The second signal to start LPCI and core spray is given. The auto-depressurization sequence begins. MSIVs are signaled to close.
Approx. 15 ~~≤10~~	All SDGs are ready to load. ~~The HPCI injection valve is signaled to open.~~ Energizing of the core spray and RHR (LPCI) pump motors begins.
Approx. ~~≤~~27	The HPCI injection valve is open and the pump is at design flow, which completes the HPCI startup.
Approx. 34 ~~≤40~~	The ~~LPCI and~~ core spray pumps are at rated flow and the core spray injection valves are open, which completes the ~~LPCI and~~ core spray system startup~~s~~.
See Figure 6.3-20	The core is effectively reflooded, assuming the worst single failure; heatup is terminated.
>10 min	The operator shifts to containment cooling.

(1) For the purpose of all but the next-to-the last entry on this table, all ECCS equipment is assumed to function as designed. Performance analysis calculations consider the effects of single equipment failures (see Sections 6.3.2.5 and 6.3.3.3).

(2) No credit is taken in the DBA LOCA analysis for ECCS initiation on the high drywell pressure signal.

Approx. 45 The LPCI pumps are at rated flow and the injection valves are open which completes the LPCI system startup.

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TABLE 6.3-2 Page 1 of 3

SIGNIFICANT INPUT VARIABLES USED IN
LOSS-OF-COOLANT ACCIDENT ANALYSIS

Variable	Value
A. Plant Parameters	
Core thermal power	~~3435~~ 3430 MWt
Vessel steam output	14.~~86~~ 87 x 10^6 lbm/h
Corresponding percent of rated steam flow	105%
Vessel steam dome pressure	1055 psia
Maximum area of recirculation line break	4.1 ft^2
B. Emergency Core Cooling System Parameters	
Low Pressure Coolant Injection System	
Vessel pressure at which flow may commence	≤295 psid (vessel to drywell)
Minimum rated flow, at vessel pressure	40,000 gpm, at 20 psid (vessel to drywell)
Initiating signals	
Low water level, or	1.0 feet above top of active fuel
High drywell pressure	2.0 psig (1)
Maximum allowable time delay from initiating signal to pumps at rated speed	40 seconds
Injection valve fully open	40 seconds after maximum suction break
Core Spray System	
Vessel pressure at which flow may commence	≤289 psid (vessel to drywell)
Minimum rated flow, at vessel pressure	6250 gpm, at 105 psid (vessel to drywell)

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TABLE 6.3-2 (cont)

Variable	Value
Initiating signals~~(a)~~	
Low water level, or	~~≤~~1.0 feet above top of active fuel
High drywell pressure	~~≥~~2.0 psig (1)
Minimum ~~Maximum~~ allowed (runout) flow per loop	~~7900~~ 7000 gpm
Maximum allowed delay time from initiating signal to pump at rated speed	27 seconds
Pressure at which injection valve may open	≤ 425 psig
Injection valve fully open	≤27 seconds after maximum break or 12 seconds after pressure permissive signal, whichever is greater.
Combined HPCI/Core Spray System	
Minimum flow rate (independent of vessel pressure)	5600 gpm
Minimum rated core spray flow available, at vessel pressure	6250 gpm, at 105 psid (vessel to pump suction)
Initiating signals~~(a)~~	
Low water level, or	~~≤10.9~~ 8.6 feet above top of active fuel
High drywell pressure	~~≥~~2.0 psig (1)
Maximum allowed delay time from initiating signal to rated flow available and injection valve wide open	27 seconds (core spray system) 85 seconds (HPCI system)
maximum ~~Minimum~~ HPCI flow rate injected through the core spray sparger	~~2000~~ 3000 gpm
Automatic Depressurization System	
Total number of relief valves installed with ADS function	5

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Variable	Value
Number of ADS valves used in analysis for 4 valves	~~5~~ 4
Total minimum flow capacity, at a vessel pressure	3.2 ~~4.0~~ x 10^6 lbm/h, at 1125 ~~psid (vessel to suppression pool)~~ psig
ADS timer Initiating signals ~~(a)~~	
a) Low water level, and	1.0 feet above top of active fuel
high drywell pressure, and a signal that at least one RHR pump or one core spray system is running at a ~~(~~pump discharge pressure~~)~~	2.0 psig [(1)] 145 psig
or,	
b) Low water level, and	1.0 feet above top of active fuel
high-drywell-pressure bypass timer timed out, and a signal that at least one RHR pump or one core spray system is running at a ~~(~~pump discharge pressure~~)~~	6 ~~8~~ minutes from initiating signal 145 psig (not modeled)
ADS timer Delay time for all initiating signals completed to the time valves are open	120 seconds

ADD INSERT B →

C. Fuel Parameters

Fuel type	control cell ~~Initial~~ core
Fuel bundle geometry	8 x 8
Lattice	C
Number of fueled rods per bundle	62
Peak technical specification linear heat generation rate	13.4 kW/ft
Initial minimum critical power ratio	1.2
Design axial peaking factor	1.4

(1) ~~This analysis is binding for initiating signals within the indicated range.~~

No credit taken in the DBA LOCA analysis for ECCS system initiation on the high drywell pressure signal

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INSERT B

High drywell pressure
bypass timer initiating
signal

Low water level

1.0 feet above top of
active fuel

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TABLE 6.3-3

SUMMARY OF RESULTS OF LOCA ANALYSIS

Break Size Location Single Failure		Peak Cladding Temperature, °F	Peak Local Oxidation, %
A. 4.1 ft^2 (DBA)/ recirc suction/ Channel A dc source		2046 ~~2009~~(1)	~~1.5~~ 1.5
B. 1.0 ft^2/ recirc suction/ Channel A dc source	Large break methods	1599 ~~1742~~(1)	<1
	Small break methods	1156 ~~1454~~(2)	<1
C. 0.09 ft^2/ recirc suction/ Channel A dc source		1694 ~~1736~~(2)	<1

(1) Core heatup model, CHASTE - large break methods.
(2) Non-DBA reflood - small break methods.

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TABLE 6.3-4

SUMMARY OF FIGURES AND TABLES IN SECTION 6.3

	Large Break	Transition Break		Small Breaks		Other Breaks			
Break size	DBA	1.0 ft²	1.0 ft²	0.09 ft²	~~0.2~~ 0.4 ft²	Core spray line	Feedwater line	Main steam line	Main steam line
FSAR section	6.3.3.7.4	6.3.3.7.5	6.3.3.7.5	6.3.3.7.6	6.3.3.7.6	6.3.3.7.7	6.3.3.7.7	6.3.3.7.7	6.3.3.7.7
Remarks	-	Large break methods	Small break methods	Worst small break	Additional small break	-	-	Inside the containment	Outside the containment
Variable:									
Core average pressure	6.3-16	6.3-26	-	-	-	-	-	-	-
Core average inlet flow	6.3-17	6.3-27	-	-	-	-	-	-	-
Core inlet enthalpy	6.3-18	6.3-28	-	-	-	-	-	-	-
Minimum critical power ratio	6.3-19	6.3-29	-	-	-	-	-	-	-
Water level inside shroud	6.3-20	6.3-30	6.3-34	6.3-38	6.3-42	6.3-46	6.3-50	6.3-54	6.3-~~56~~ 58
Reactor vessel pressure	6.3-21	6.3-31	6.3-35	6.3-39	6.3-43	6.3-47	6.3-51	6.3-55	6.3-~~57~~ 59
Fuel rod convective heat transfer coefficient	6.3-22	6.3-32	6.3-36	6.3-40	6.3-44	6.3-48	6.3-52	6.3-56	6.3-~~58~~ 60
Peak cladding temperature	6.3-23	6.3-33	6.3-37	6.3-41	6.3-45	6.3-49	6.3-53	6.3-57	6.3-~~59~~ 61
Hot pin average fuel temperature	6.3-24	-	-	-	-	-	-	-	-
Hot pin fuel internal pressure	6.3-25	-	-	-	-	-	-	-	-

Miscellaneous Tables and Figures

Input variables	Table 6.3-2 and Figure 6.3-15
Operational sequence of ECCS for DBA	Table 6.3-1
Peak cladding temperature, maximum local oxidation, and MAPLHGR versus exposure	Table 6.3-5
Summary of results of LOCA analysis	Table 6.3-3
Single failure evaluation	Table 6.3-6
ECCS head versus flow curves	Figures 6.3-4, 6.3-5, 6.3-9, and 6.3-11
Peak cladding temperature and maximum local oxidation versus break area	Figure 6.3-14
Total time highest powered node remains uncovered versus break area	Figure 6.3-~~60~~ 62

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TABLE 6.3-5

MAPLHGR, MAXIMUM LOCAL OXIDATION, AND PEAK CLADDING TEMPERATURE VERSUS EXPOSURE(1)(2)(3)

Average Planar Exposure, MWd/t	MAPLHGR, kW/ft	Peak Cladding Temperature, °F	Oxidation Fraction
A. Fuel type 8CR183			
200.0	12.0	1966	0.0097
1,000.0	12.1	1961	0.0094
5,000.0	12.7	1981	0.0096
10,000.0	12.8	1981	0.0094
15,000.0	12.9	2009	0.0128
20,000.0	12.7	1997	0.0101
25,000.0	11.7	1883	0.0066
30,000.0	10.8	1771	0.0042
B. Fuel type 8CR233			
200.0	11.9	1972	0.0098
1,000.0	12.0	1961	0.0093
5,000.0	12.1	1937	0.0083
10,000.0	12.2	1932	0.0080
15,000.0	12.2	1957	0.0088
20,000.0	12.1	1960	0.0090
25,000.0	11.6	1909	0.0075
30,000.0	11.2	1855	0.0061
C. Fuel type 8CR711			
200.0	11.5	1878	0.0066
1,000.0	11.4	1838	0.0056
5,000.0	11.4	1806	0.0049
10,000.0	11.5	1792	0.0045
15,000.0	11.5	1797	0.0046
20,000.0	11.0	1751	0.0039
25,000.0	10.4	1684	0.0029
30,000.0	9.7	1602	0.0020

Replace with insert B

(1) The core-wide metal-water reaction has been calculated using method 1 described in Reference 6.3-1. The value is as follows:

Add Insert C → Core-wide metal-water reaction (%) = ~~0.09~~ 0.10

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INSERT B (sheet 1 of 2)

A. Fuel type P8CRB071

200	11.5	1910	0.009
1,000	11.4	1872	0.008
5,000	11.4	1810	0.006
10,000	11.5	1794	0.006
15,000	11.5	1792	0.006
20,000	11.1	1747	0.005
25,000	10.4	1688	0.004
30,000	9.8	1621	0.003
35,000	9.1	1546	0.002
40,000	8.5	1468	0.001
45,000	7.8	1394	0.001

B. Fuel type P8CRB094

200	10.7	1912	0.009
1,000	11.0	1909	0.009
5,000	11.6	1879	0.008
10,000	11.9	1860	0.007
15,000	11.7	1820	0.006
20,000	11.3	1774	0.005
25,000	10.5	1694	0.004
30,000	9.8	1619	0.003
35,000	9.2	1547	0.002
40,000	8.5	1474	0.001
45,000	7.9	1407	0.001

C. Fuel type P8CRB163

200	11.8	1990	0.012
1,000	11.8	1985	0.012
5,000	12.4	1994	0.011
10,000	12.8	1990	0.011
15,000	12.9	2015	0.012
20,000	12.9	2017	0.012
25,000	12.2	1923	0.009
30,000	11.2	1788	0.006
35,000	10.6	1716	0.004
40,000	10.1	1658	0.003
45,000	9.4	1599	0.003

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INSERT (B) (sheet 2 of 2)

D. Fuel type P8CRB248

200	12.1	2046	0.015
1,000	12.1	2037	0.014
5,000	12.3	1981	0.011
10,000	12.1	1949	0.010
15,000	12.1	1952	0.010
20,000	11.9	1941	0.010
25,000	11.2	1873	0.008
30,000	10.7	1790	0.006
35,000	10.0	1714	0.004
40,000	9.4	1650	0.003
45,000	8.7	1589	0.002

E. Fueltype P8CRB278

200	11.7	1960	0.011
1,000	11.8	1957	0.011
5,000	12.4	1959	0.010
10,000	12.5	1951	0.010
15,000	12.4	1946	0.010
20,000	12.2	1936	0.009
25,000	11.5	1859	0.007
30,000	10.8	1779	0.006
35,000	10.2	1699	0.004
40,000	9.5	1633	0.003
45,000	8.9	1568	0.002

INSERT

(C)

(2) The analyses ~~contained herein~~ were performed with the assumption that all lower tie plates are fully drilled.

(3) This analysis is valid for operation at all points on the power-flow map (see Figure 4.4-6) bounded by the most restrictive of the following:

a) Less than the 100%-rated-power line
b) Less than the APRM-rod-block line
c) Less than the 100%-rated-core-flow line

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TABLE 6.3-6

SINGLE FAILURE EVALUATION

The following table shows the single active failures considered in the ECCS performance evaluation.

Assumed Failure[1]	Systems Remaining[2]
Channel A dc source	1 core spray loop + 3 LPCI + ~~8~~ 4 ADS
SDG	1 core spray loop + HPCI + 3 LPCI + 4 ~~8~~ ADS
LPCI injection valve	2 core spray loops + HPCI + 3 LPCI + 4 ~~8~~ ADS
HPCI	2 core spray loops + 4 LPCI + ~~8~~ 4 ADS
One ADS valve	2 core spray loops + 4 LPCI + HPCI + 4 ADS

(1) Other postulated failures are not specifically considered, because they all result in at least as much ECCS capacity as one of the failures designated above.

(2) Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

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TABLE 15.6-7

SEQUENCE OF EVENTS FOR A STEAM LINE BREAK OUTSIDE PRIMARY CONTAINMENT~~(1)~~

Approximate Time, s	Event
0	Break of one main steam line outside primary containment
Approx. 0.5	High steam line flow signal initiates closure of MSIVs
<1	Reactor begins to scram~~s~~
≤5.5	MSIVs fully closed
~~46~~ Approx. 30	SRVs open upon high vessel pressure. The valves then open and close to maintain vessel pressure at approximately ~~1100~~ 1000 psi.
~~1160~~ Approx 490	ADS initiates on low water level, L1; ~~following time delays imposed by both the ADS timer and the~~ ADS high-drywell-pressure bypass timer~~s~~ started. ~~Vessel depressurizes rapidly.~~
Approx 1215 ~~1370~~	Low-pressure ECCS systems begin injection with reactor fuel partially uncovered.
Approx 1290 ~~1450~~	Core reflooded and clad temperature heatup terminated; no fuel rod failure.
Approx. 27	RCIC and HPCI would have initiated on low water level, L2 (RCIC considered unavailable and HPCI assumed disabled by channel A DC power source failure).
Approx. 90	Reactor water level above core begins to drop slowly due to the loss of steam through the SRVs. Reactor pressure remains at approximately 1000 psi.
Approx. 970	All ADS timer's time delays are completed; ADS valves are actuated; rapid depressurization of vessel initiated.

~~(1) The event times presented here are typical of BWR 4 plants with ADS logic modification (see Section 1.10.2.II.K.3.18). HCGS-unique values will be provided when the HCGS-unique ECCS analysis is submitted in July 1985~~.

SER ITEM C-14

QUESTION 440.0 (SECTION 6.3.3.7.3)

Provide the date when the plant-specific LOCA analysis will be submitted in an ammendent.

RESPONSE

The plant-specific LOCA analysis ~~will be provided in July 1985.~~ has been completed. Section 6.3 has been revised to provide the results of the HCGS-specific analysis.

QUESTION 440.27 (SECTION 6.3)

The references provided for the ECCS analysis must include references for the latest model changes and corrections used in the HCGS analysis.

RESPONSE

The HCGS-specific ECCS analysis has been completed ~~will be provided in July 1985~~ and it ~~will~~ utilized the LOCA evaluation models approved by the NRC in Reference 1 and described in Reference 2.

REFERENCES

1. Letter to G. G. Sherwood (General Electric) from R. L. Tedesco (NRC), "Acceptance for Referencing of Topical Reports NEDE-20566P, NEDO-20566-1 Revision 1, and NEDE-20566-4 Amendment 4," February 4, 1981.

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K," NEDE-20566P, November 1975.

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QUESTION 440.28 (SECTION 6.3)

Justify selection of a Lead plant for the LOCA break spectrum analysis. HCGS is committed to submit a plant specific LOCA analysis. We require a schedule for submittal of the plant specific LOCA analysis.

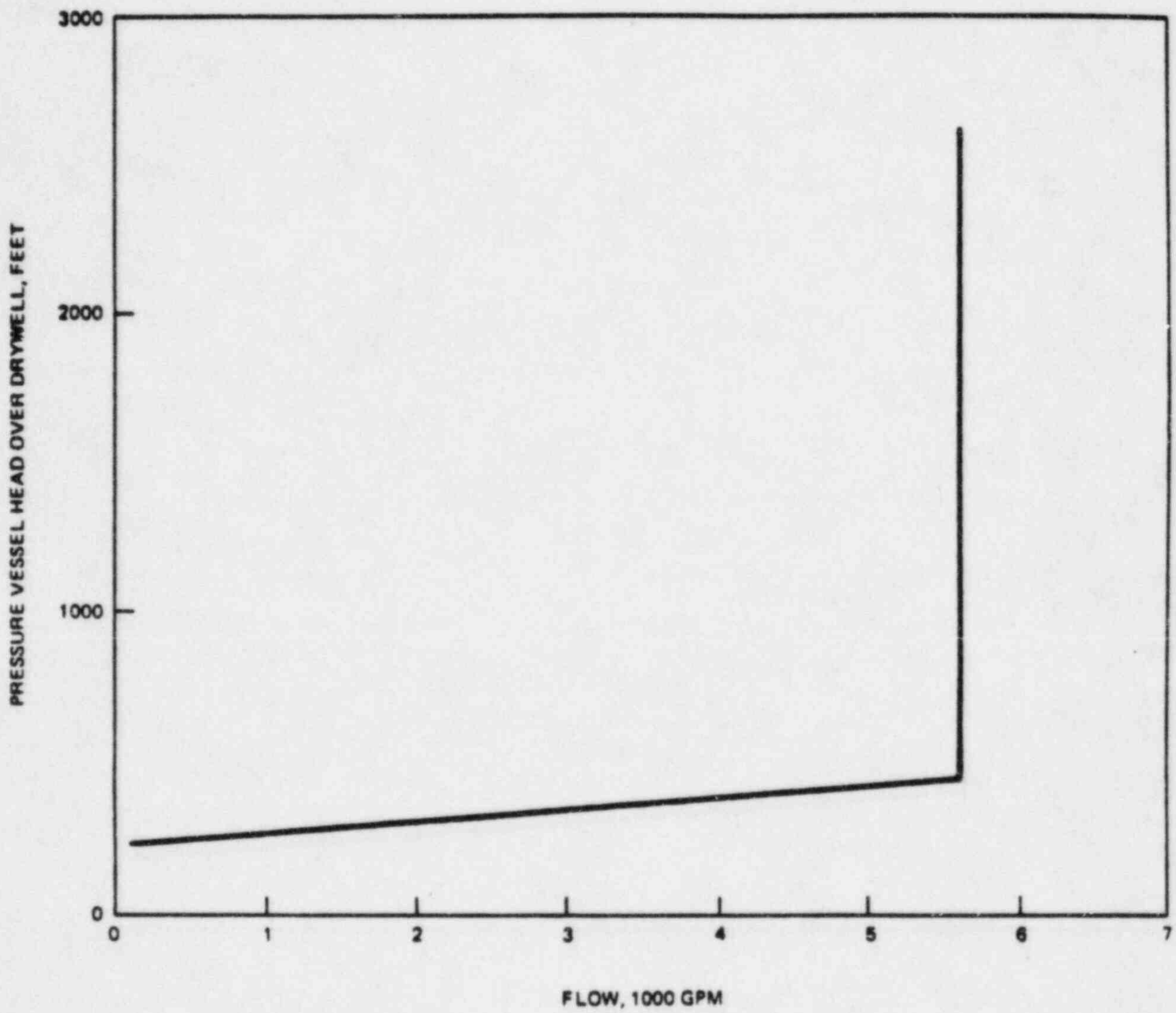
RESPONSE

The lead-plant LOCA analysis is an appropriate and representative break-spectrum analysis for the HCGS because the LOCA characteristics of BWR plants with similar ECCS configuration have been shown to be quite similar. The lead-plant analysis serves to identify the limiting failures and breaks and to describe the general LOCA characteristics of these plants. Lead-plant sensitivity studies have demonstrated that the location of the limiting break is insensitive to slight variations in ECCS configuration and to changes in power level or fuel type. HCGS-specific analyses will be provided at the limiting locations to define the specific HCGS response for the limiting cases. This is the basis of the lead-plant concept.

~~The results of~~ the HCGS-specific ECCS analysis ~~will be submitted in July 1985. See the response to Question 440.27 for a discussion of the LOCA evaluation model that will be used.~~

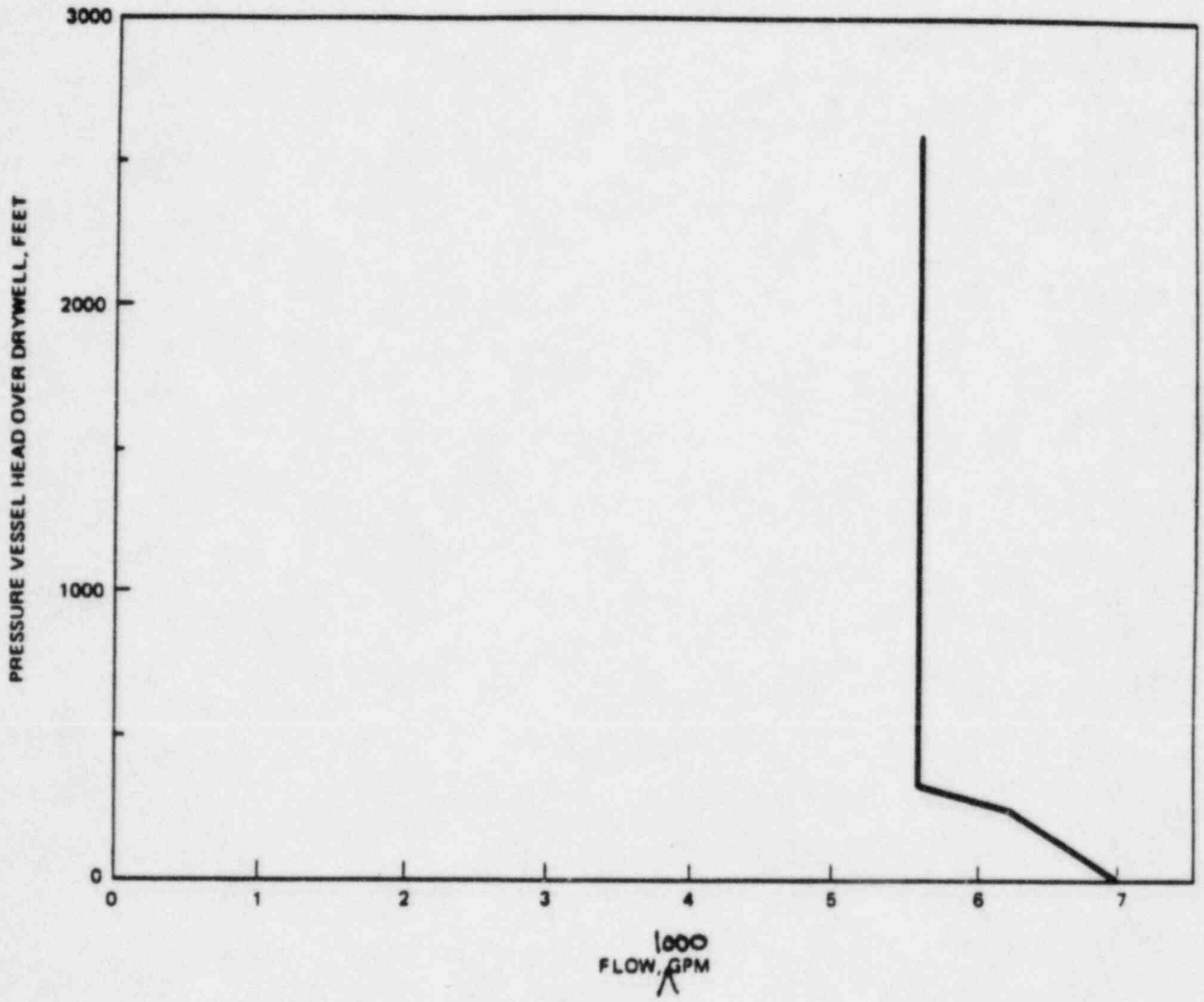
has been completed and section 6.3 has been revised to provide the HCGS-specific results.

This replaces Figure 6.3-4



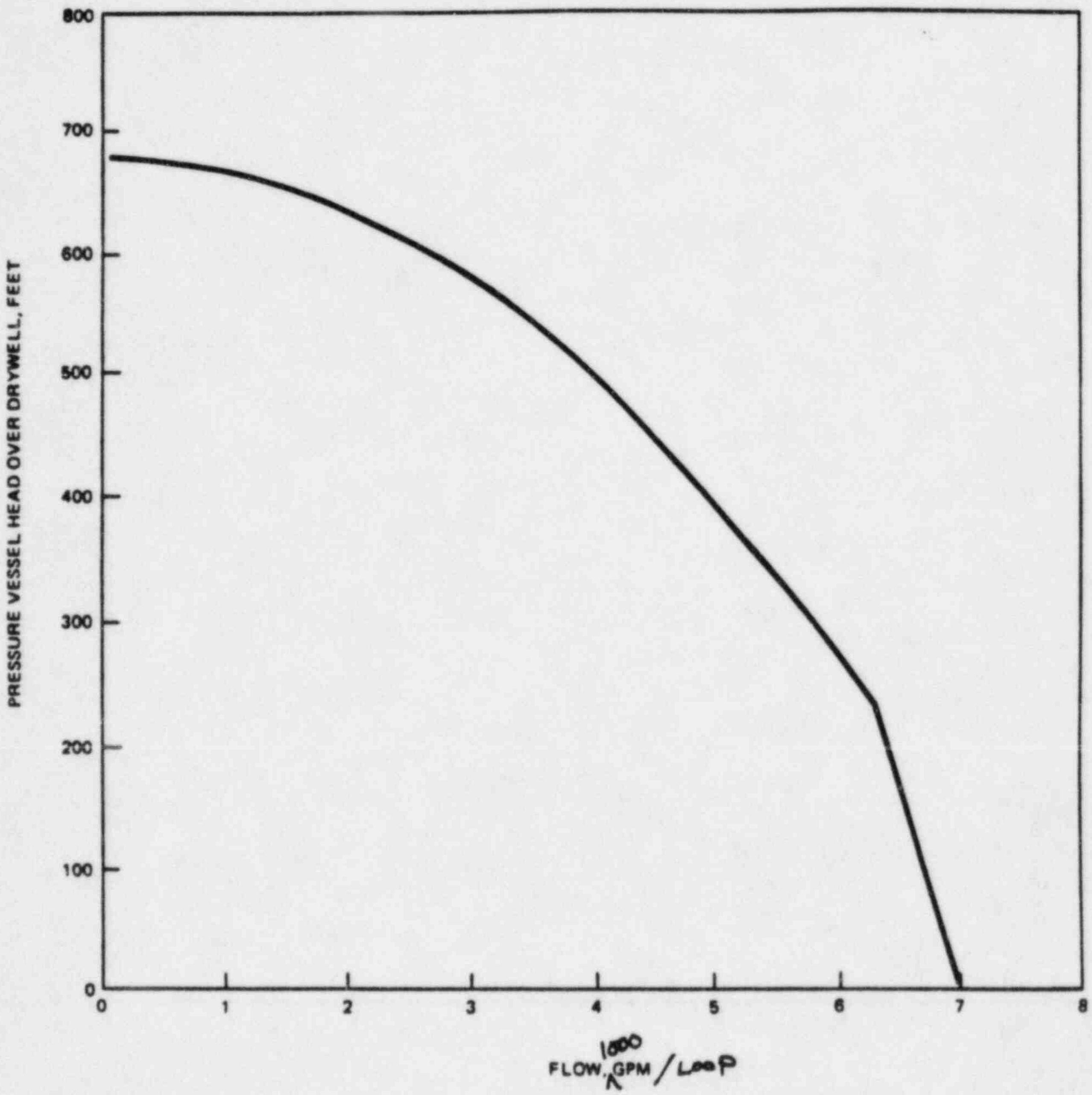
SER ITEM C-14

This replaces Figure 6.3-5



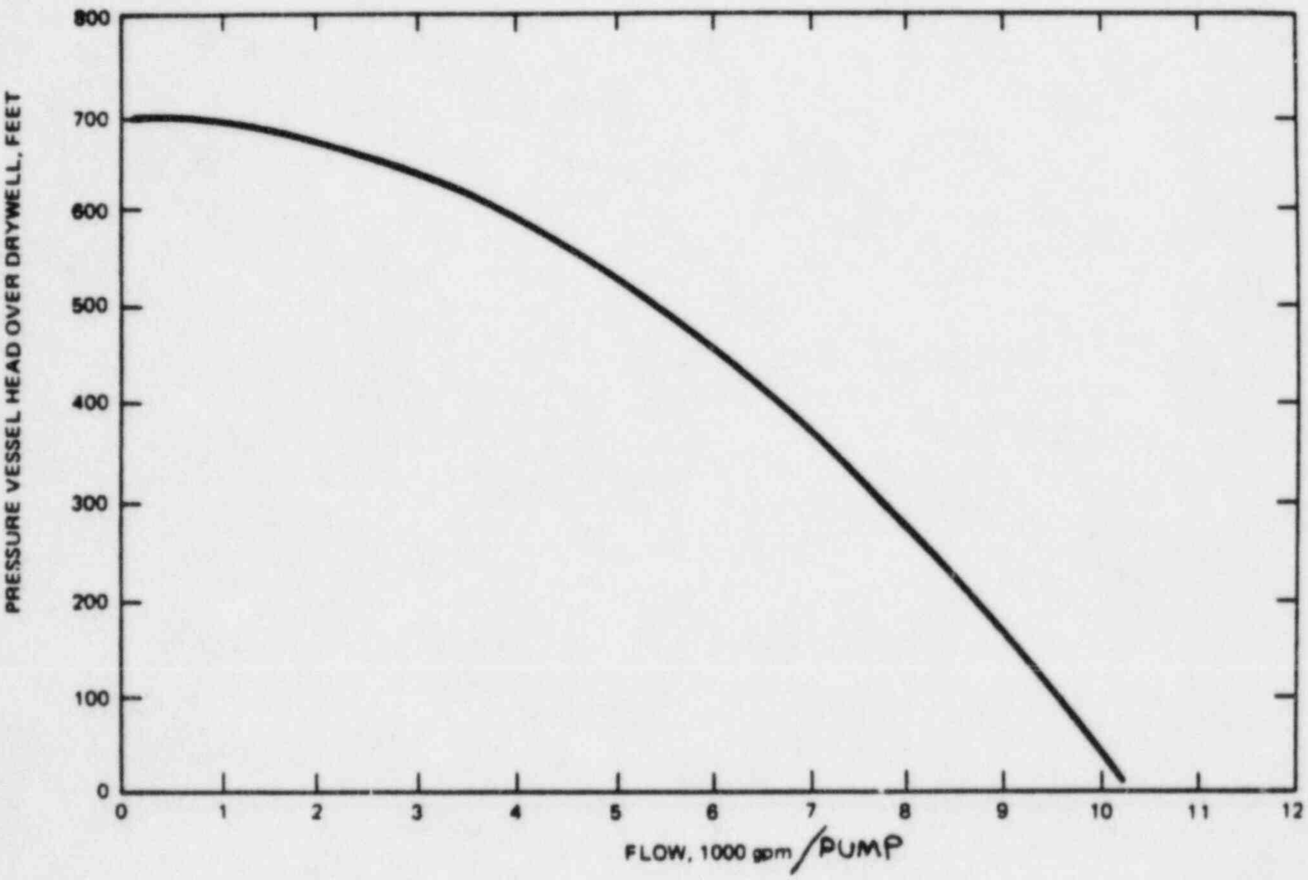
SER ITEM C-14

This replaces Figure 6.3-9



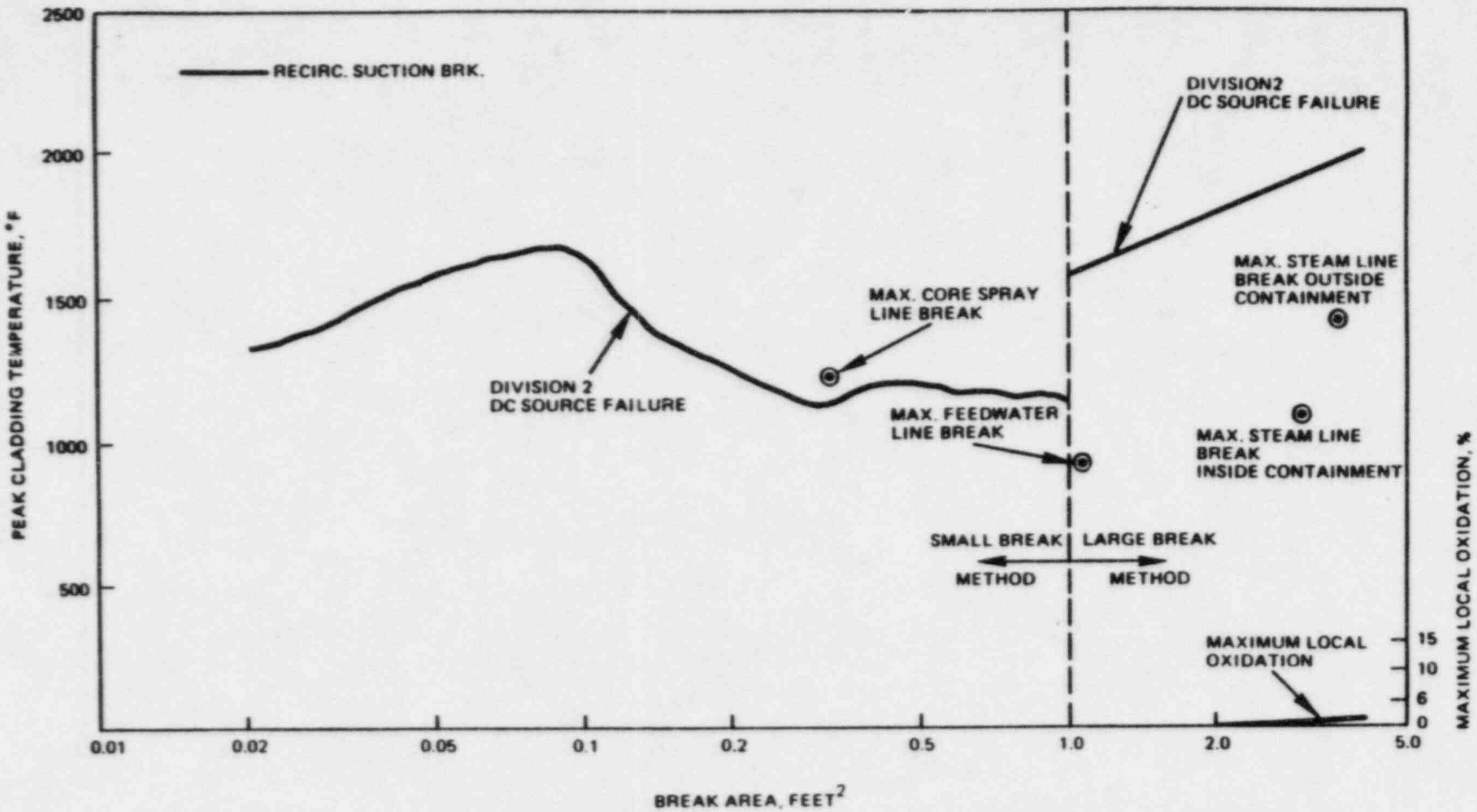
SER ITEM C-14

This replaces Figure 6.3-11



SER ITEM C-14

This replaces Figure 6.3-14

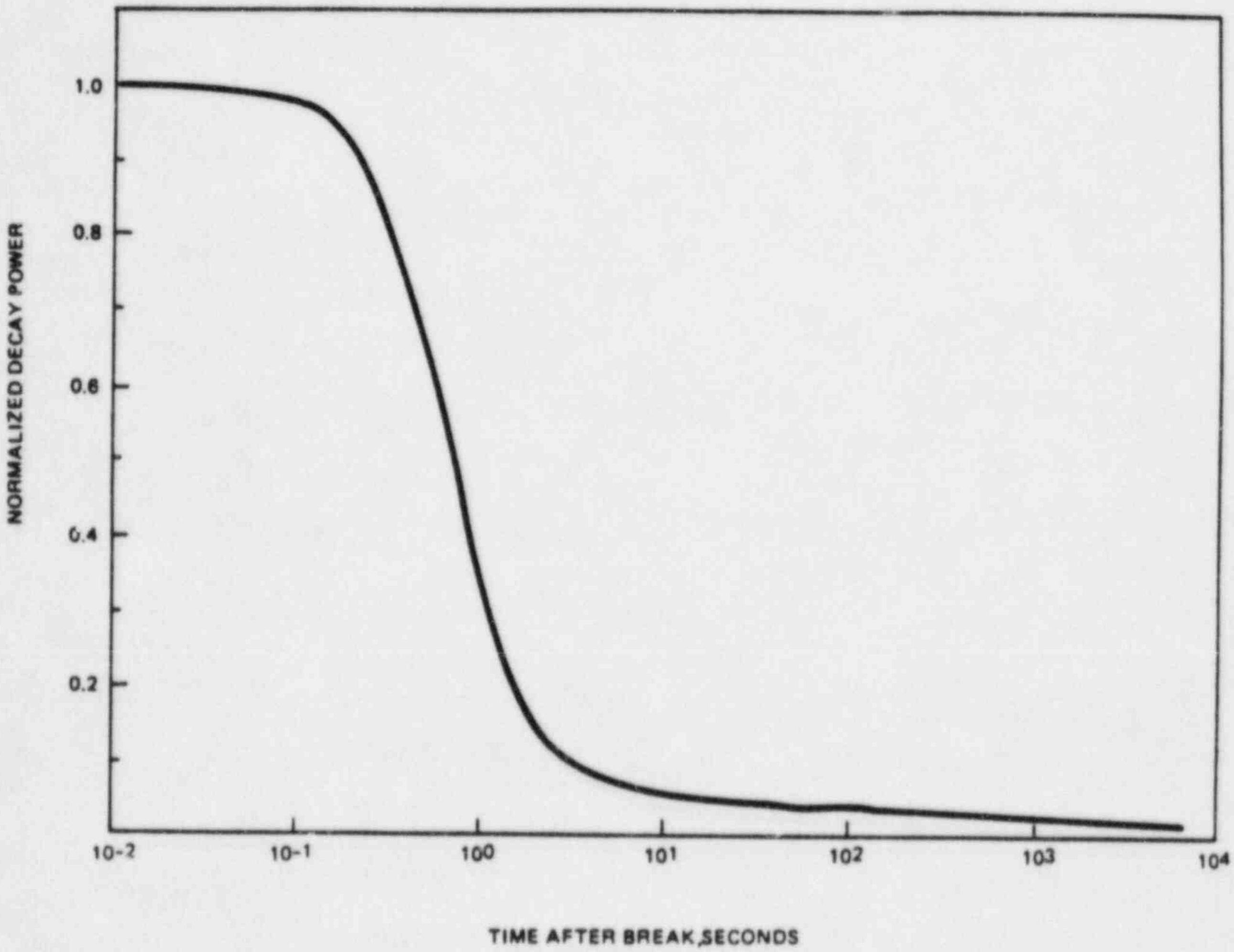


SER ITEM C-14

Title: Peak Cladding Temperature and Maximum Local Oxidation

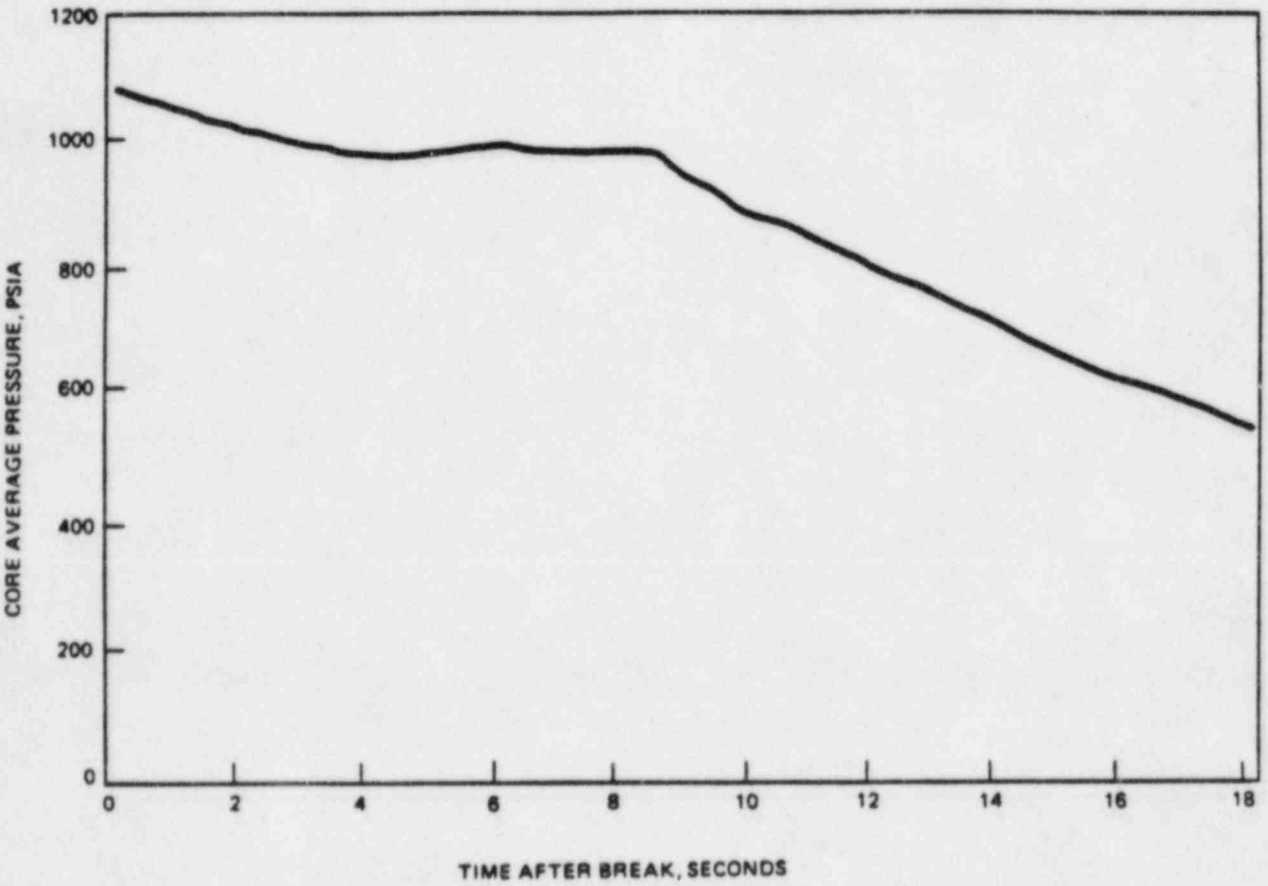
6.3-15

This replaces Figure 6.3-15.



SER ITEM C-14

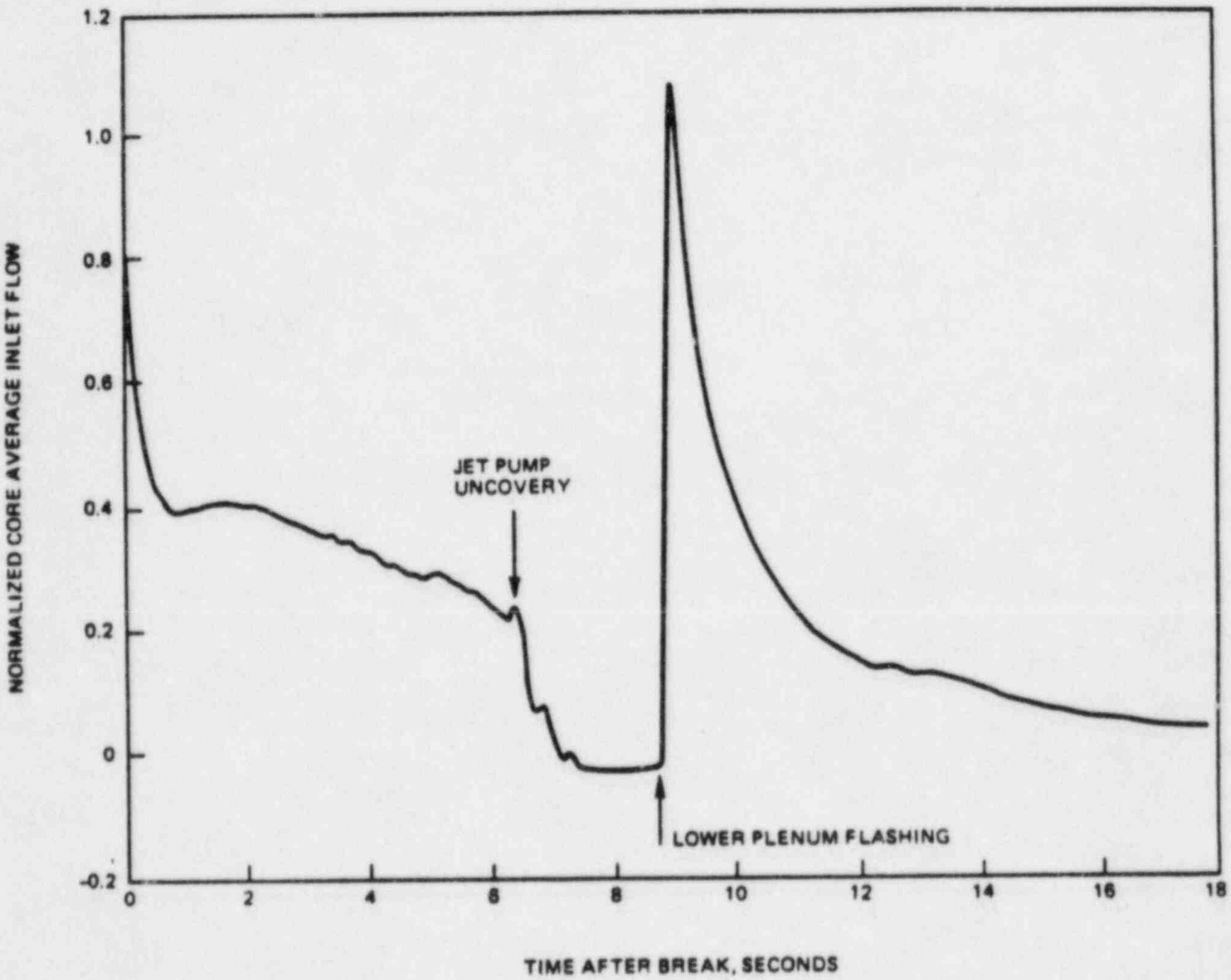
This replaces Figure 6.3-16



SER ITEM C-14

i ... 7

This replaces Figure 6.3-17.



SER ITEM C-14

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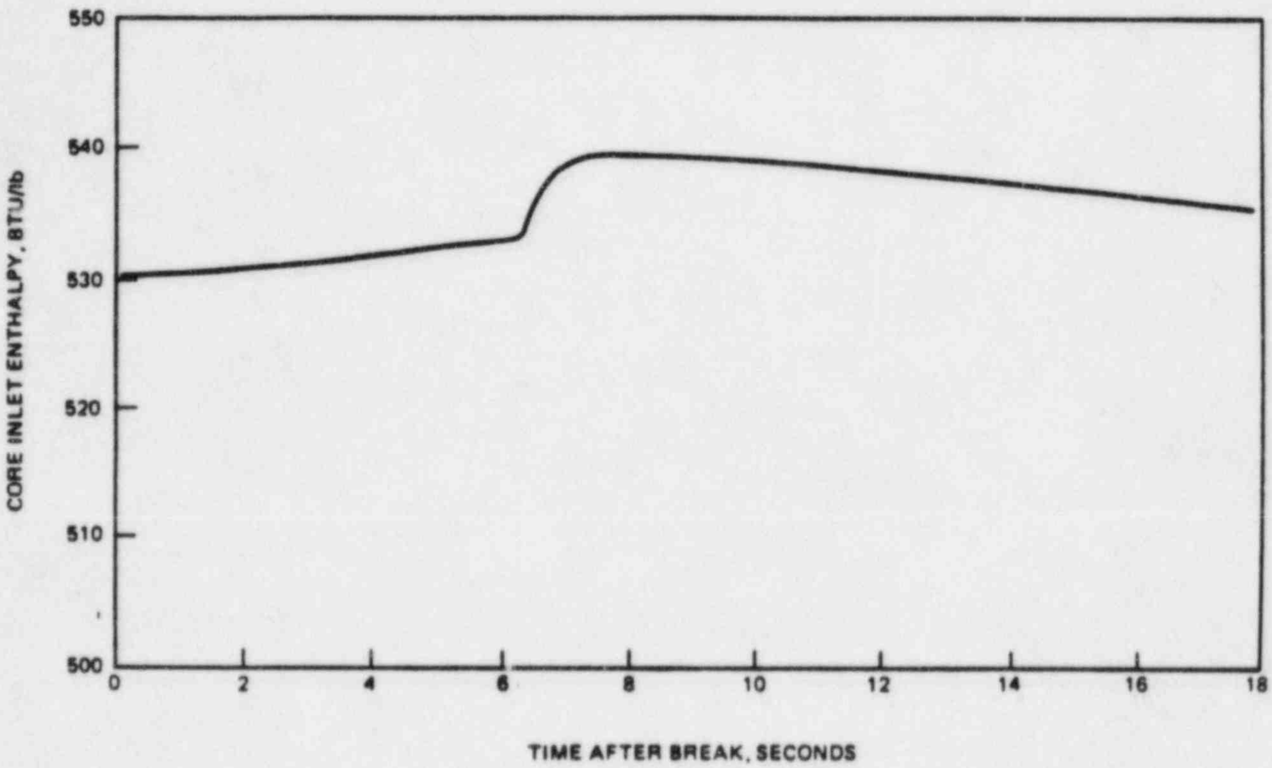
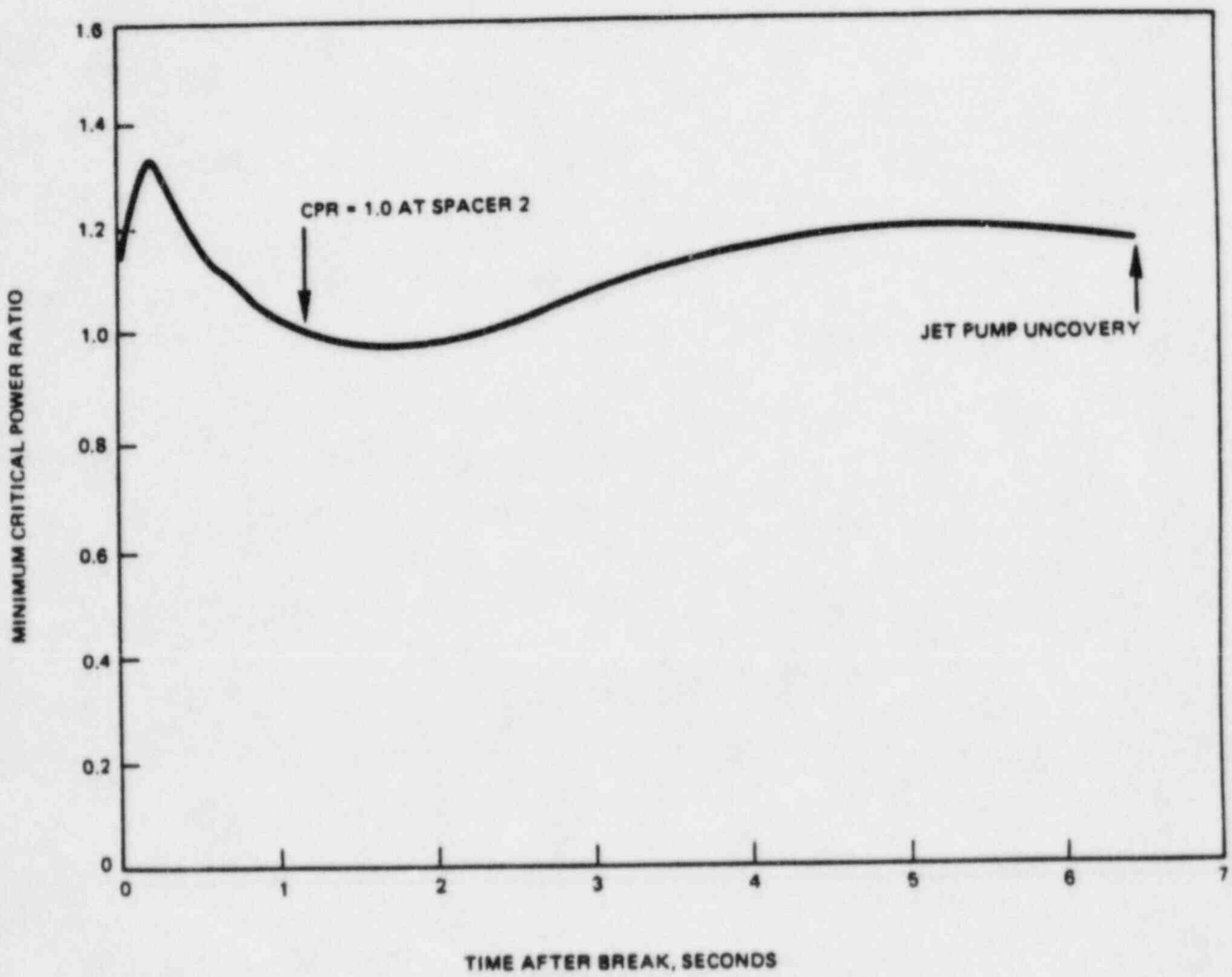


SER ITEM C-14

This replaces Figure 6.3-19.



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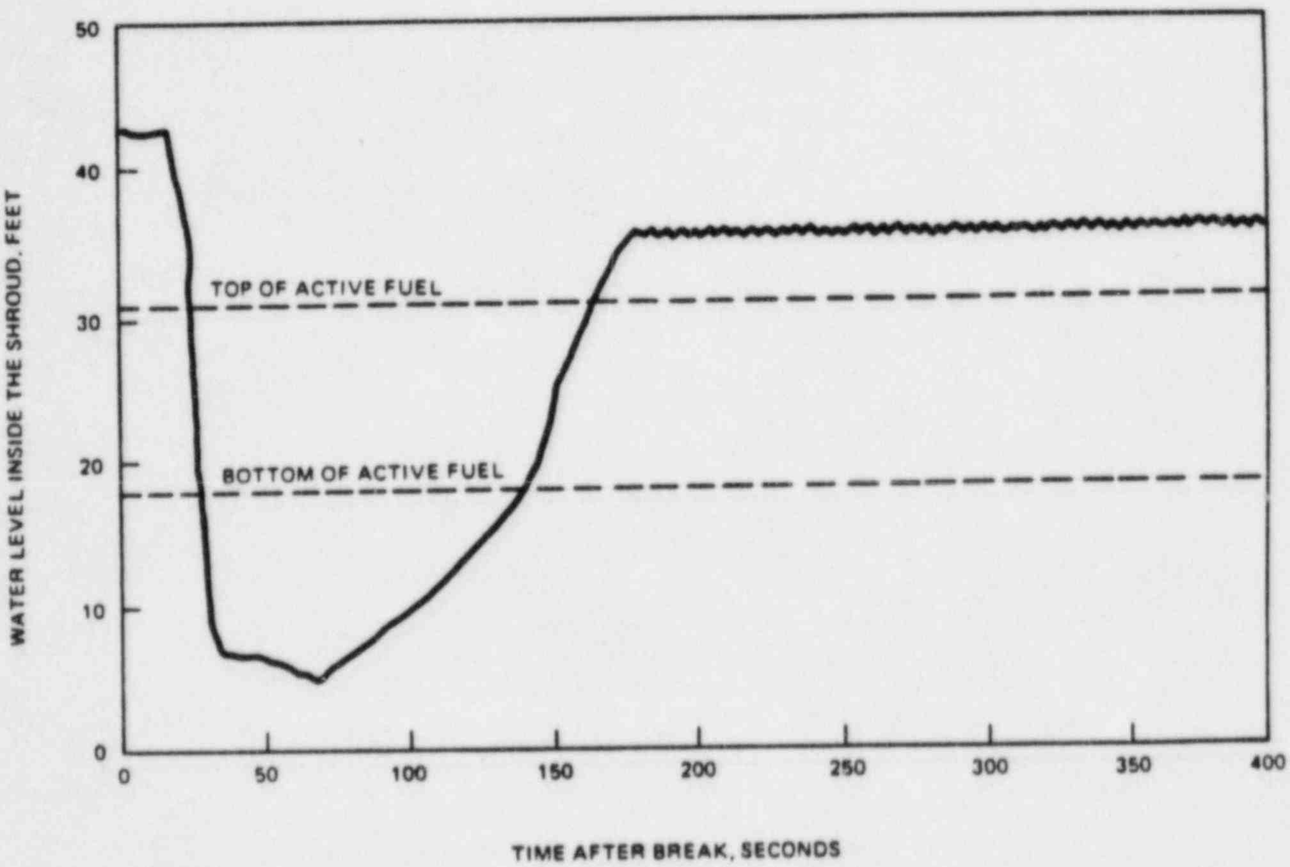
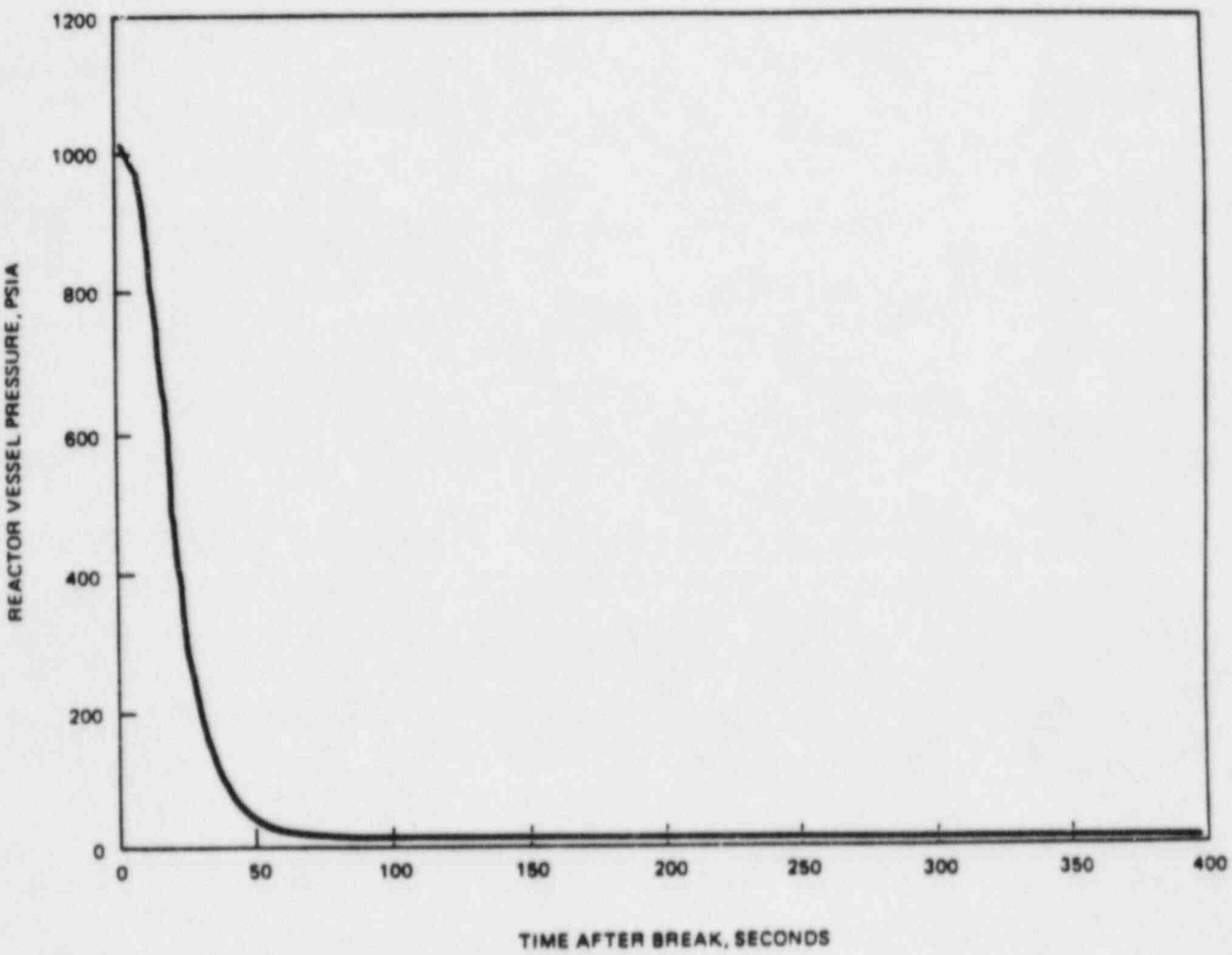


SER ITEM C-14

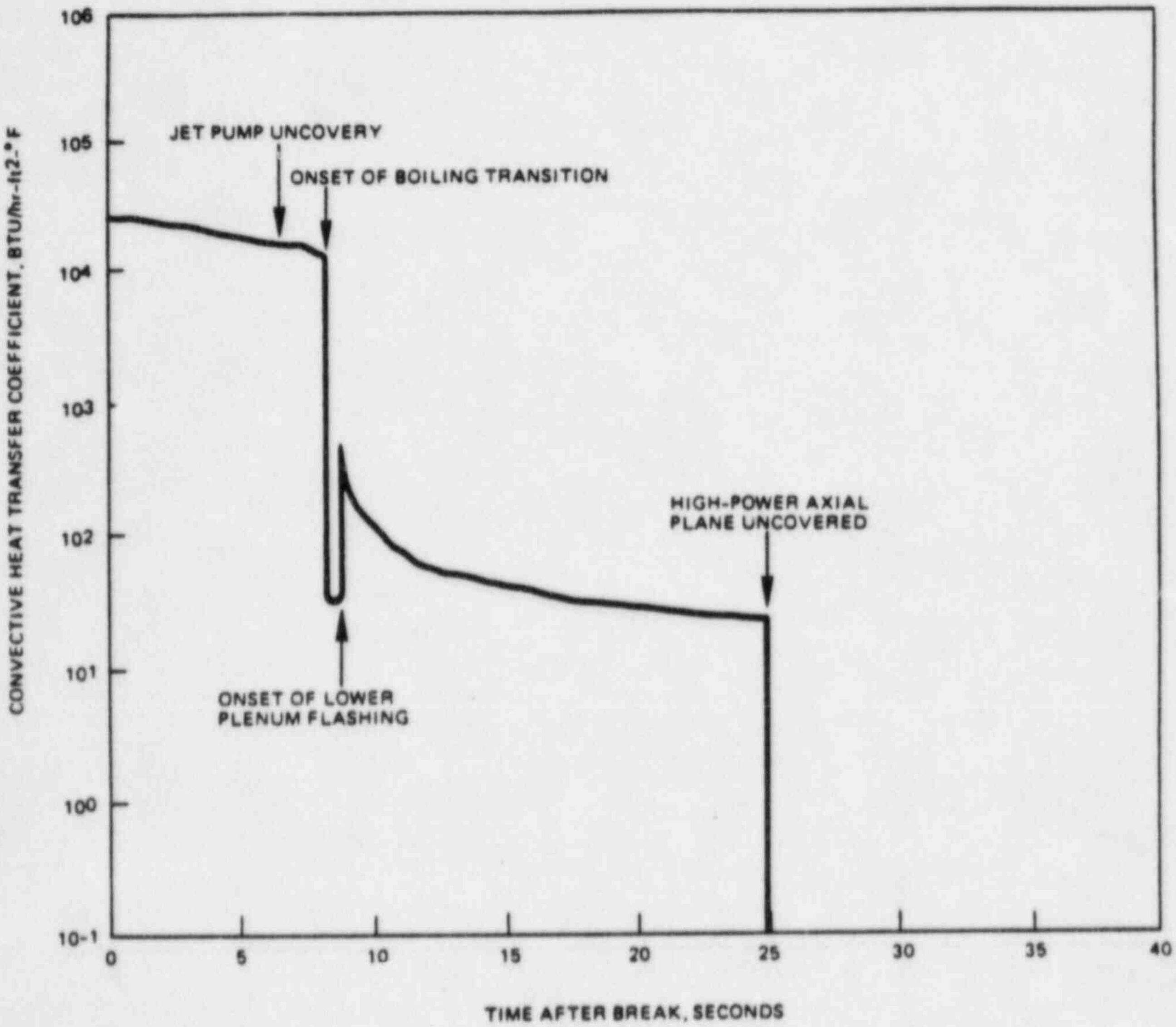
6.3-21

This replaces Figure 6.3-21.



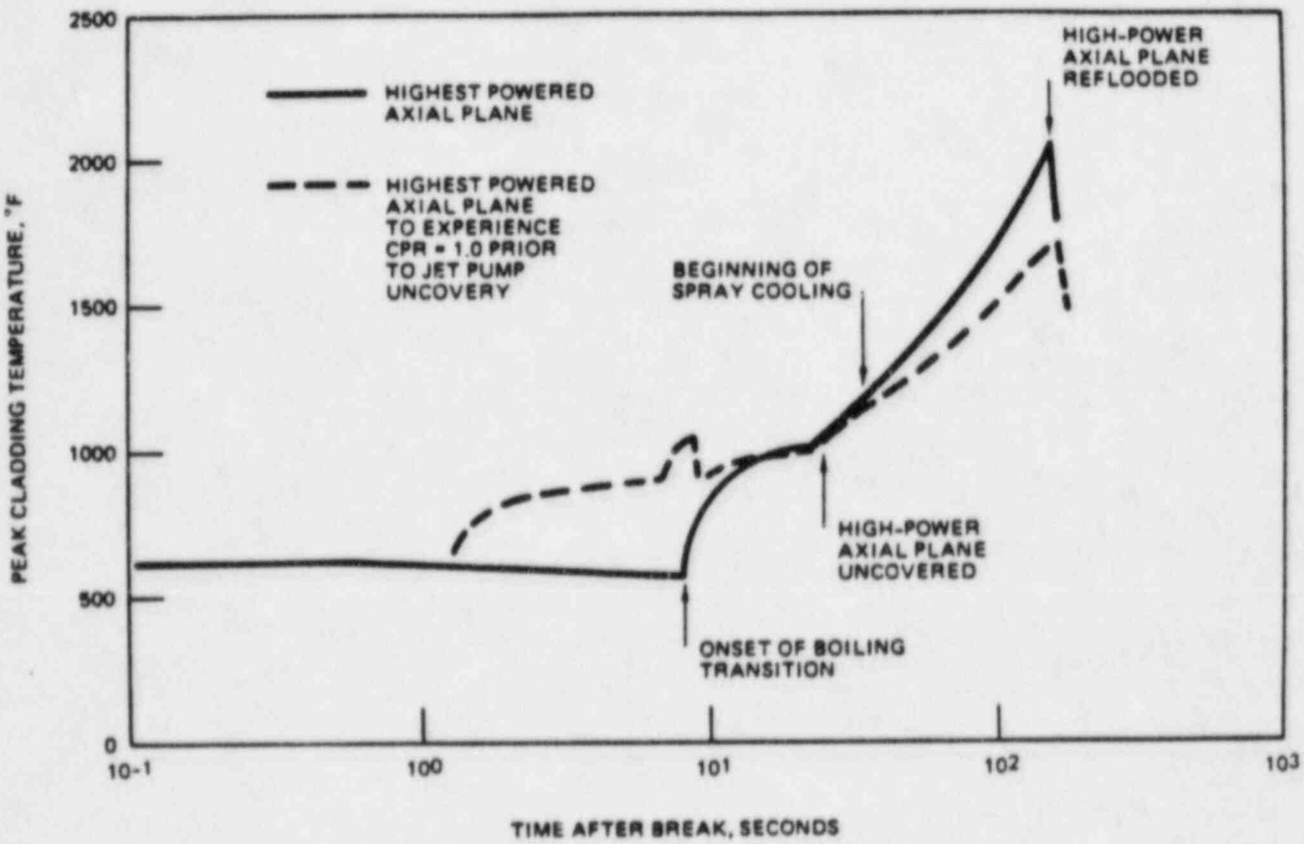
SER ITEM C-14

This replaces Figure 6.3-22.



SCR ITEM C-14

This replaces Figure 6.3-23



SCR ITEM C-14

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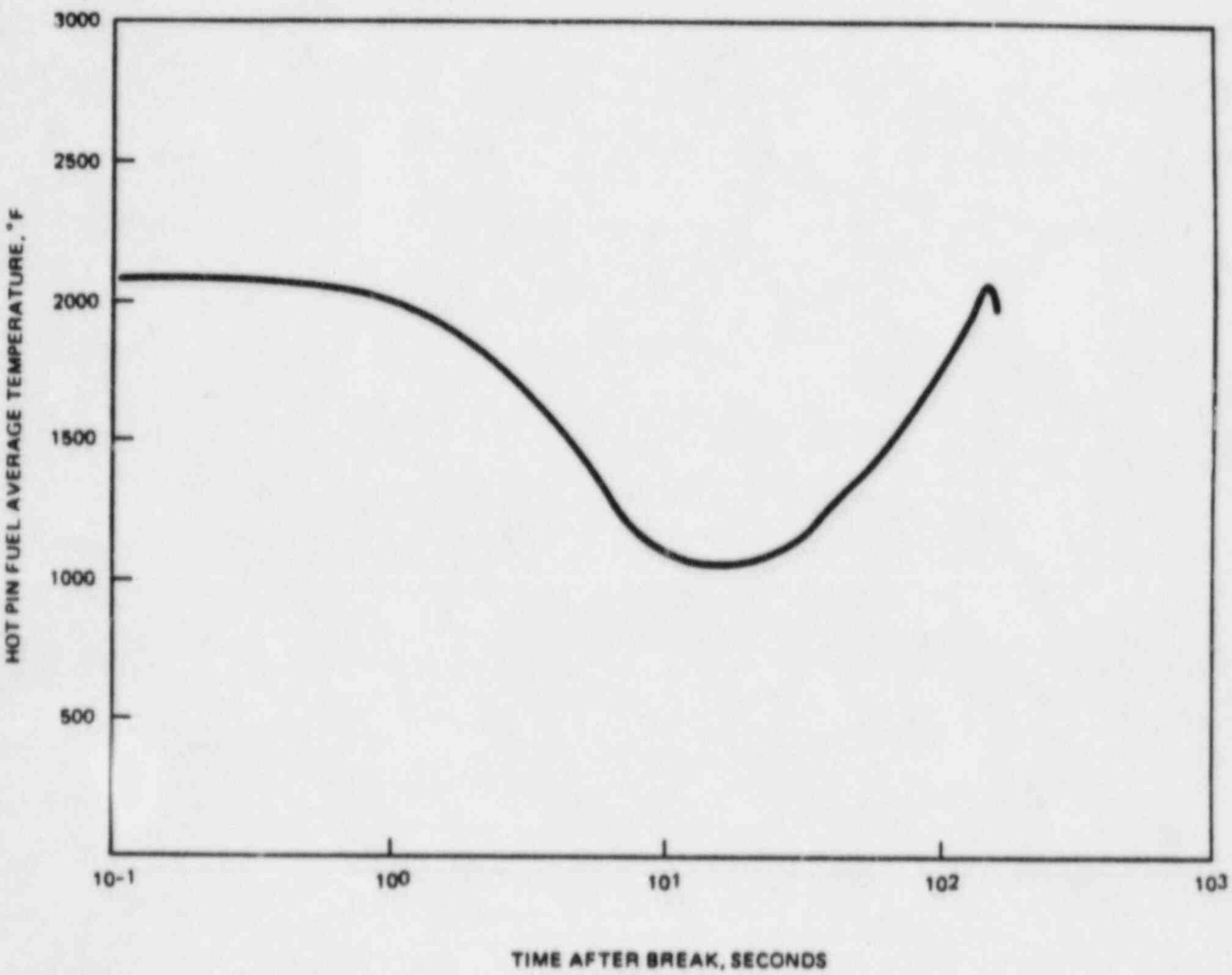
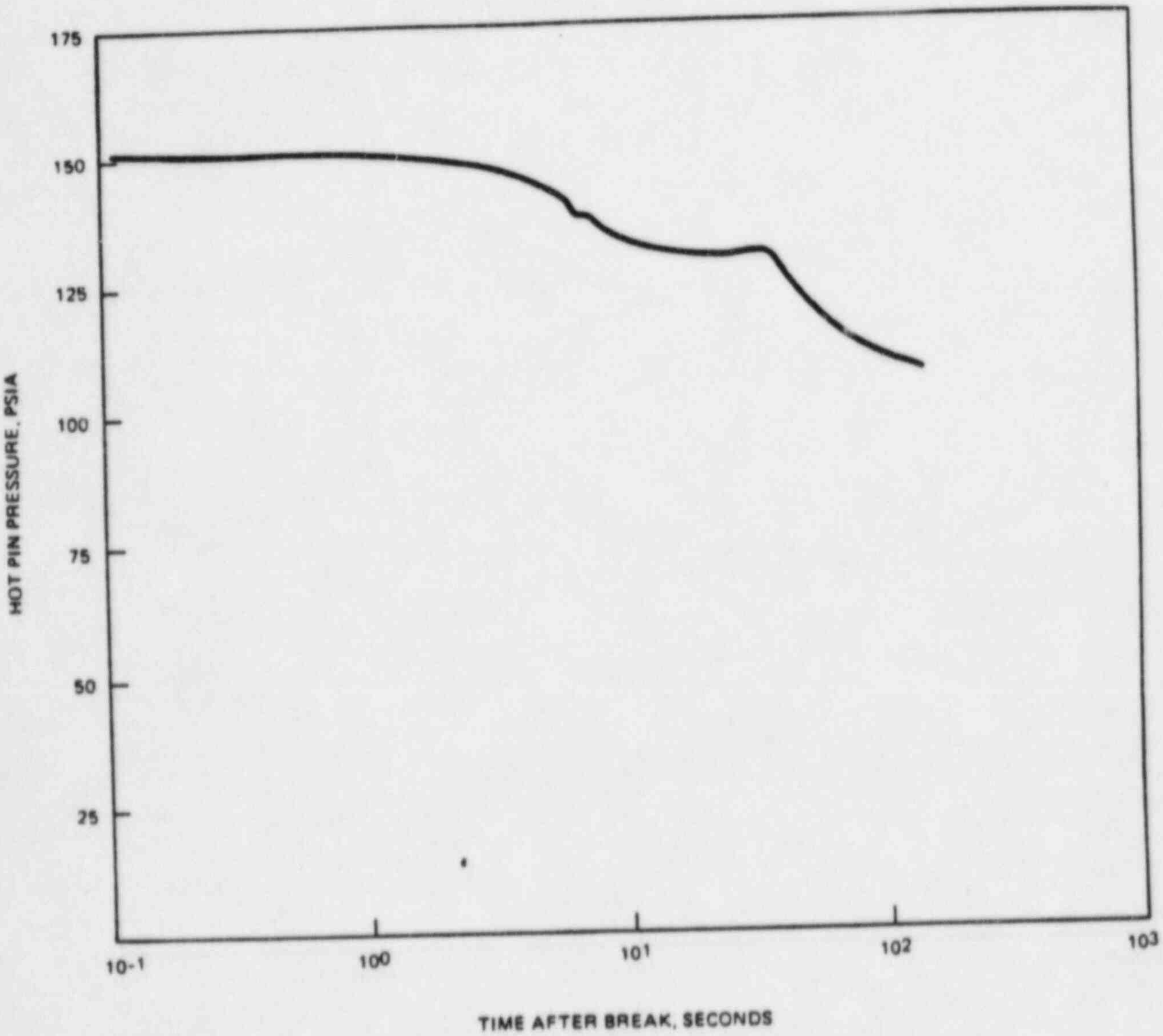


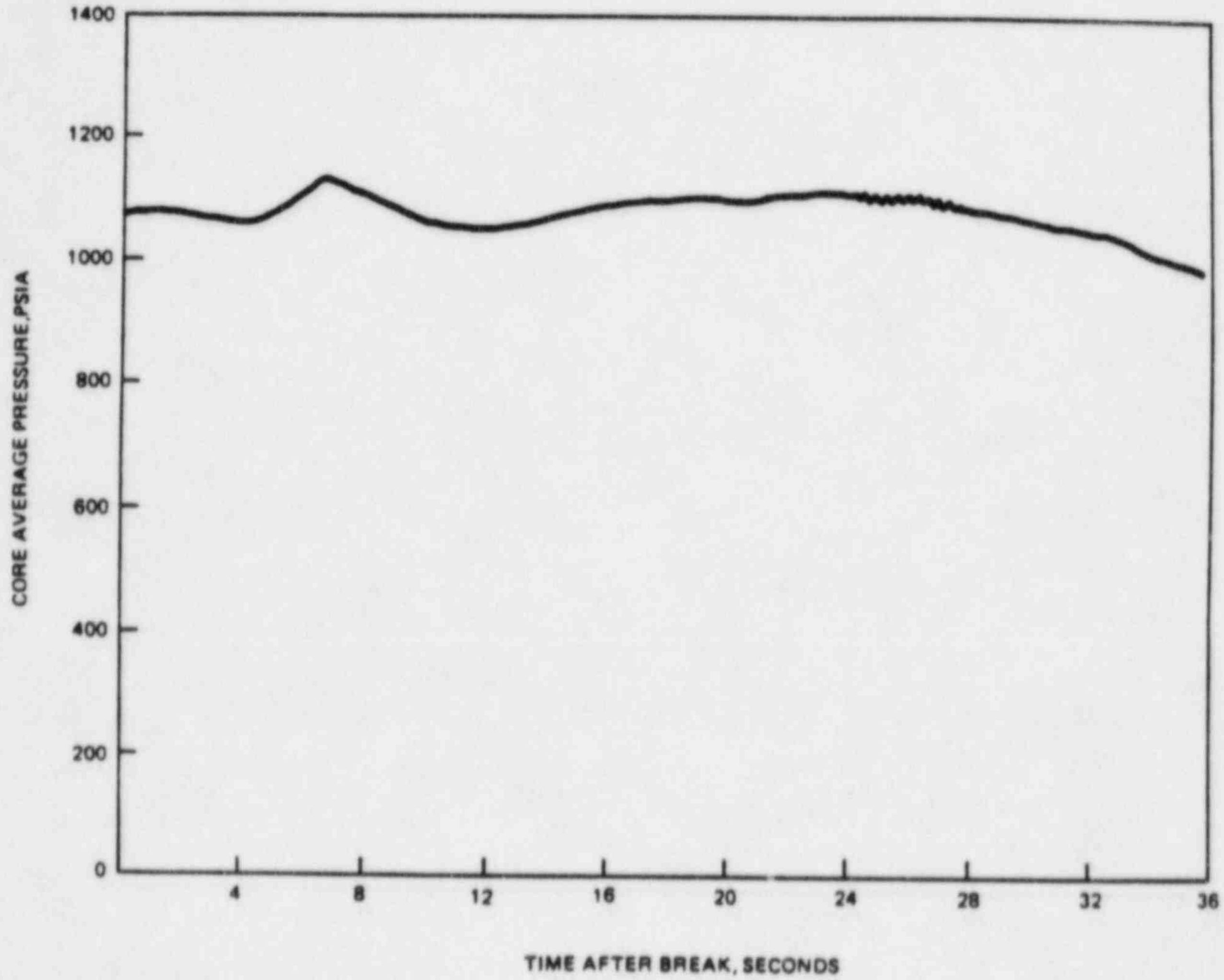
SER ITEM C-14

This replaces Figure 6.3-25.



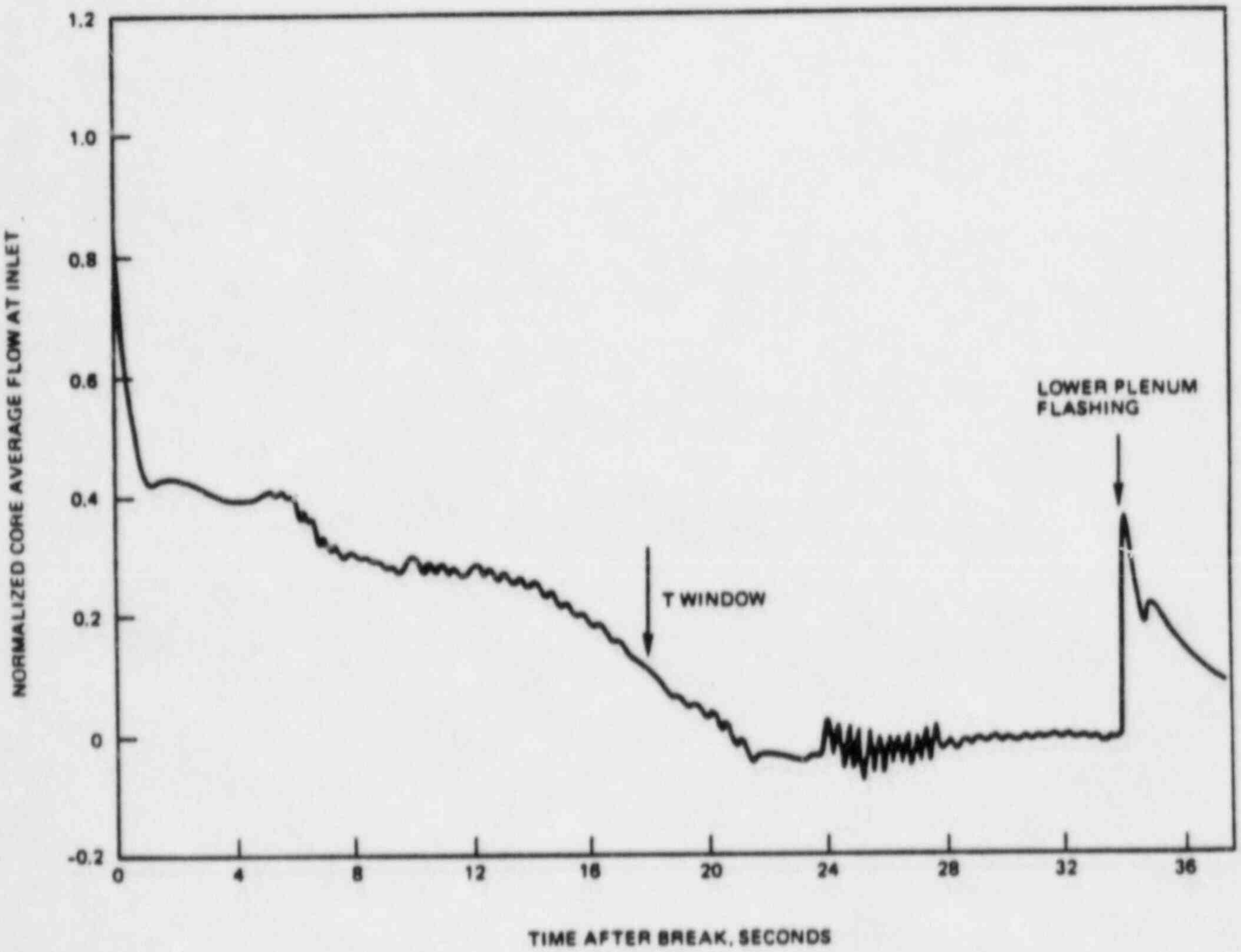
SER ITEM C-14

This replaces Figure 6.3-26



SER ITEM C.14

This replaces Figure 6.3-27.



SCR ITEM C-14

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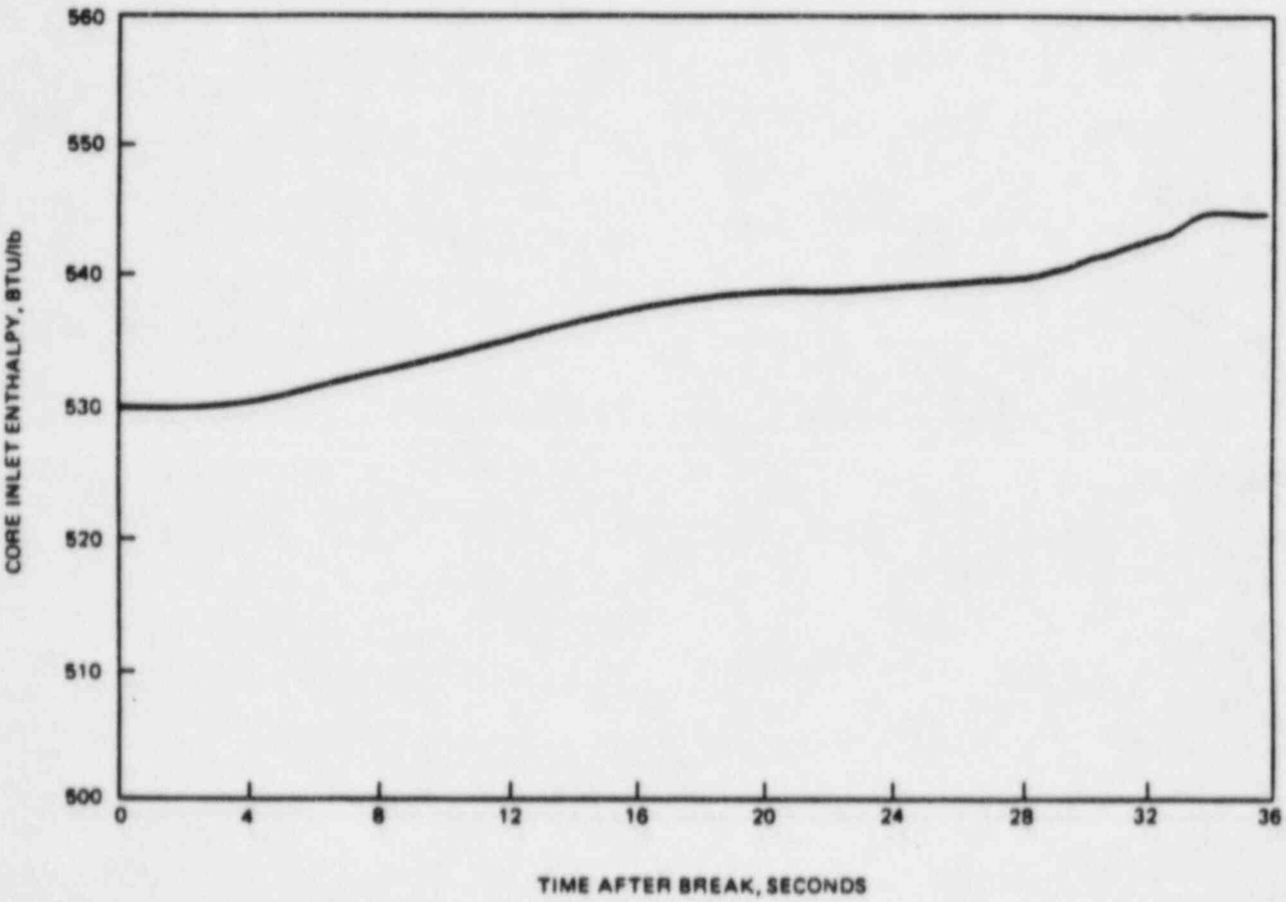
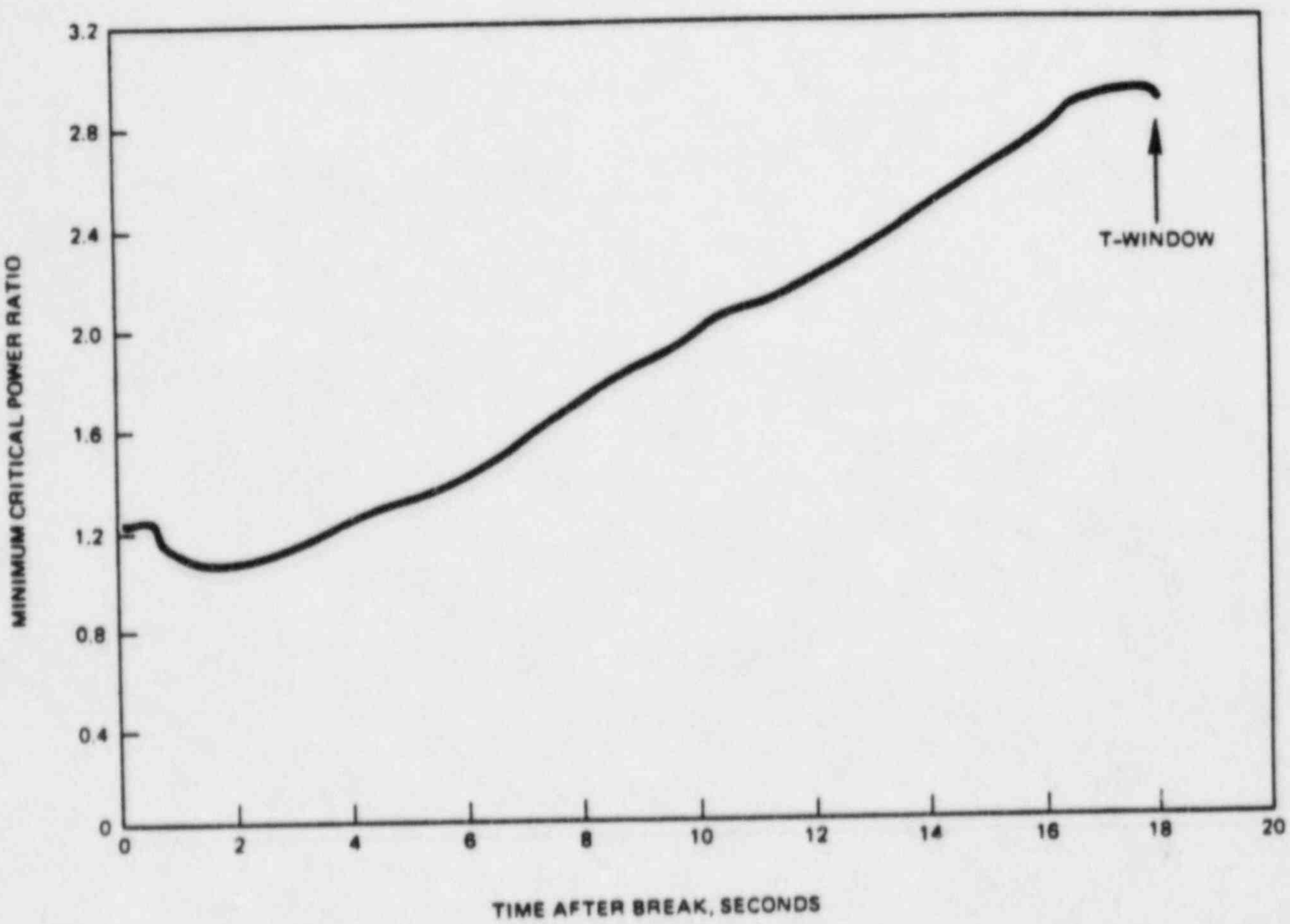


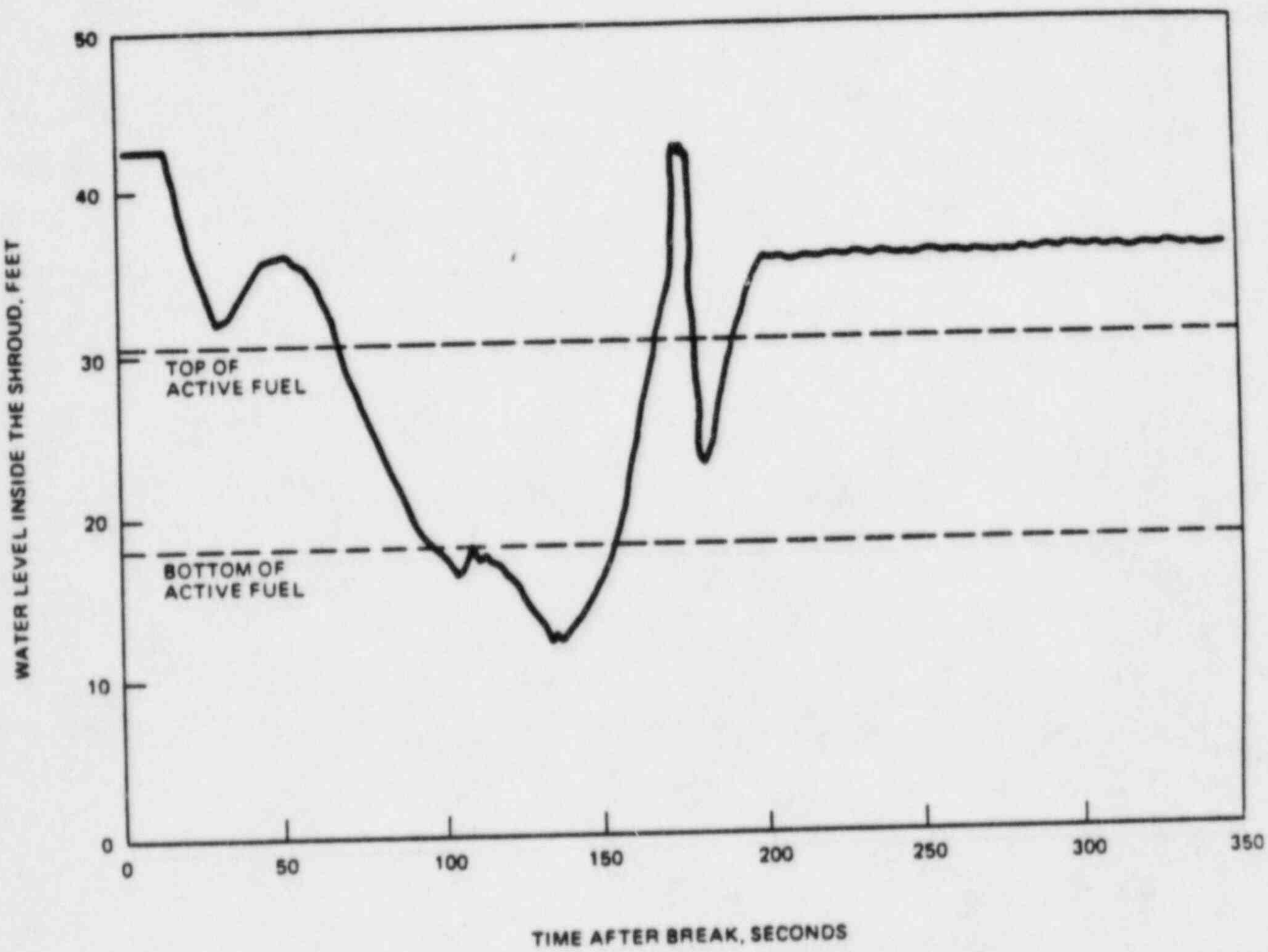
SER ITEM C-14

This replaces Figure 6.3-29.



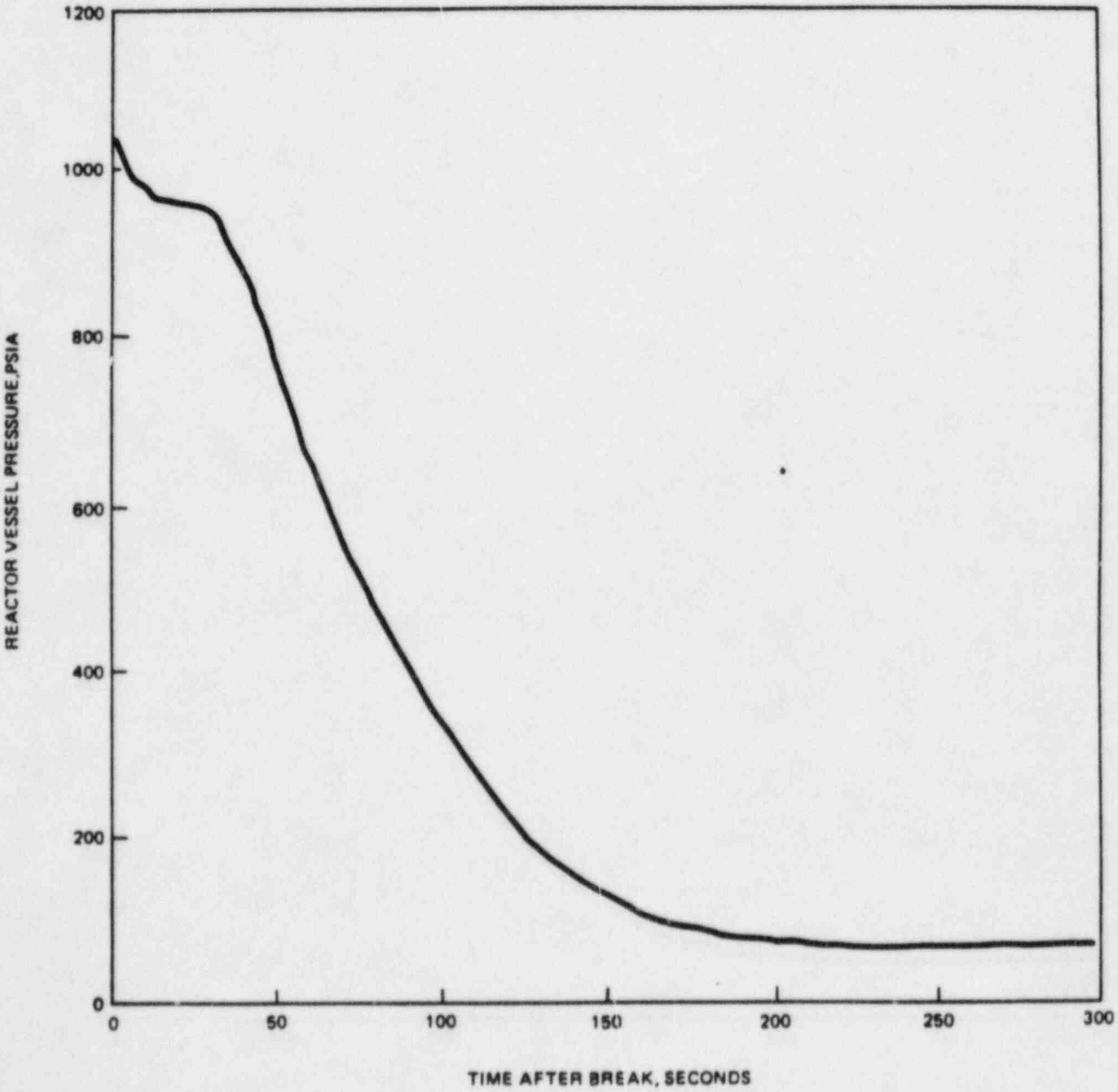
SCR ITEM C-14

This replaces Figure 6.3-30.



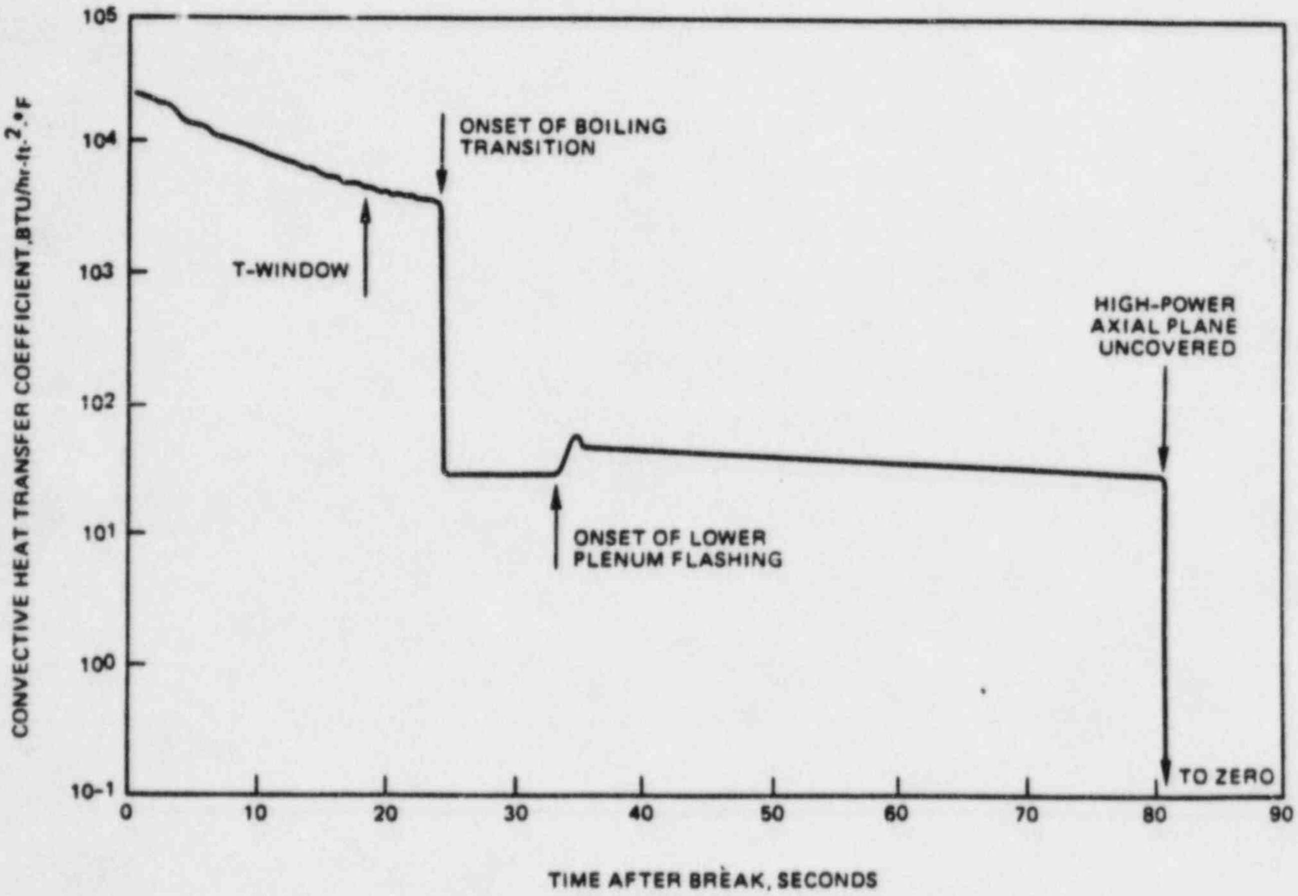
SER ITEM C-14

This replaces Figure 6.3-31.



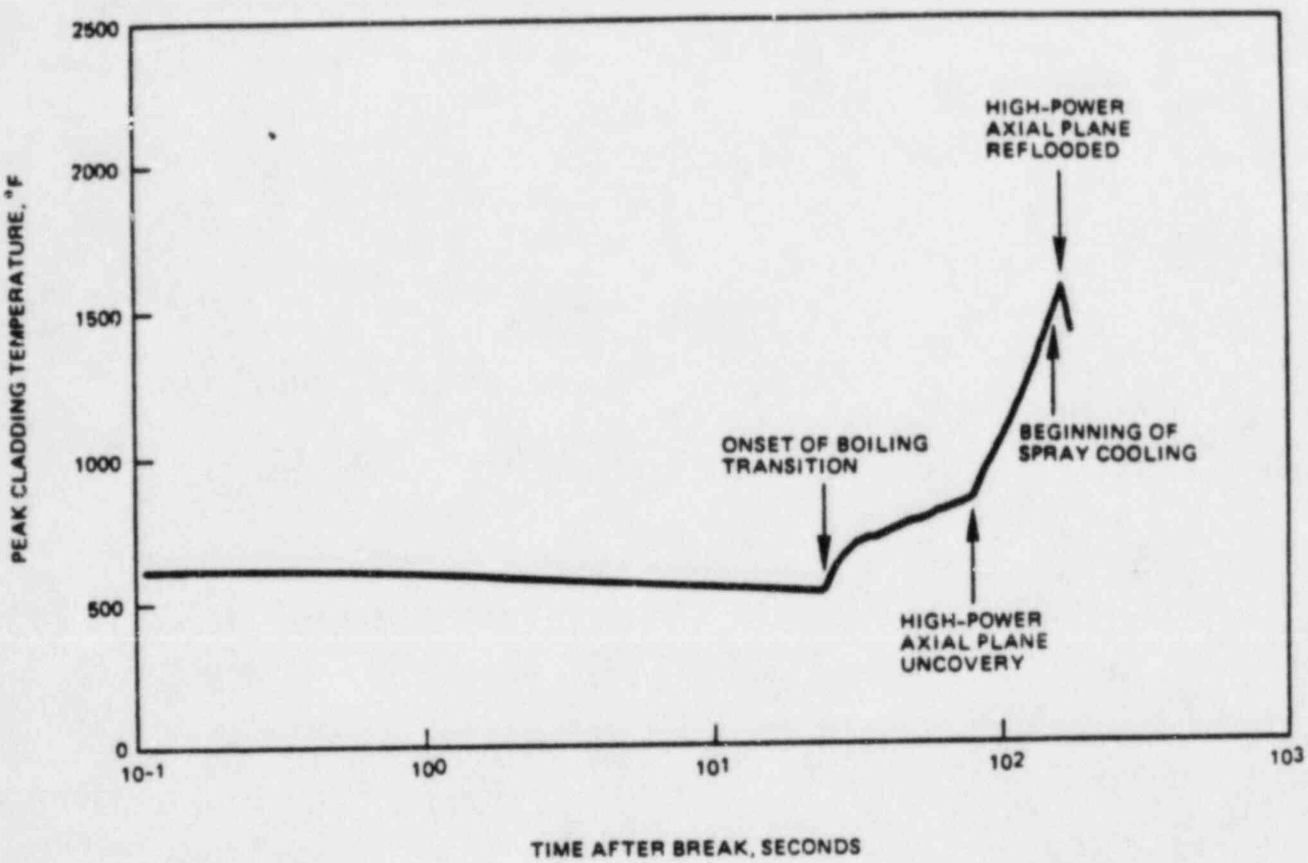
SER ITEM C-14

This replaces Figure 6.3-32.



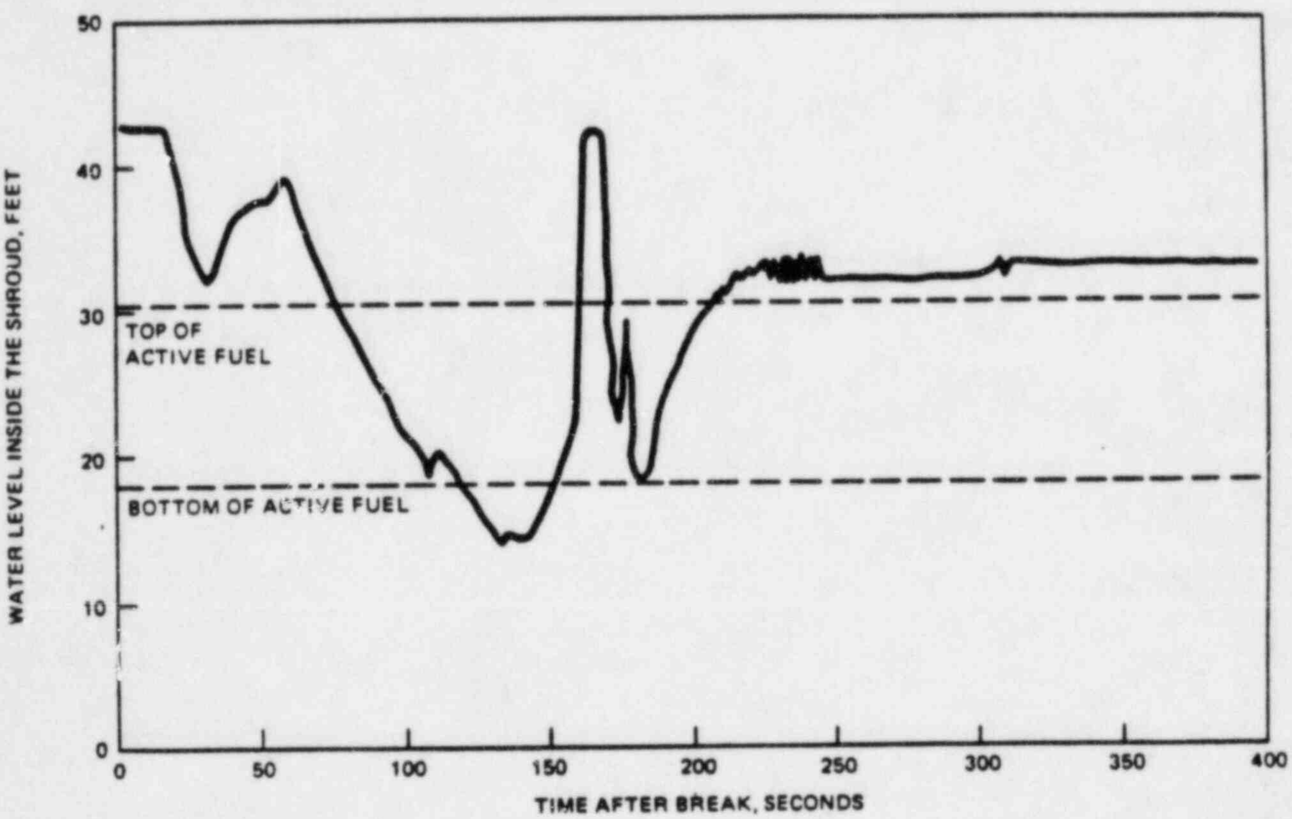
SER ITEM C-14

This replaces Figure 6.3-33.



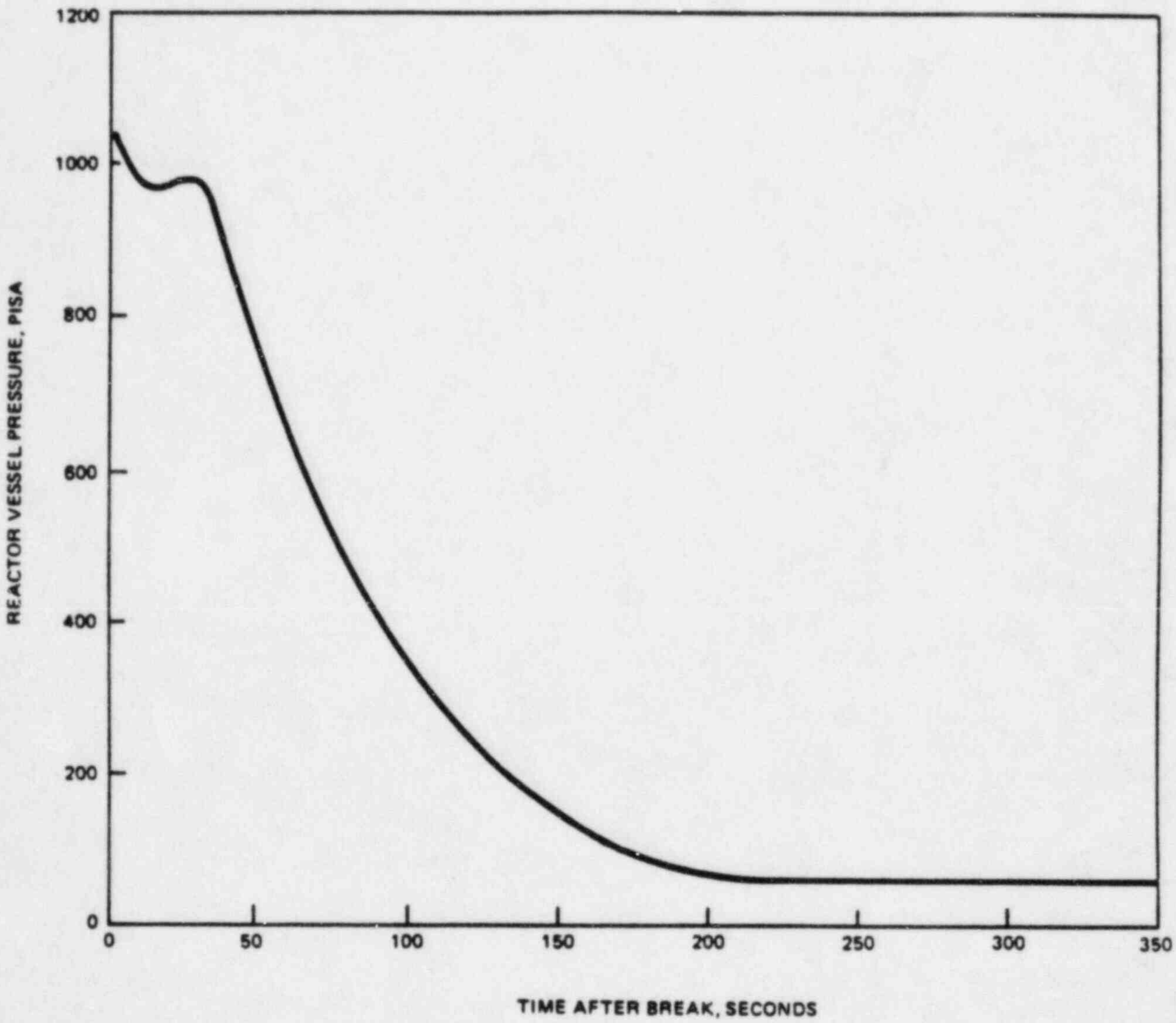
SER ITEM C-14

This replaces Figure 6.3-34.



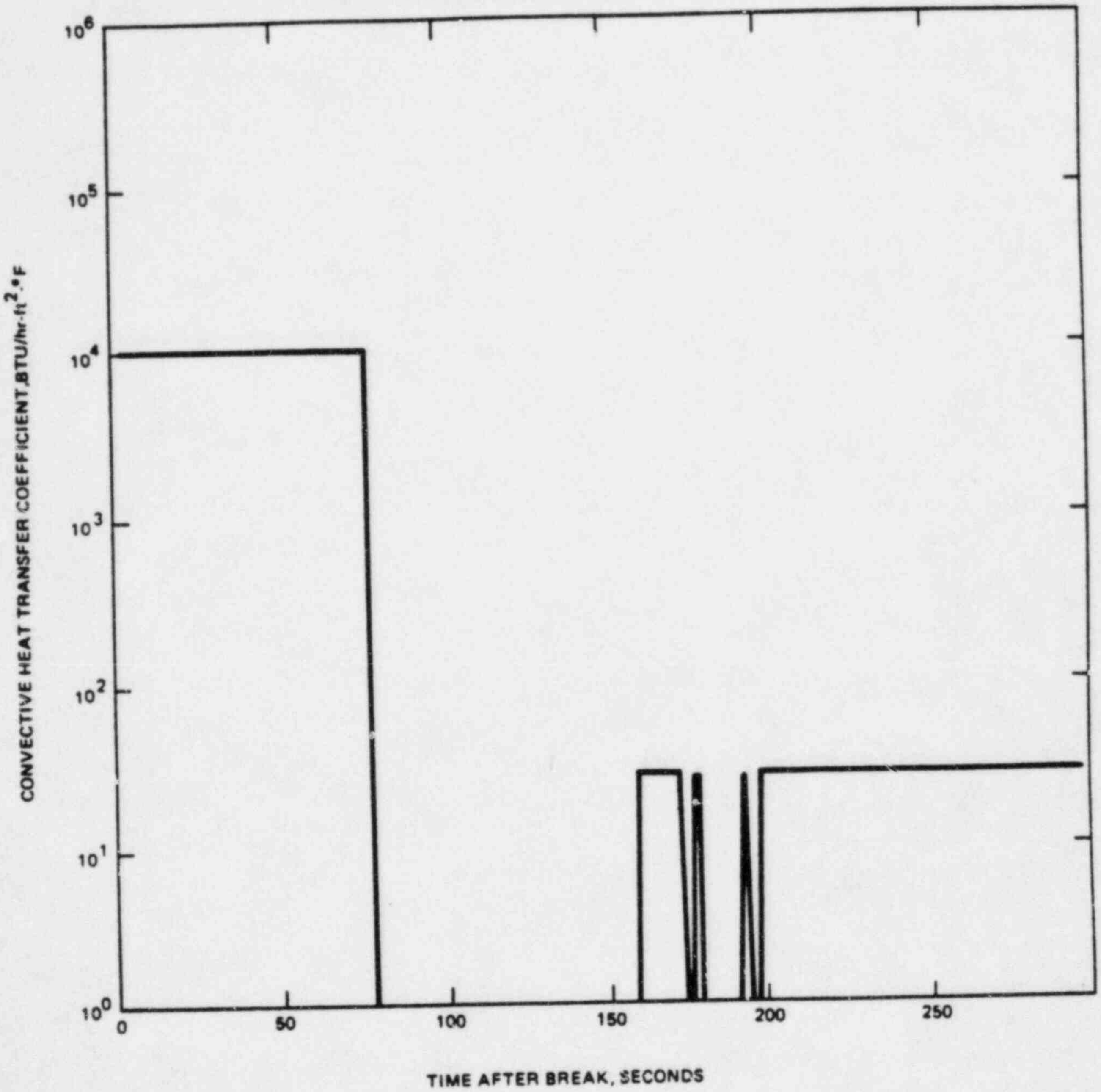
SER ITEM C-14

This replaces Figure 6.3-35.



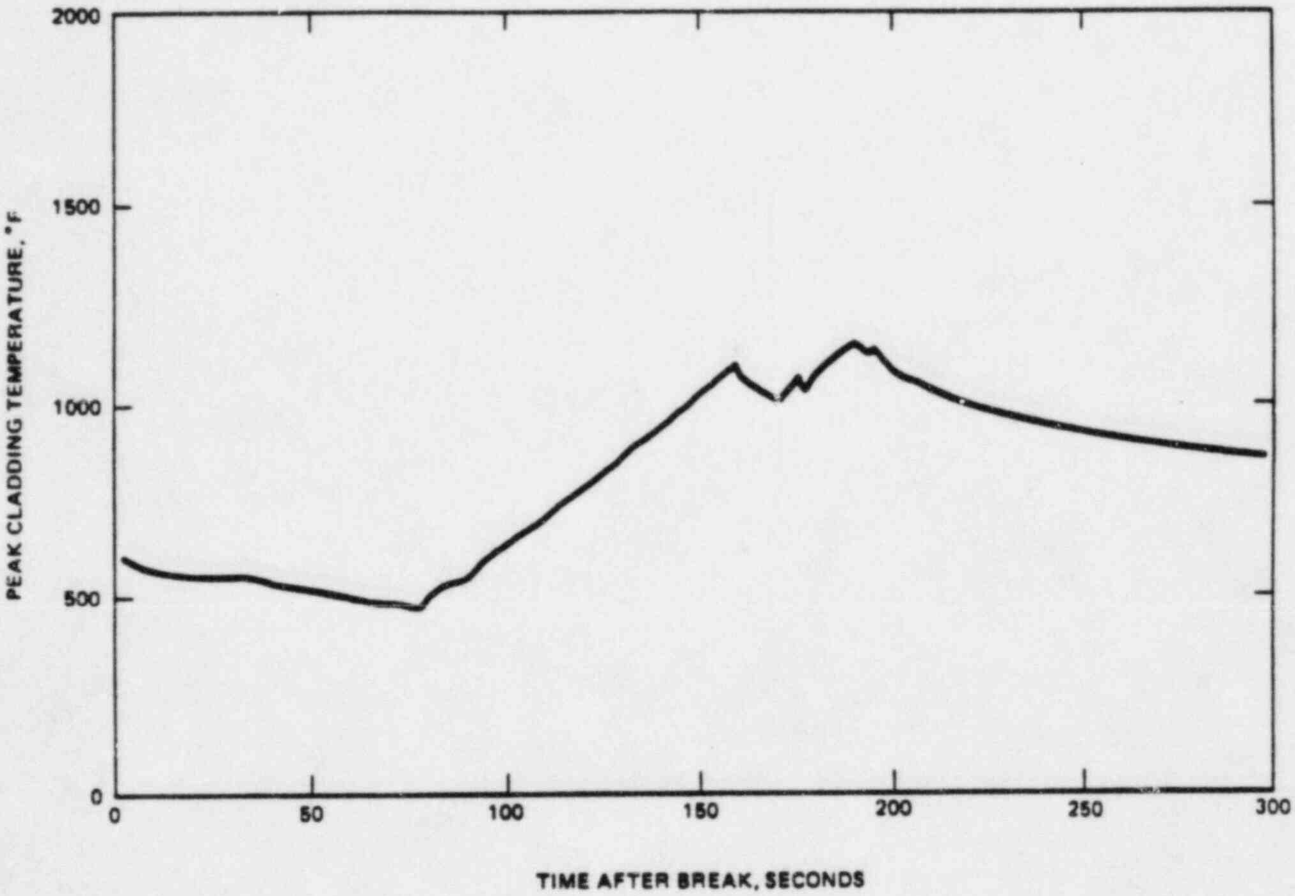
SER ITEM C-14

This replaces Figure 6.3-36.



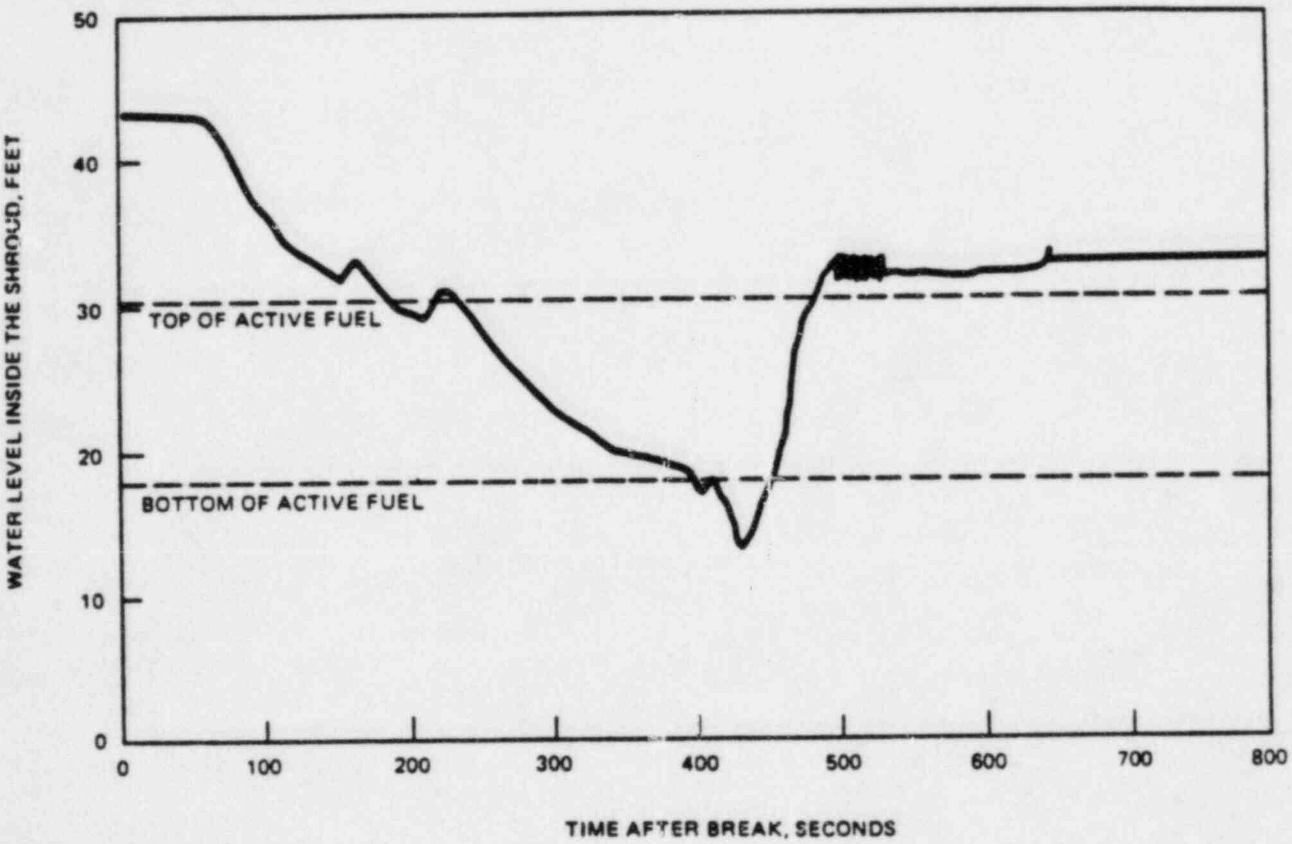
SER ITEM C-14

This replaces Figure 6.3-37.



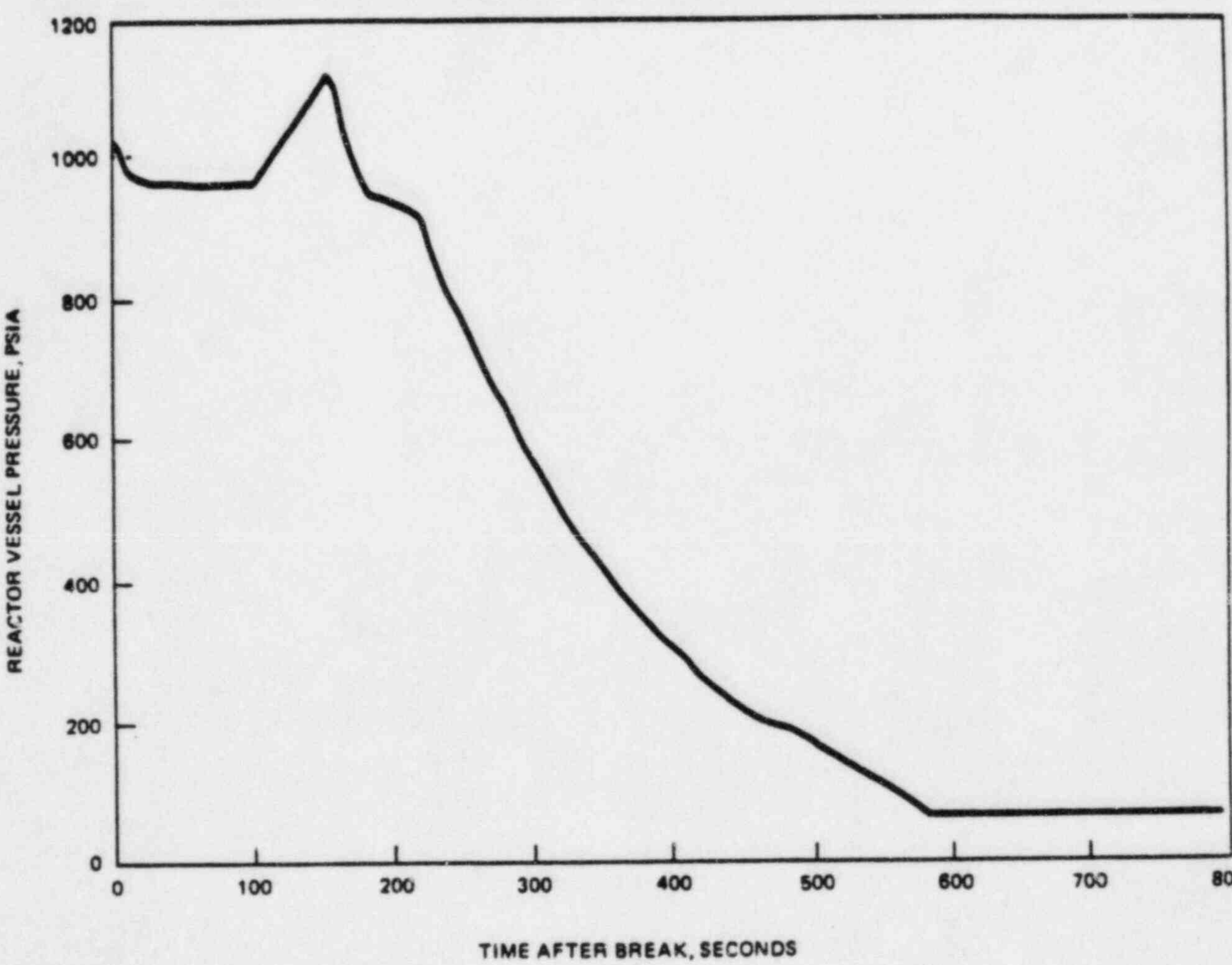
SER ITEM C-14

This replaces Figure C.3-38.



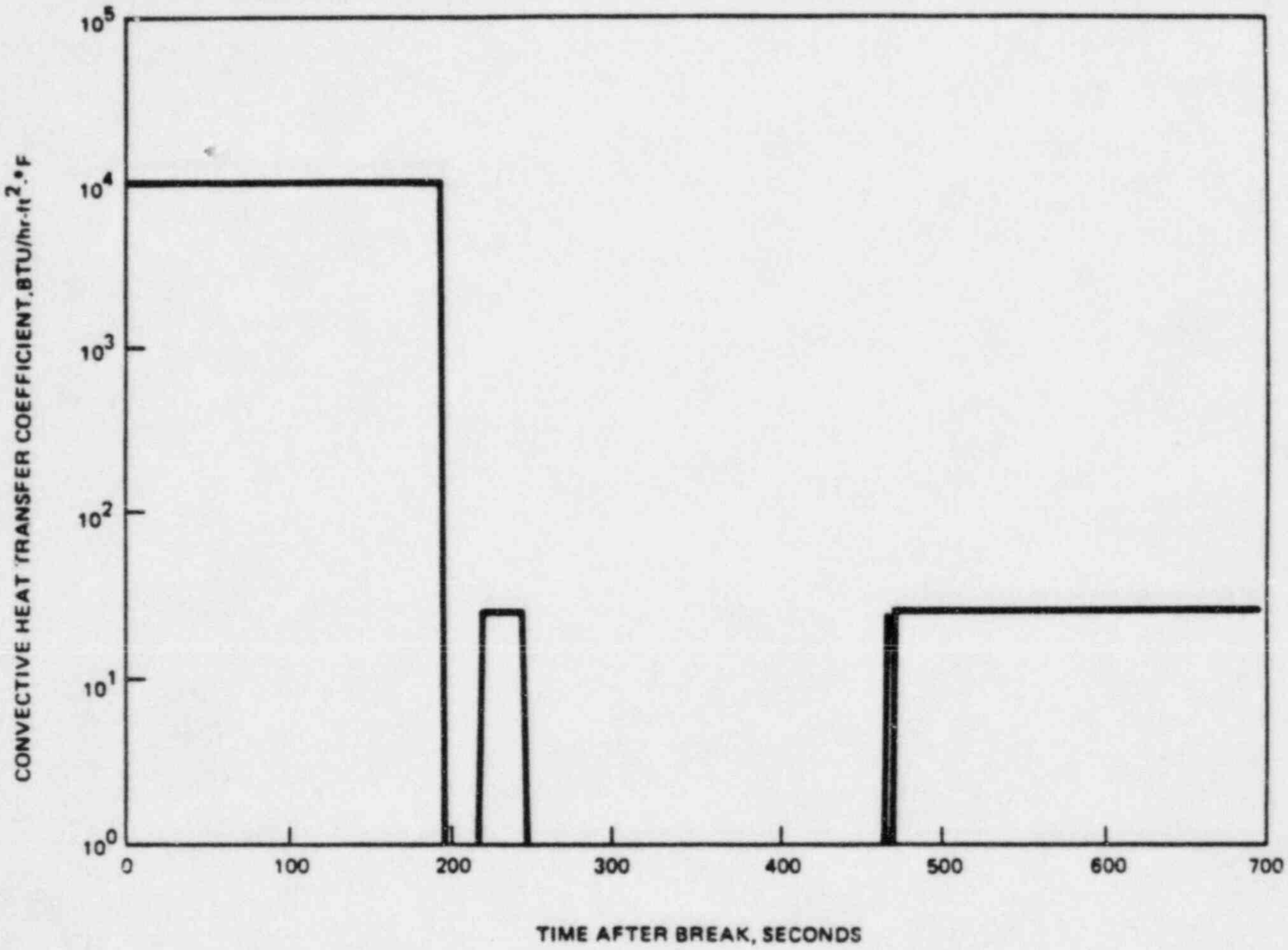
SER ITEM C-14

This replaces Figure 6.3-39.



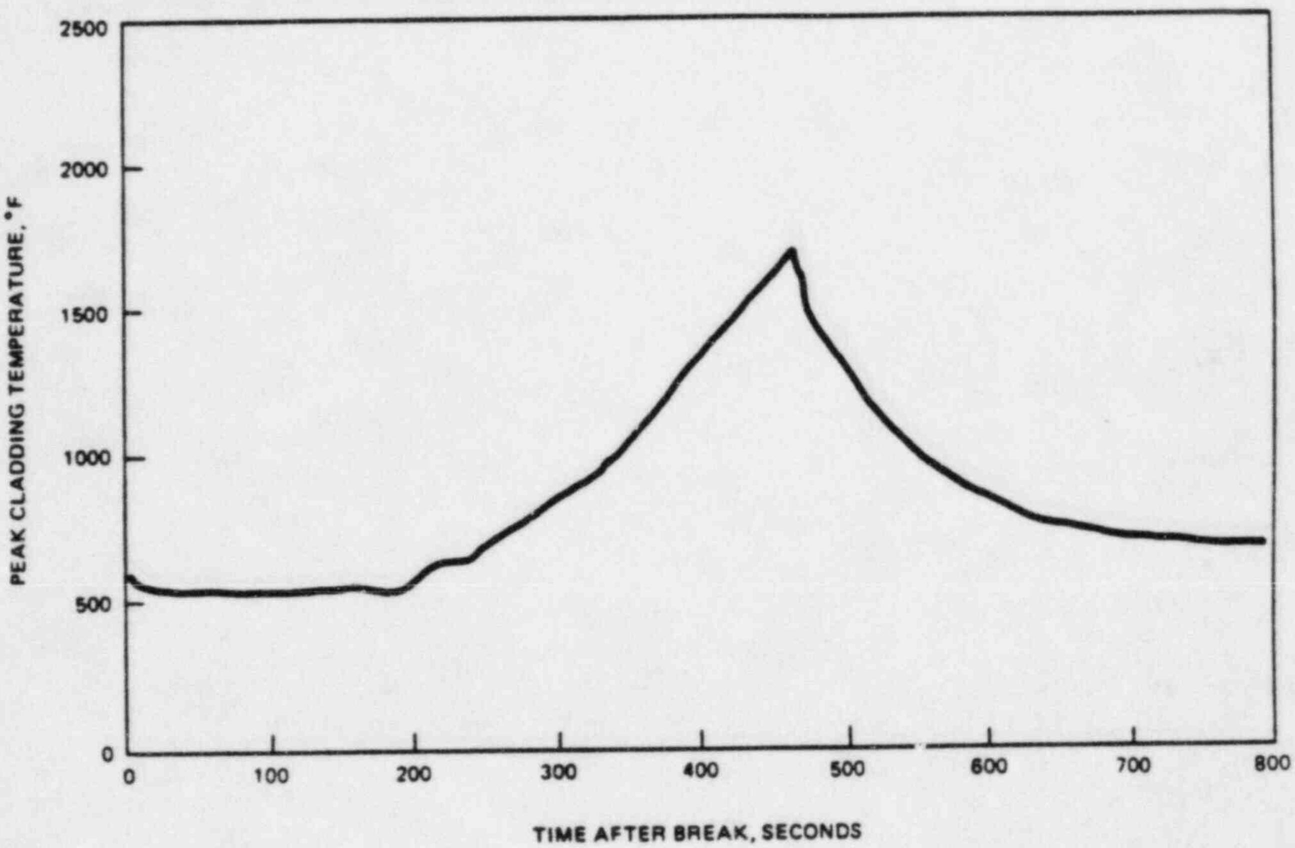
SER ITEM C-14

This replaces Figure 6.3-40.



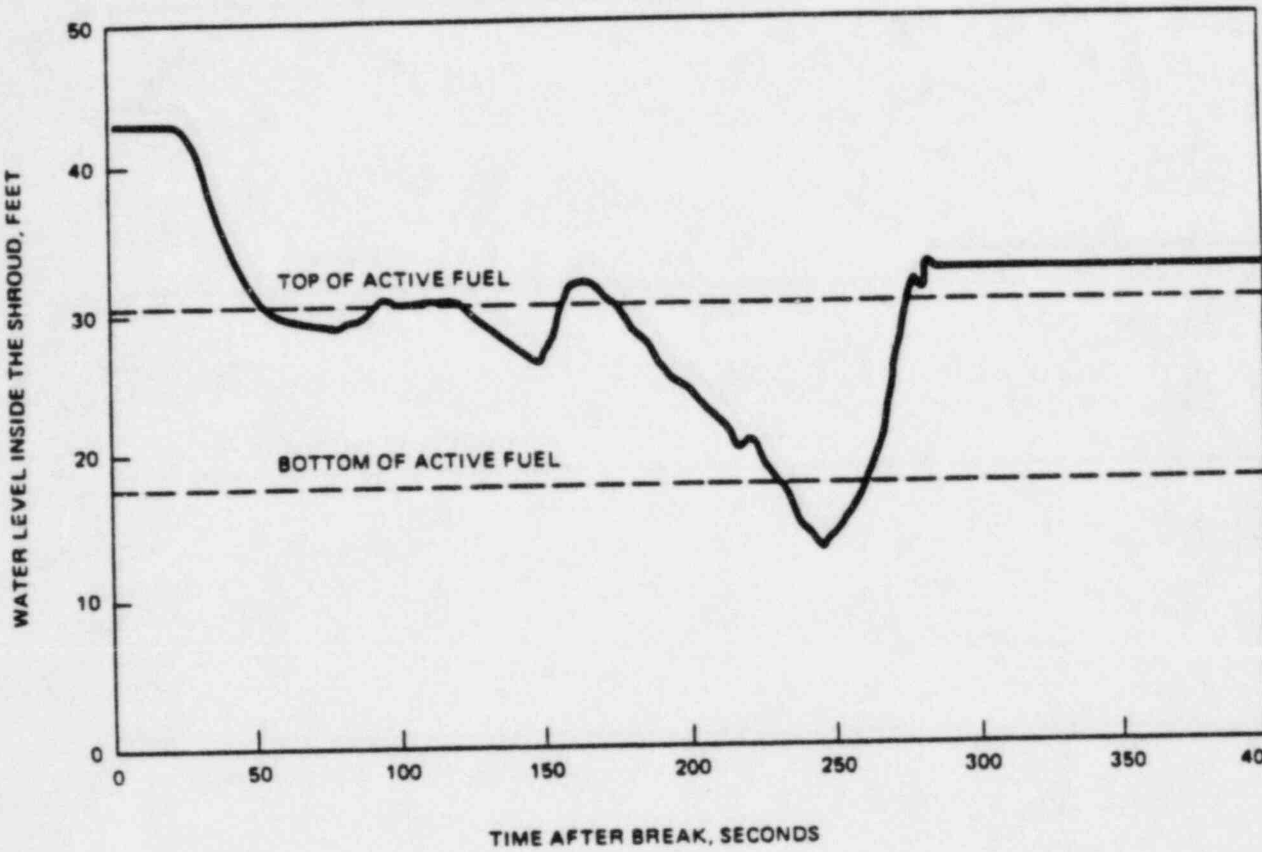
SER ITEM C-14

This replaces Figure 6.3-41.



SER ITEM C-14

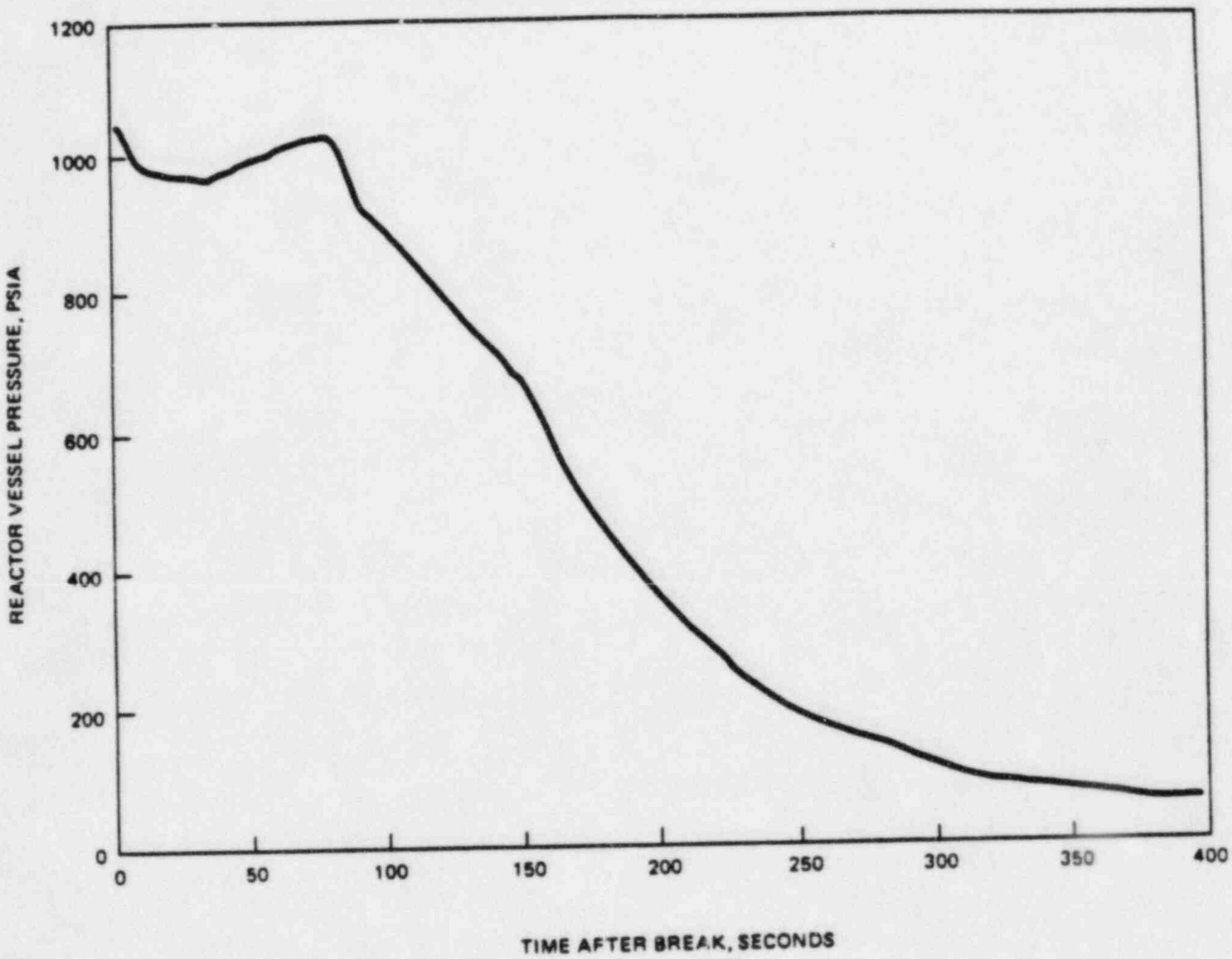
This replaces Figure 6.3-42.



Title: Water Level Inside Shroud vs. Time After Break (Small Break Model) (0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)

SER ITEM C-14

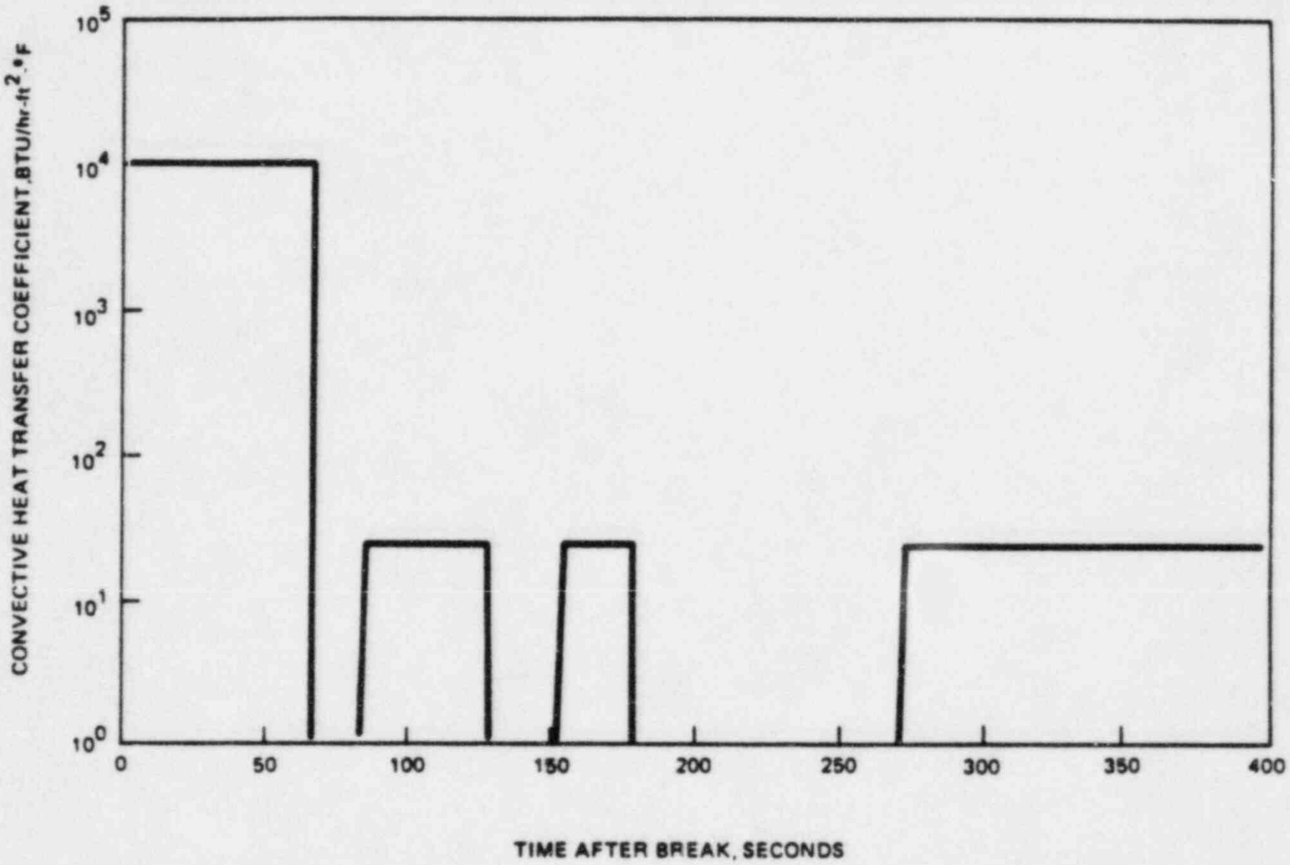
This replaces Figure 6.3-43.



SER ITEM C-14

Title: Reactor Vessel Pressure vs Time After Break (Small Break Model) (0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)

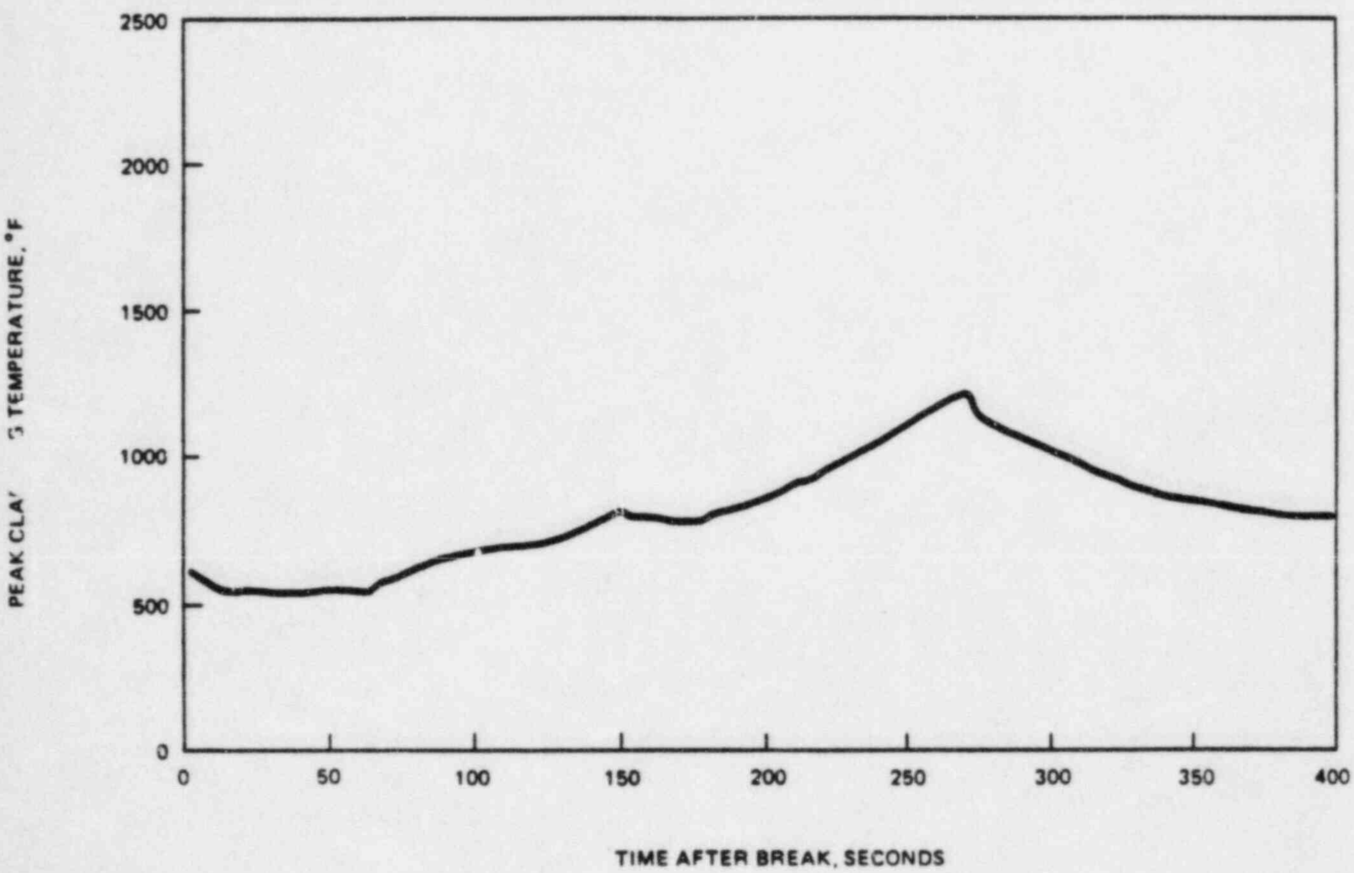
This replaces Figure 6.3-44.



SER ITEM C-14

Title: Fuel Rod Convective Heat Transfer Coefficient vs Time After Break (Small Break Model) (0.4 ft^2 Recirculation Suction Break, Failure of Channel A DC Source)

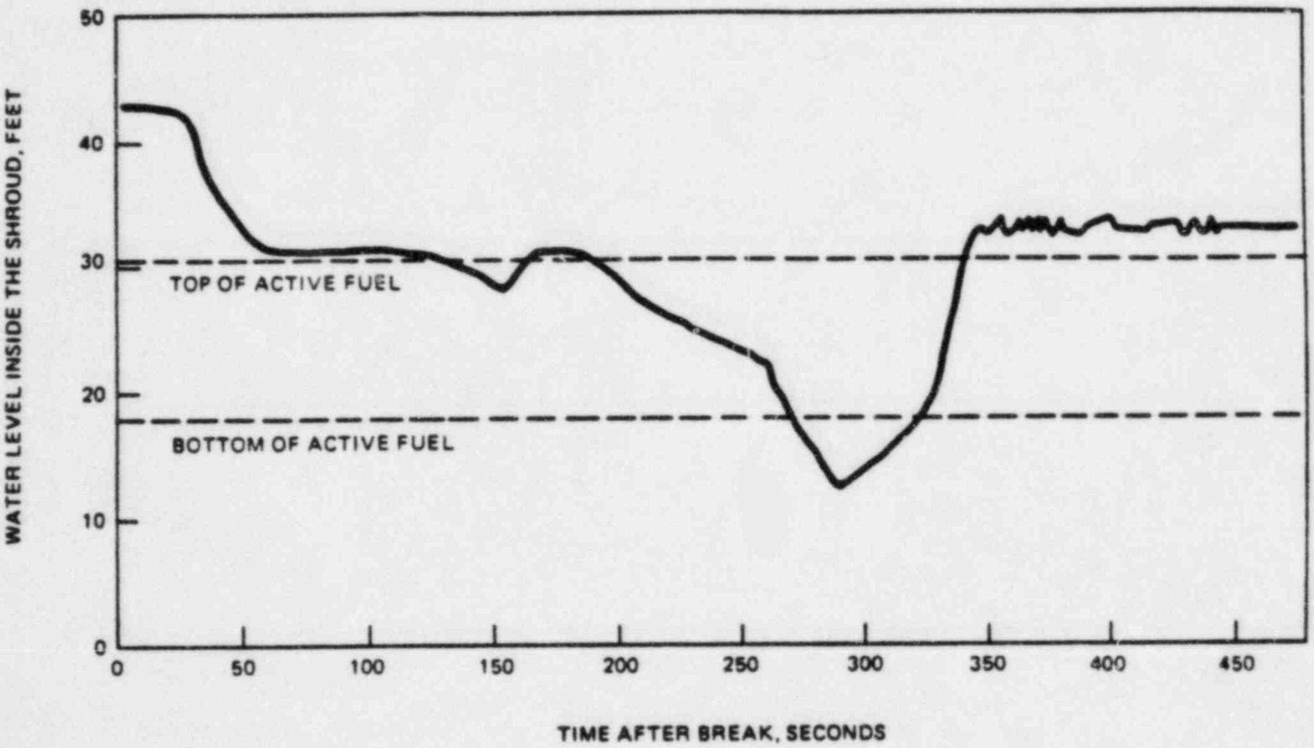
This replaces Figure 6.3-45.



SER ITEM C-14

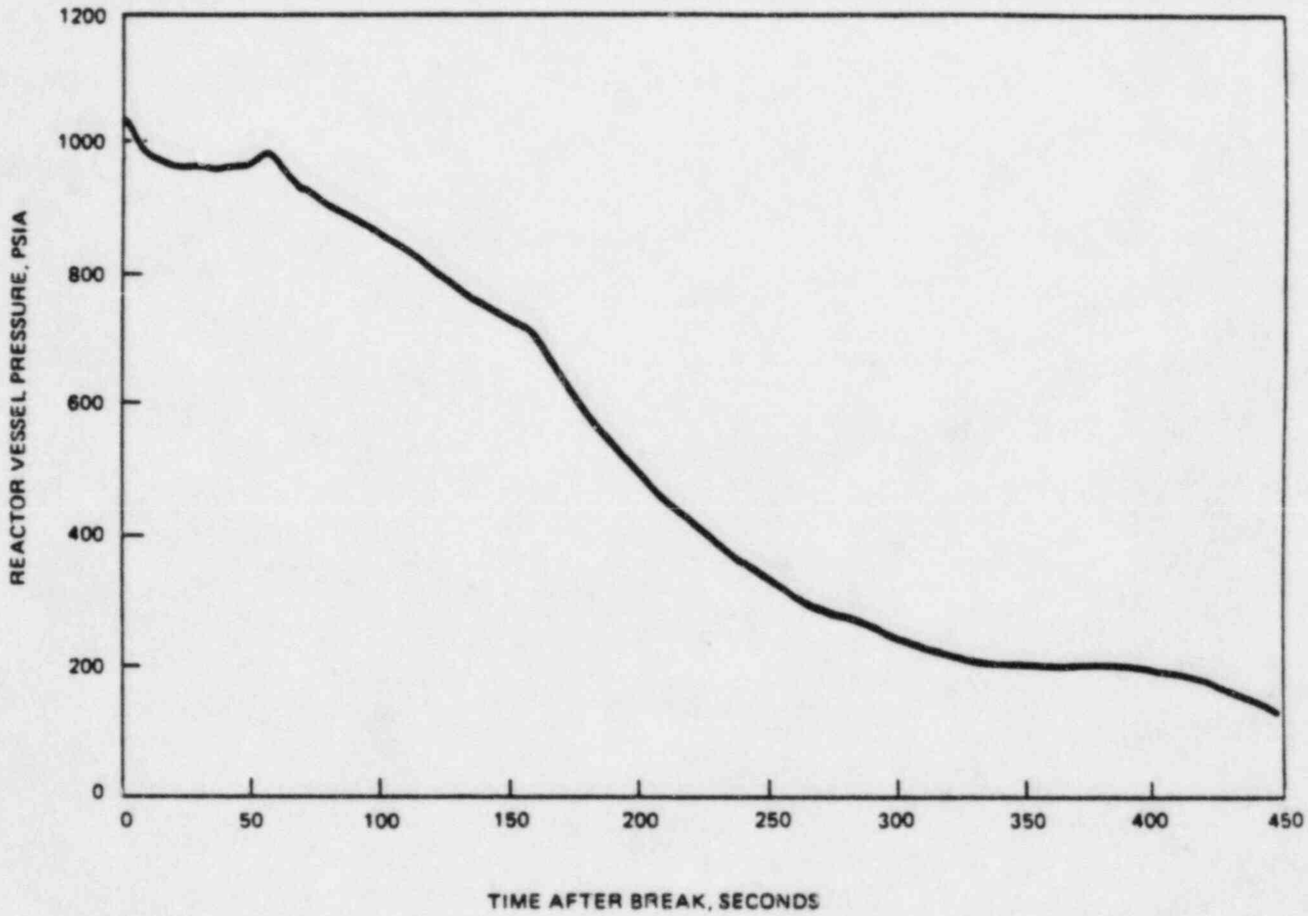
Title: Peak Cladding Temperature vs Time After Break (Small Break Model)(0.4 ft² Recirculation Suction Break, Failure of Channel A DC Source)

This replaces Figure 6.3-46.



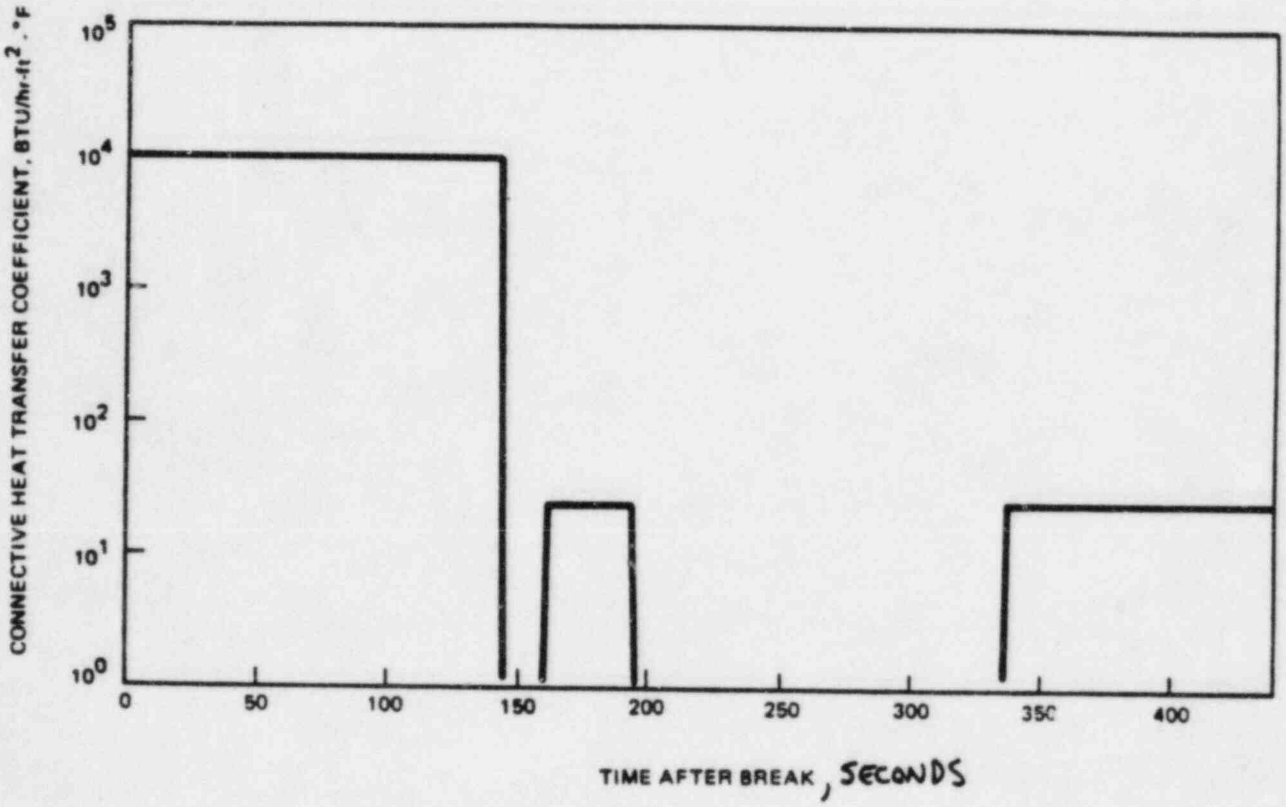
SER ITEM C-14

This replaces Figure 6.3-47.



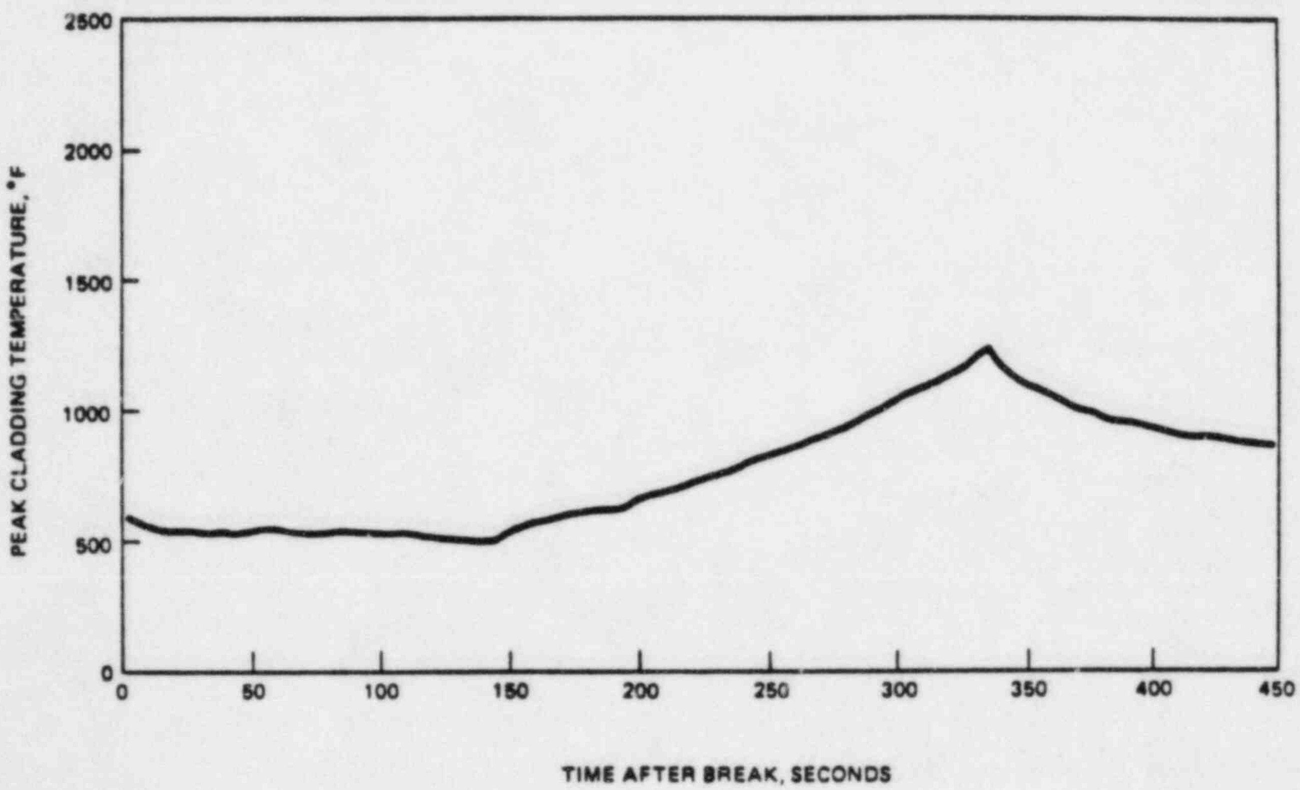
SER ITEM C-14

This replaces Figure 6.3-48.



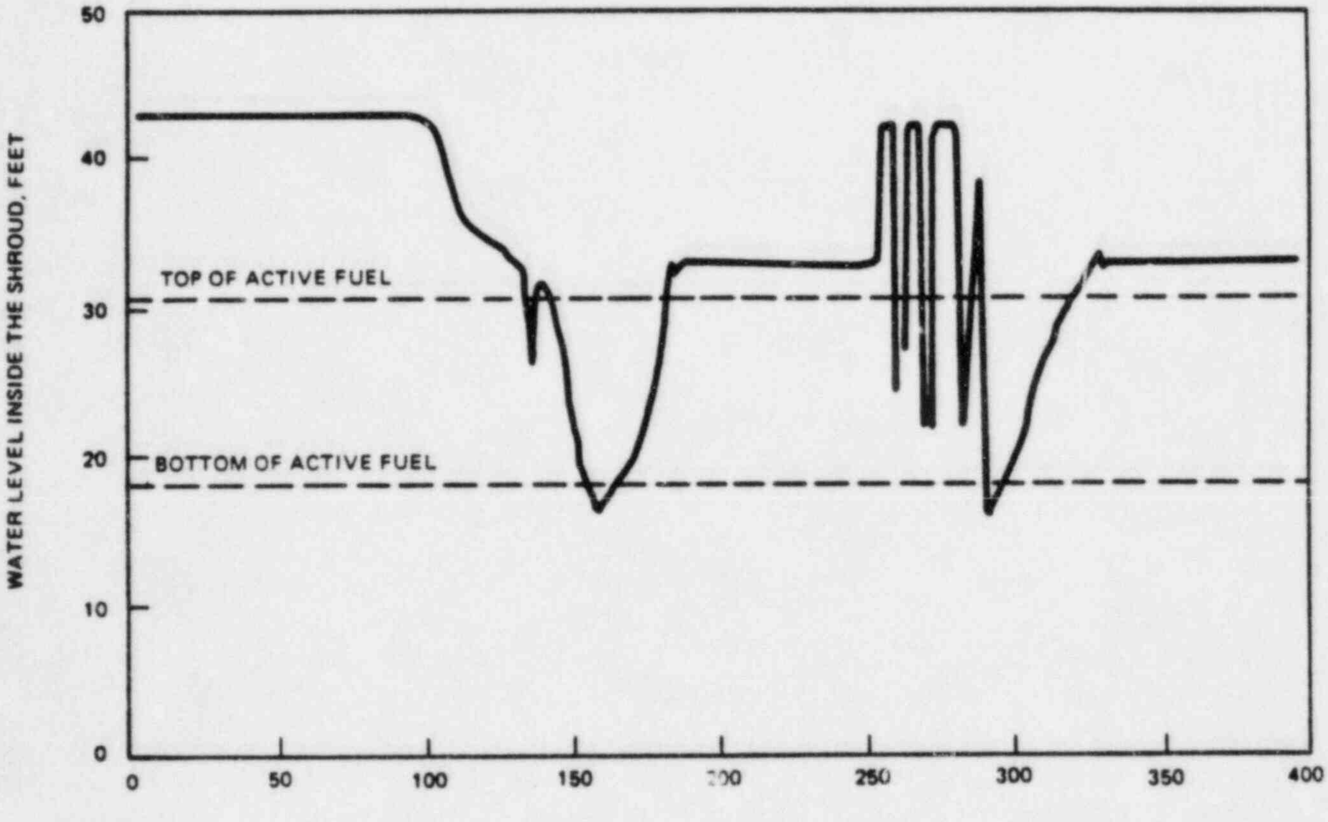
SER ITEM C-14

This replaces Figure 6.3-49.



SER ITEM C-14

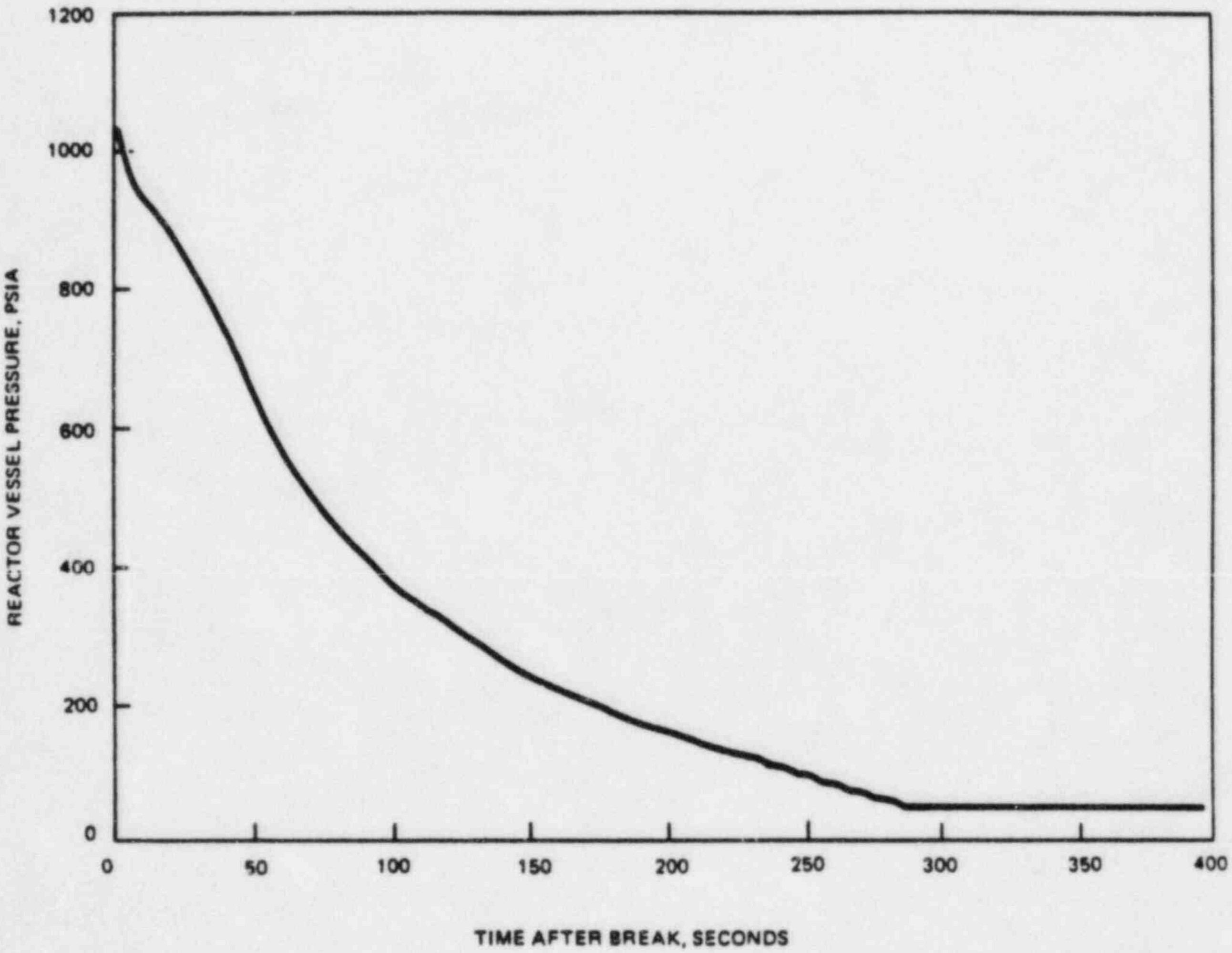
This replaces Figure 6.3-50.



TIME AFTER BREAK, SECONDS

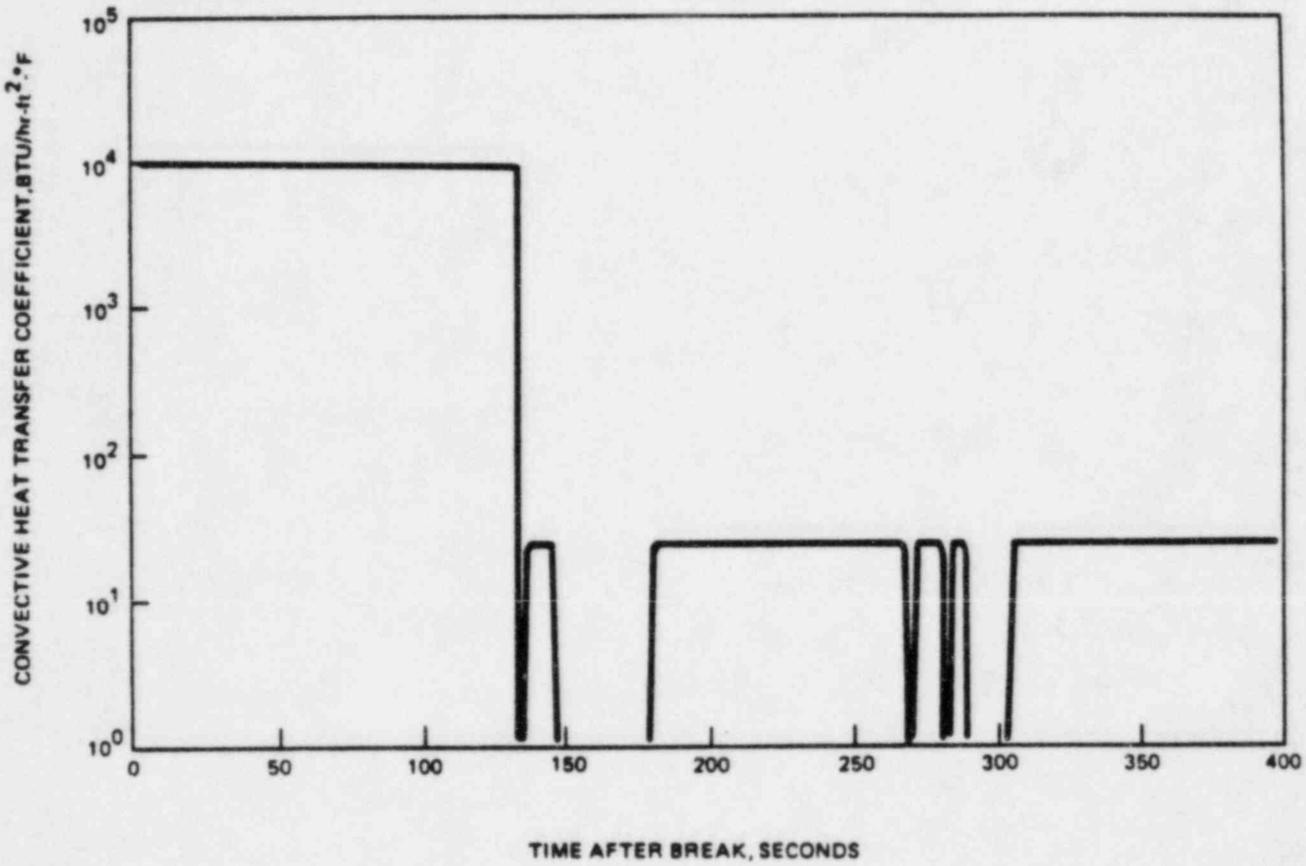
SER ITEM C-14

This replaces Figure 6.3-51.



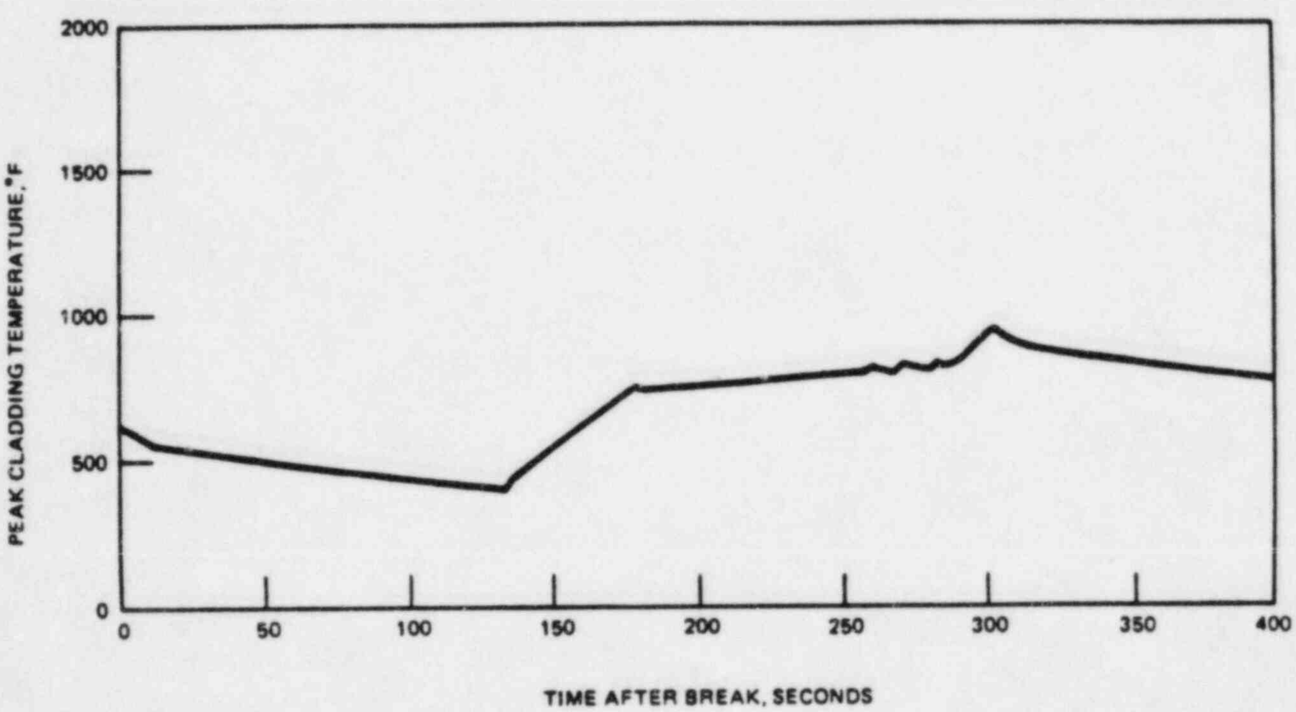
SCR ITEM C-14

This replaces Figure 6.3-52.



SER ITEM C-14

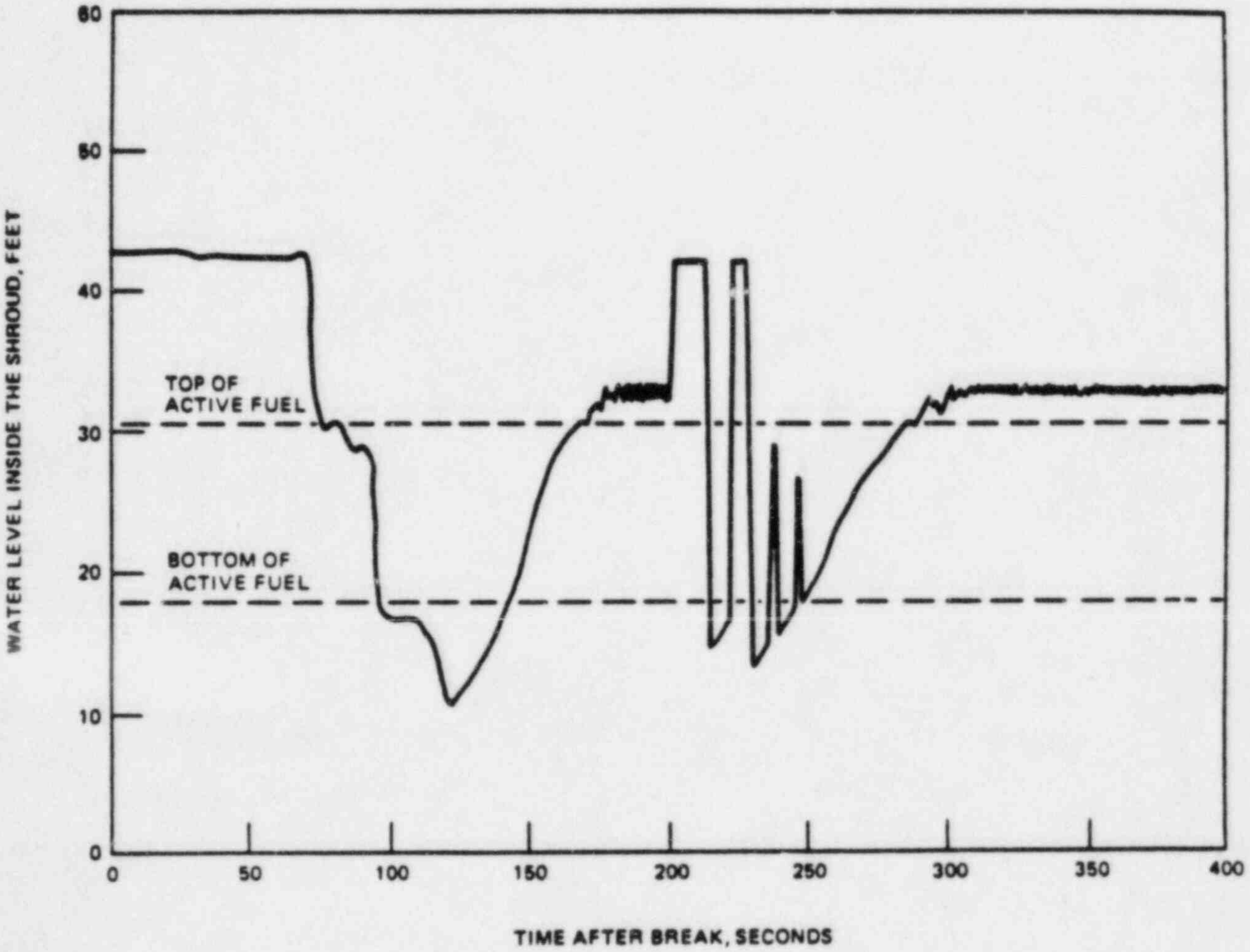
This replaces Figure 6.3-53.



SER ITEM C-14

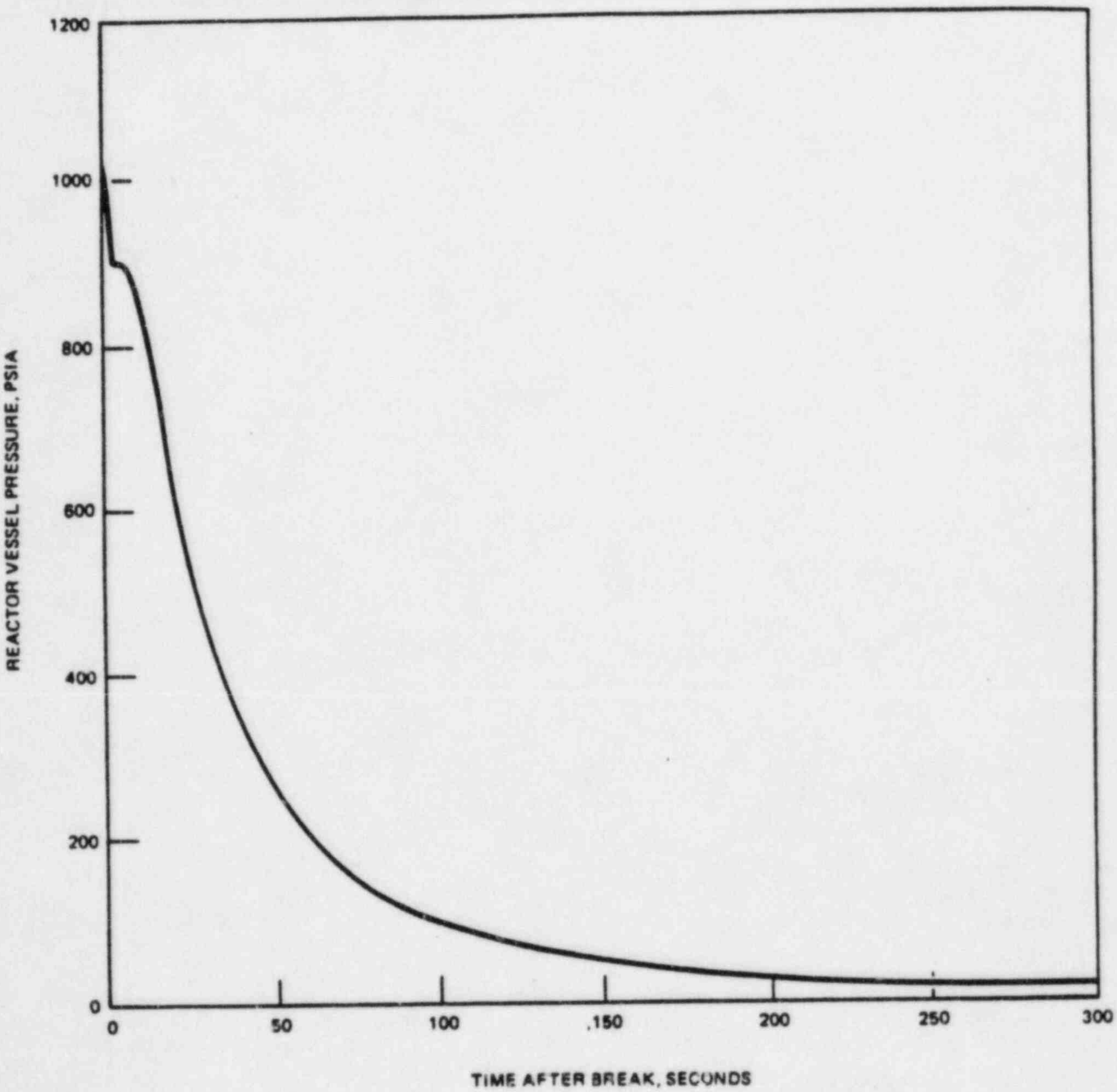
4-3-84

This replaces Figure 6.3-54.



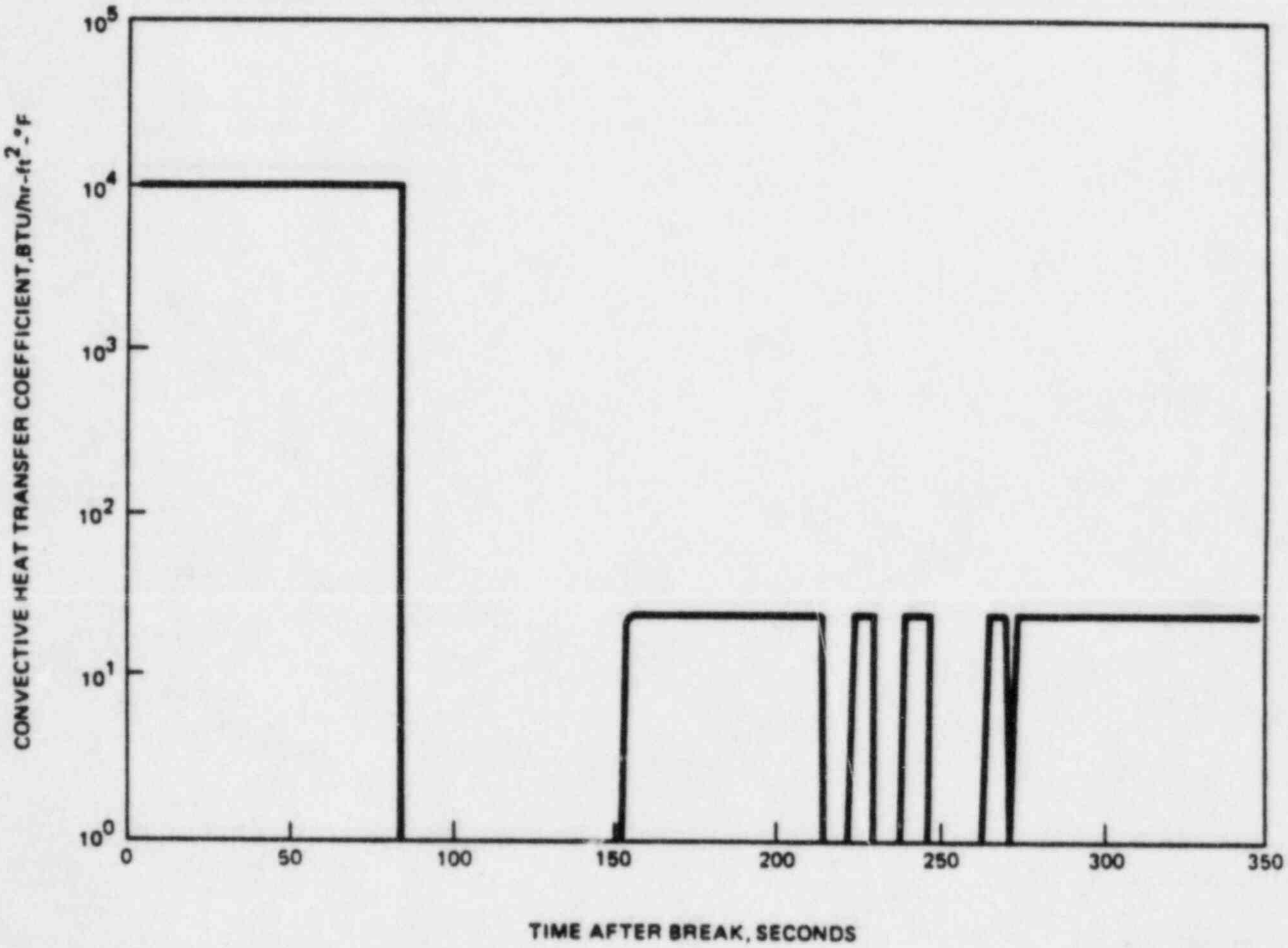
SCR ITEM C-14

This replaces Figure 6.3-55.



SER ITEM C-14

This is to replace Figure 6.3-56.

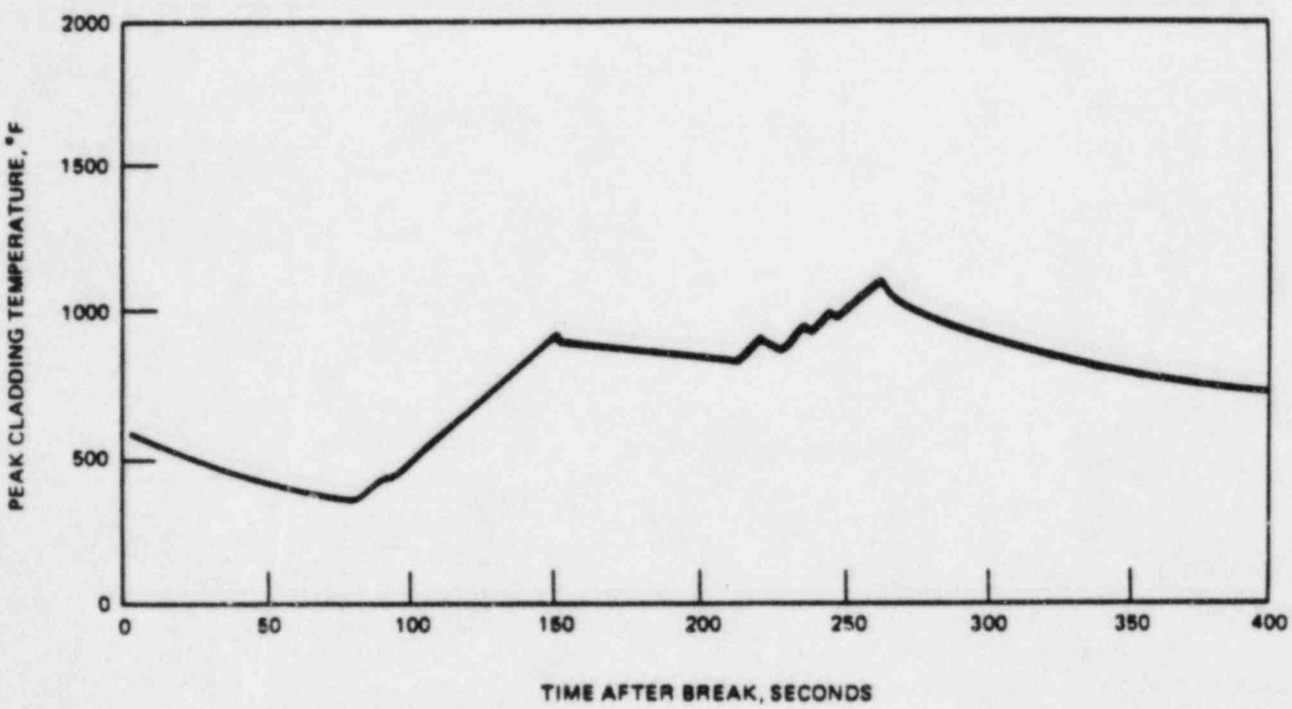


Title: Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Maximum Main Steam Line Break Inside Containment, Failure of Channel A DC Source)

SER ITEM C-14

fig - 7-57

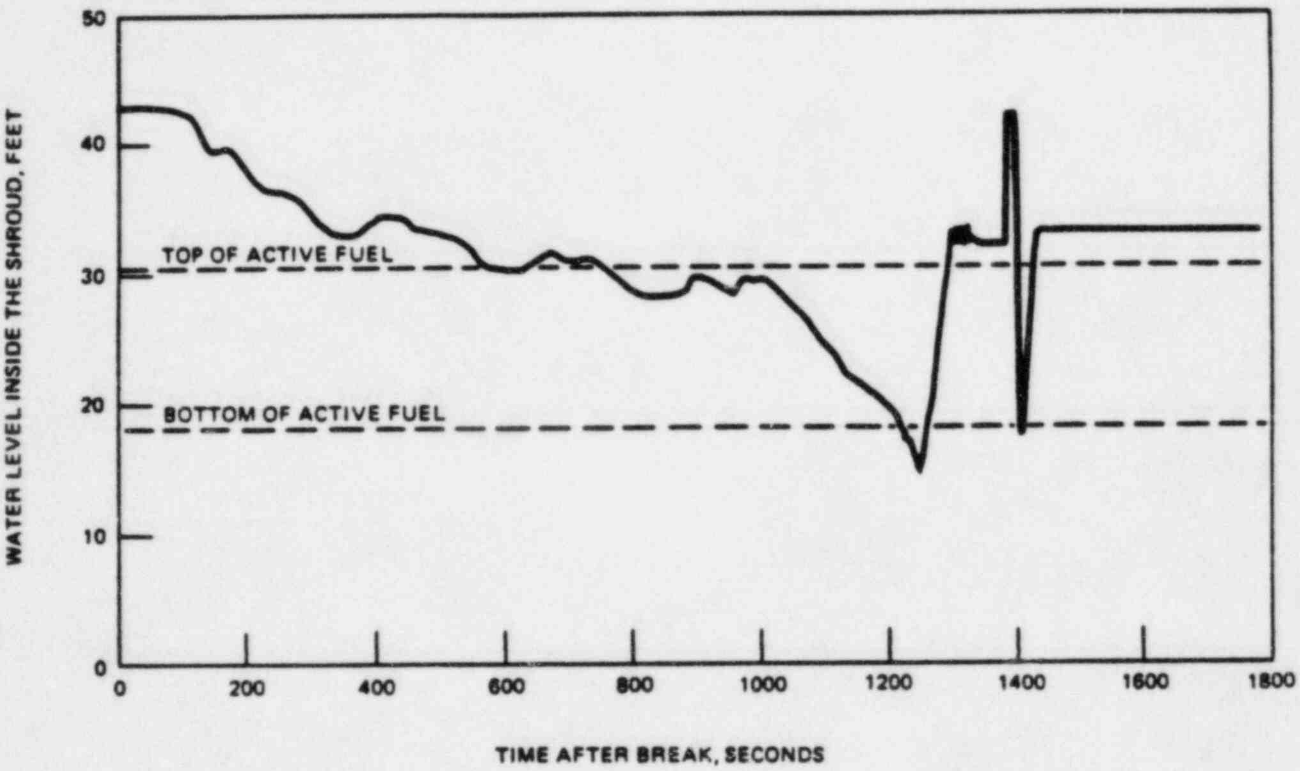
This replaces Figure 6.3-57.



Title: Peak Cladding Temperature vs. Time After Break (Maximum Main Steam Line Break Inside Containment, Failure of Channel A DC Source)

SCR ITEM C-14

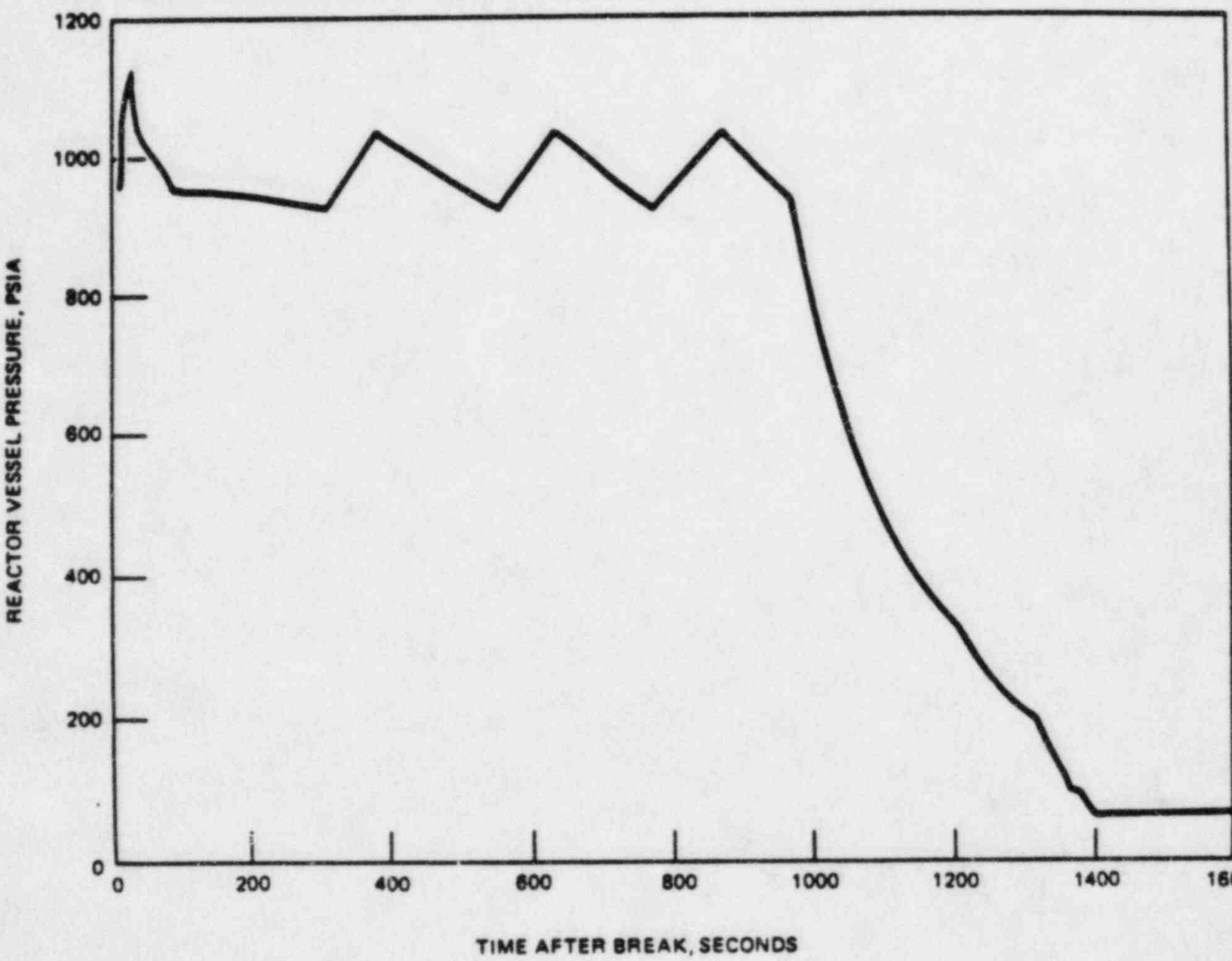
This replaces Figure 6.3-58.



Title: Water Level Inside Shroud vs Time After Break (Maximum Main Steam Line Break Outside Containment, Failure of Channel A DC Source)

SER ITEM C-14

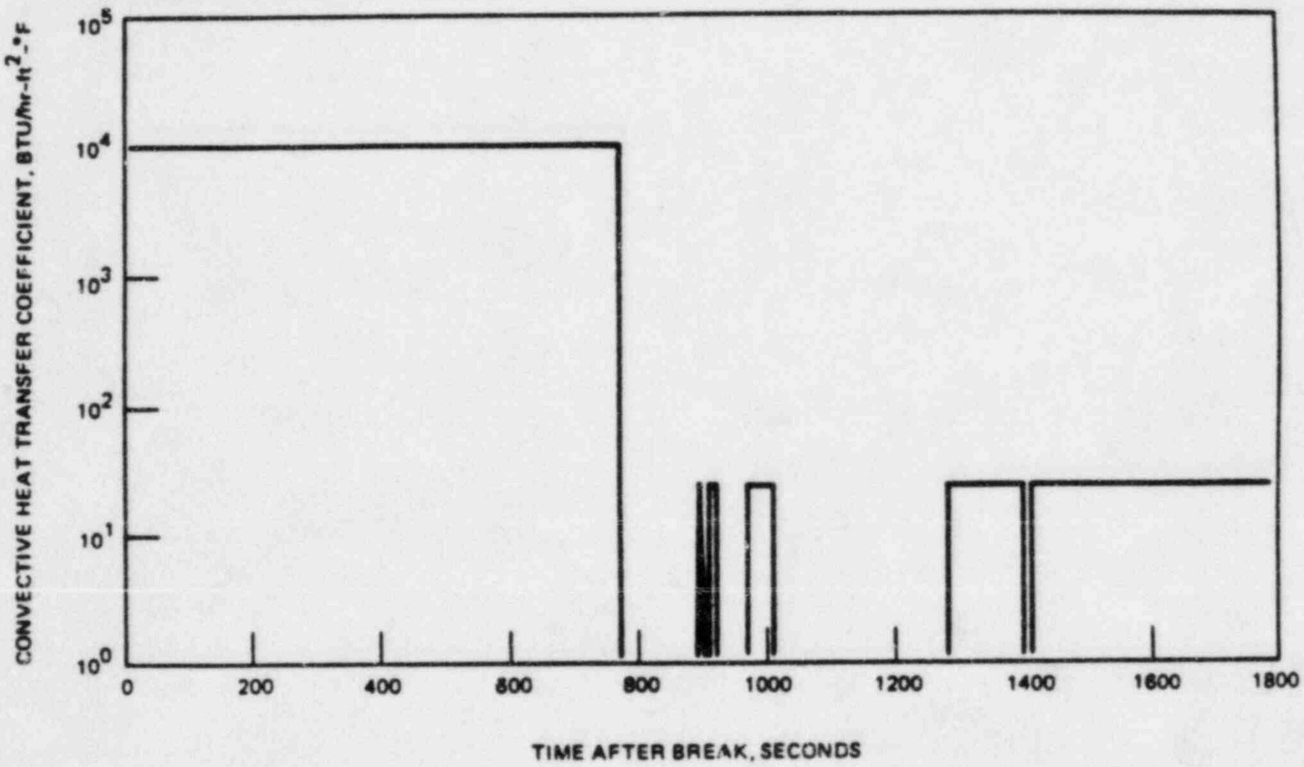
This replaces Figure 6.3-59.



Title: Reactor Vessel Pressure vs Time After Break (Maximum Main Steam Line Break Outside Containment, Failure of Channel A DC Source)

SCR ITEM C-14

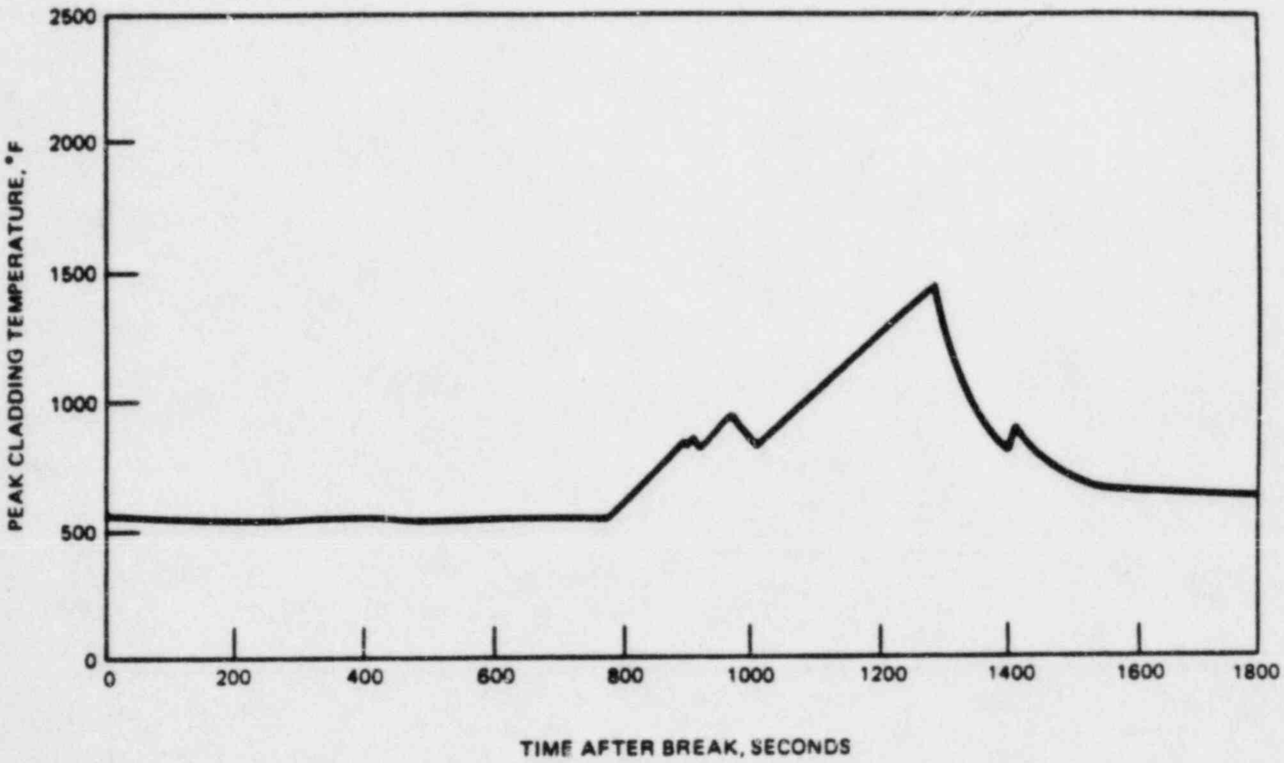
This replaces Figure 6.3-60.



Title: Fuel Rod Convective Heat Transfer Coefficient vs. Time After Break (Small Break Model) (Maximum Main Steam Line Break Outside Containment, Failure of Channel A DC Source)

SCR ITEM NO C-14

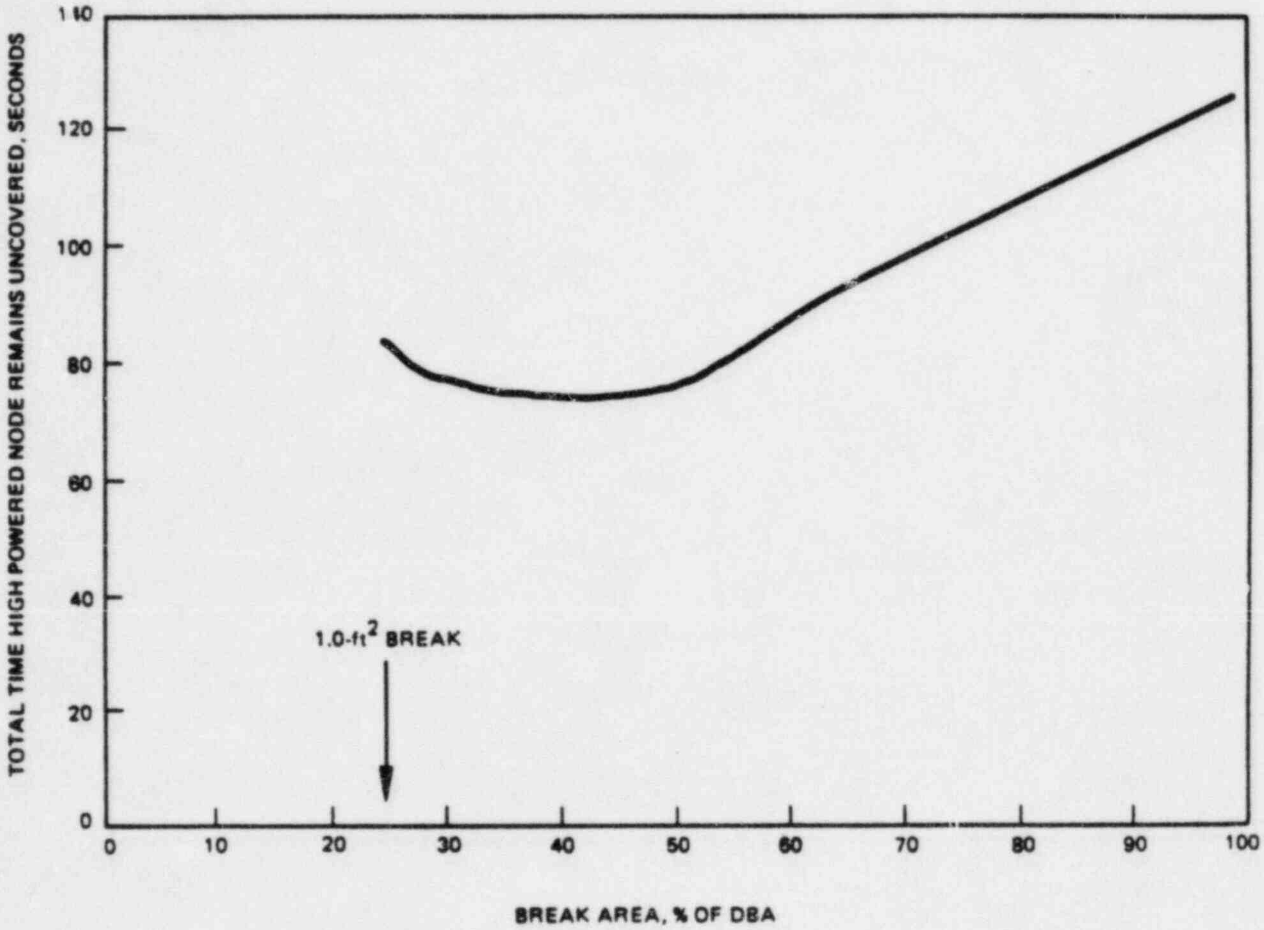
This is new Figure 6.3-61.



Title: Peak Cladding Temperature vs Time After Break (Small Break Model)(Maximum Main Steam Line Break Outside Containment, Failure of Channel A DC Source)

SER ITEM C-14

This is new Figure 6.3-62



Title: Total Time Highest Powered Node Remains Uncovered vs Break Area (Failure of Channel A DC Source)

SCR ITEM C-14

SER Confirmatory Item No. 25 (SER Section 7.6.2)

"The staff review of the elementary diagrams does not indicate that the EOC RPT transfers the pumps to low-frequency M/G sets after tripping their main power supplies. At previously reviewed BWRs (e.g., Susquehanna (NUREG-0776) and River Bend (NUREG-0989), this transfer takes place after the RPT and the pumps run at approximately one-quarter their normal speed.

"There is not sufficient information for the staff to complete its review regarding the EOC RPT. The applicant is required to submit design details showing the transfer of the recirculation pump power supply to a lower frequency motor/generator set upon EOC RPT. This is a confirmatory item."

Response:

End-of-cycle recirculation pump trip (EOC RPT) provides for the insertion of negative core reactivity to improve thermal margins for certain pressurization transients. The effectiveness of the EOC RPT arises from the rapid decrease in core flow that causes an increase in core voids immediately following the trips of the pump breakers. The early part of the transient and the core void reactivity the EOC RPT produces are not dependent on whether the final recirculation flow is determined by natural circulation or by a small power input to the recirculation pumps from a low-frequency motor/generator set. None of the GE BWR/4 plants has installed a BWR/5/6-type of low-frequency M/G set. Such installations serve no safety function in the BWR/5/6 plants, and their absence is in no way detrimental to the effectivness of the EOC RPT for the BWR/4 plants. The above SER statement, which infers the existence of a low-frequency M/G set for Susquehanna, is incorrect.

SER Confirmatory Item No. 37 (SER Section 15.9.3)

A plant-specific analysis must be provided to justify the bypass timer setting.

The staff finds the conceptual design for ADS logic modification proposed by the applicant acceptable confirmatory on completion of the above specified actions.

Response:

Plant-specific analyses have been completed to support Hope Creek's modified ADS logic design that includes a bypass of the high drywell pressure trip after a sustained low water level signal and the addition of an ADS manual inhibit switch.

The analyses considered possible design requirements for both a minimum bypass timer setting consistent with ATWS considerations and Hope Creek's RRCS logic design and for a maximum setting based on ECCS performance evaluations. The ATWS evaluation determined that there would be no Level 1 interaction expected for postulated events. The results of the ECCS evaluations are provided in response to confirmatory Item No. 14. These analyses are used to establish the technical specifications for the ADS timer and the bypass timer.

JS:vw

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