Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

July 28, 1998
NOC-AE-00151
File No.:
G20.02.01
G21.02.01
10 CFR 50.90
10 CFR 50.46

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Proposed License Amendment Associated with an
Operator Action for a Small Break Loss of Coolant Accident

Reference: Letter from T. H. Cloninger to U.S. Nuclear Regulatory Commission dated May 7, 1998 (ST-NOC-AE-00080)

The STP Nuclear Operating Company (STPNOC) proposes to amend the South Texas Project (STP) Operating License Nos. NPF-76 and NPF-80 to address an operator action to be used in the event of a small break loss of coolant accident (LOCA). The operator action and the associated revised small break LOCA analysis are reflected in a proposed revision to the South Texas Project Updated Final Safety Analysis Report (UFSAR).

STPNOC intends to replace the original Westinghouse Model E steam generators in STP Units 1 and 2 with Westinghouse Delta (Δ) 94 steam generators. The Unit 1 steam generators are scheduled to be replaced at the end of Cycle 9, currently planned for spring, 2000. The schedule for the Unit 2 replacement is the end of Cycle 9, currently planned for 2002. The proposed design changes are described in Reference 1.

The design and performance differences of the modifications associated with the replacement Δ94 steam generators, as compared with existing Model E SGs, have required re-analysis of the small break LOCA. The re-analysis was performed using the currently approved evaluation model for small break LOCAs.

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The Δ94 steam generators small break LOCA re-analysis credits operator action to reduce the setpoints of the safety-grade steam generator Power-Operated Relief Valves (PORV) at least to 1000 psig in order to provide a more rapid cooldown of the primary side by depressurizing the steam generator secondary. This operator action must be completed within 45 minutes of accident initiation to satisfy the 10 CFR 50.46 acceptance criteria.

Pursuant to concerns raised in Nuclear Regulatory Commission (NRC) Information Notice 97-78, modeling of this proposed operator action in the small break LOCA analysis is deemed an unreviewed safety question in accordance with 10 CFR 50.59. The proposed license amendment addresses the unreviewed safety question associated with this operator action.

STPNOC has reviewed the proposed amendment pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, STPNOC has determined that the proposed amendment satisfies the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an environmental assessment. The STP Nuclear Safety Review Board has reviewed and approved the proposed amendment.

The proposed amendment consists of the following attachments:
1. Affidavit,
2. Description, justification and safety evaluation,
3. No Significant Hazards Determination,
4. Information Notice 97-78 Assessment, and
5. Revised Updated Final Safety Analysis Report

In accordance with 10 CFR 50.91(b), STPNOC is notifying the State of Texas of this request for license amendment by providing a copy of this letter and its attachments.

STPNOC requests that the NRC review and approve the proposed license amendment by November 1999 to support the timely implementation of the resulting amendment and the return to power from the Unit 1 outage during which the associated steam generators will be replaced.

The small break LOCA re-analysis has demonstrated that the use of the operator action reduces the existing Peak Cladding Temperature (PCT) of 1849°F for the most limiting small break LOCA transient as reported in STPNOC letter dated July 7, 1998, (ST-NOC-AE-000189) to 1654°F. The reduction in PCT is considered a significant change in accordance with 10 CFR 50.46. This letter serves to satisfies the 30-day reporting requirements of 10 CFR 50.46(a)(3)(ii). Since a re-analysis has already been performed and the revised small break LOCA PCT is below the regulatory limit of 2200°F, a schedule for re-analysis is not provided.

If there are any questions regarding the proposed amendment, please contact Mr. M. A. McBurnett at (512) 972-7206 or me at (512) 972-8787.

T. H. Cloninger
Vice President, Nuclear
Engineering

MAL

Attachments:
1. Affidavit for Proposed Amendment
2. Description of Proposed Amendment and Safety Evaluation
3. Determination of No Significant Hazards
4. Information Notice 97-78 Assessment
5. Proposed Changes to Technical Specifications

cc:

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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ATTACHMENT 1

AFFIDAVIT FOR PROPOSED AMENDMENT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
STP Nuclear Operating Company	)	Docket Nos. 50-498
	)	50-499
South Texas Project Units 1 & 2	)	

AFFIDAVIT

I, T. H. Cloninger, being duly sworn, hereby depose and say that I am Vice President, Nuclear Engineering, of STP Nuclear Operating Company; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached proposed amendment to the STP Operating License Nos. NPF-76 and NPF-80; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

T. H. Cloninger
Vice President, Nuclear Engineering

STATE OF TEXAS	)
	)
COUNTY OF MATAGORDA	)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 28th day of July, 1998.

LINDA RITTENBERRY
Notary Public, State of Texas
My Commission Expires
OCT. 9, 2001

Linda Rittenberry
Notary Public in and for the
State of Texas

ATTACHMENT 2

DESCRIPTION OF THE PROPOSED CHANGES

RELATED TO

REPLACEMENT STEAM GENERATOR
SMALL BREAK LOSS OF COOLANT ACCIDENT
REQUIRED OPERATOR ACTIONS

1. BACKGROUND

The South Texas Project (STP) will replace the original Westinghouse Model E steam generators (SGs) with Westinghouse Model Delta (Δ) 94 steam generators. Unit 1 SG replacement is scheduled to occur at the end of Cycle 9, currently planned for the spring of the year 2000. Unit 2 replacement is scheduled to occur at the end of Cycle 9, currently planned for the year 2002. Thus, STP will be operating with two different SG models for a period of time.

The new Δ94 SGs to be installed in each unit have a lower flow resistance. As a part of the SG modification, additional bypass flow will be introduced by increasing the upper head (UH) flow nozzle flow area to address the increased core flow in the Δ94 SGs. As a result, the STP design with the Δ94 SGs will change from a T_{HOT} UH design to a T_{COLD} UH design. The proposed design changes are described in STPNOC letter dated May 7, 1998 (NOC-AE-00080).

The design and performance differences of the modifications associated with the replacement Δ 94 SGs, as compared with existing Model E SGs, have required re-analysis of the small break loss of coolant design basis accident (LOCA). The re-analysis was performed using the currently approved evaluation model for small break LOCAs.

The Δ94 SGs small break LOCA re-analysis credits operator action to reduce the setpoints of the safety-grade SG Power-Operated Relief Valves (PORV) at least to 1000 psig in order to provide a more rapid cooldown of the primary side by depressurizing the SG secondary. The Emergency Operating Procedures will direct the operator to reduce the setpoint to between 990 and 1000 psig. This operator action must be completed within 45 minutes of accident initiation to satisfy the 10 CFR 50.46 acceptance criteria. In fact, the re-analysis has demonstrated that the use of the operator action reduces the existing Peak Cladding Temperature (PCT) of 1849°F for the most limiting small break LOCA transient as reported in STPNOC letter dated July 7, 1998, (ST-NOC-AE-00189) to 1654°F. This reduction in PCT is considered a significant change in accordance with 10 CFR 50.46.

Pursuant to concerns raised in NRC Information Notice 97-78, modeling of this proposed operator action in the small break LOCA analysis is deemed an unreviewed safety question in accordance with 10 CFR 50.59. The proposed license amendment addresses the unreviewed safety question associated with this operator action.

The Δ94 SG re-analysis also assumes reduced main feedwater temperature and the automatic operation of the safety-grade SG PORV. The reduced main feedwater temperature is consistent with a future STPNOC license amendment as stated in STPNOC letter dated May 7, 1998 (ST-NOC-AE-00159). The automatic operation of the SG PORVs is consistent with the license amendment proposed by STPNOC in its letter dated August 18, 1997 (ST-HL-AE-05689). Although not a specific part of this proposed license amendment, STPNOC requests the NRC to include the automatic operation of the safety-grade SG PORVs and the reduced main feedwater temperature as an integral part of its review.

2. PROPOSED CHANGES

STPNOC proposes to revise the STP Updated Final Safety Analysis Report (UFSAR), as shown by the draft in Attachment 5, to reflect the revised small break LOCA analysis for the Δ94 SGs. The proposed license amendment herein addresses the operator action to reduce the SG PORV setpoint as addressed by the revised small break LOCA analysis for the Δ94 SGs.

The proposed license amendment reflects the steam generator model rather than the STP unit in which the Δ94 SGs and T_{COLD} UH design are installed. Therefore, the proposed license amendment is applicable to either unit and no future license amendment would be necessary to support the Δ94 SGs and T_{COLD} UH modifications to Unit 2.

3. SAFETY EVALUATION

3.1 Introduction

The current STP small break LOCA analysis is documented in Chapter 15.6.5 of the UFSAR, and reflects the Model E SG and T_{HOT} UH design. A re-analysis has been performed to model the Δ94 SGs and the conversion to the T_{COLD} UH design. Other initial conditions and analysis assumptions are discussed below.

The results of the small break LOCA analysis for the Δ94 SGs demonstrate conformance with the 10 CFR 50.46 acceptance criteria (Ref. 1).

3.2 Method of Analysis

The NOTRUMP small break LOCA evaluation model was used to re-analyze the small break LOCA for the Δ94 SGs. The NOTRUMP evaluation model (Refs. 2 and 3) consists of the NOTRUMP code (Ref. 4) and the LOCTA-IV code (Ref. 5). The NOTRUMP computer code is a state-of-the-art, one-dimensional general network code used to perform the system hydraulic calculations throughout the transient. Clad thermal analyses were performed with the LOCTA-IV code which uses as input the reactor coolant system (RCS) pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations.
The STP small break LOCA analysis is fundamentally the same as other small break LOCA analyses performed by Westinghouse, but some variation from the typical Westinghouse small

break LOCA analysis application were incorporated owing to the unique features of the STP design and small break LOCA behavior. These unique modeling features include:

- Use of an explicit loop (4-loop) version of the NOTRUMP model,
- Modeling of the safety-grade SG PORVs for a small break LOCA, and
- Credit for operator action to reduce the SG PORV setpoint

The STP units have a number of other unique design features which were considered in the application of the evaluation model. In addition to a 14 foot core, the STP Emergency Core Cooling System (ECCS) includes accumulators and independent, dedicated (unheadered) pumped ECCS in three of the four reactor coolant loops. The remaining loop (Pressurizer loop - loop D) has no accumulator or pumped ECCS. STP units also have no automatic safety-grade charging flow resulting from a safety injection (SI) signal. The STP units feature safety-grade PORVs for each of the four steam generators. The SG PORVs will lift, providing secondary steam relief when the pre-set pressure is reached. These unique design features, in conjunction with various limiting single failure (LSF) assumptions can lead to large asymmetries between reactor coolant loops. To adequately and accurately model the effects of these asymmetries, it is necessary to use an explicit loop (4-loop) version of the NOTRUMP model.

Use of an explicit loop model allowed appropriate modeling capability to capture the potential loop-to-loop asymmetries in ECCS, auxiliary feedwater (AFW) delivery, and PORVs (depending on the LSF assumed and the assumed location of the faulted loop). With the additional noding required to model the four explicit loops, all the typical features of the NOTRUMP evaluation model were employed in the Δ94 SGs analysis, including:

- Loop seal restriction,
- SI in the Broken Loop (except for LSF scenarios in which no SI is available to the broken loop),
- COSI condensation model, and
- Those models and assumptions as set forth in the approved version of the NOTRUMP topical report.

The safety-grade SG PORVs were modeled in the same manner as other atmospheric relief valves (i.e., MSSVs) as described in the approved version of the NOTRUMP topical report. To ensure conservatism in the modeling, uncertainties were calculated for the STP-specific instrumentation, and these uncertainties were added to the opening pressure based on the current setpoint modeled in the calculations. Opening of the safety-grade SG PORVs is assumed to be automatic, and no operator action is required to actuate this system. Operator action to facilitate the automatic mitigation capability of the safety-grade SG PORVs is described in Section 3.3. Potential failure of one or more PORVs due to a partial loss of emergency power and ultimate discharge of the associated nitrogen accumulators has been included in the LSF studies (see Section 3.3.3).

3.3 Design Basis

3.3.1 Initial Condition Analysis Assumptions

- The design basis small break LOCA is assumed to result from an instantaneous rupture of a pipe in the RCS pressure boundary consistent with 10 CFR 50, Appendix K. The limiting small break LOCA event occurs with the plant operating at 102% Licensed Core Power in accordance with existing licensing basis assumptions.

The analysis includes the calculation of a spectrum of three break sizes to determine the limiting small break LOCA break size. The analysis was performed with a reactor vessel UH temperature equal to the RCS cold leg temperature consistent with the modified STP Unit 1 design. The analyses evaluated a range of operating temperatures (582.3°F < T_{AVG} < 593.0°F) plus RCS temperature uncertainties of 6° F, and a range of main feedwater (MFW) temperatures (390.0°F < T_{MFW} < 440.0°F). The reduced main feedwater temperature is consistent with a future STPNOC license amendment as stated in STPNOC letter dated May 7, 1998 (ST-NOC-AE-00159). In addition, the results bound both Zirc-4 fuel and fuel with ZIRLO cladding, instrument tubes, thimbles, and grids.

Other important input parameters and initial conditions used in the Δ94 SGs small break LOCA analysis are listed in Table 1.

3.3.2 Operator Action Assumptions

Operator actions assumed in the design basis accident analysis are identified in UFSAR Sections 15.6.5.3.2.3 and Q211.52 of the NRC Questions and Responses.

The revised small break LOCA analysis for the Δ94 SGs assumes operator action at or prior to 45 minutes after the initiation of the accident to provide a more rapid cooldown of the primary side by depressurizing the steam generator secondary at least to 1000 psig. The Emergency Operating Procedures (EOPs) will direct the operator to reduce pressure, if possible, using the steam dumps first, then the safety-grade SG PORVs, if steam dumps are unavailable. The small break LOCA analysis takes credit for the safety-grade SG PORVs only, since the Steam Dumps are not safety-grade. The operator is instructed to lower the SG PORV setpoint, an operation achievable from the plant control room. Consistent with the intent of the EOPs, this operator action facilitates the automatic mitigation capability of the SG PORVs, and does not initiate the mitigating safety function.
NRC Information Notice (IN) 97-78 provides the guidelines for review of licensee analyses of operator actions. These guidelines and the STP assessment for this proposed operator action are presented in Attachment 3.

3.3.3 System Failure Assumptions

LOCA analyses are performed under Single Failure Criterion in accordance with 10 CFR 50, Appendix K I.D.1, and 10 CFR 50, Appendix A, General Design Criterion 35.

Due to the asymmetries presented by the unique design features of the STP design, multiple combinations of limiting single failures were considered. These included common mode failures (emergency diesel generator failure) that could lead to various failures of pumped ECCS, AFW, and SG PORVs. Sensitivity studies of these multiple scenarios were performed, using the explicit loop NOTRUMP model (See Section 3.2), to establish the limiting scenario in terms of small break LOCA PCT results. The limiting scenario was found to be a postulated actuation signal failure due to the assumed loss of emergency power to Train A. This leads to a loss of pumped ECCS and the Motor-Driven AFW in Loop A, and the loss of the Turbine-Driven AFW in Loop D, with the break in loop B. The Loop A and D SG PORVs are also assumed inoperable with the loss of Train A emergency power because of the ultimate discharge of the associated nitrogen accumulators. The failure of the SG PORVs to open was included in the limiting scenario. A failure of the SG PORV to close was concluded not to represent a degradation to the mitigation capability for small break LOCA. The results of this limiting scenario are discussed in Section 3.4.

3.4 Results

3.4.1 Transient Progression

The Δ94 SGs analysis follows a typical small break LOCA transient progression, accelerated by the automatic opening of the safety-grade SG PORVs and operator action to provide a more rapid cooldown of the primary side by depressurizing the steam generator secondary.

The limiting case for the Δ94 SGs small break LOCA analysis is the 2-inch equivalent diameter break, with low RCS temperature, and low MFW temperature (See Section 3.3.1). Table 2 (time sequence of events) and the following transient plots for the limiting case demonstrate the general trends of the transient.

Pressurizer pressure	Figure 1
Core mixture level	Figure 2
Peak clad temperature	Figure 3

3.4.2 Results Discussion

The most limiting small break LOCA was demonstrated to be a 2-inch equivalent diameter break, with low RCS and low MFW temperatures within the range of operation conditions described in Section 3.3.1. Table 3 provides the results for the limiting 2-inch low RCS temperature, low

MFW case. The revised PCT for the most limiting small break LOCA transient is calculated to be 1654° F, which remains below the 10 CFR 50.46 acceptance criteria of 2200° F.

4. Implementation

The modifications associated with the Model Δ94 RSGs and the T_{COLD} UH modification are scheduled to be completed by the Spring of the year 2000 for Unit 1 and the year 2002 for Unit 2. The proposed license amendment is applicable to both STP units. STPNOC requests that the NRC review and approve the proposed license amendment by November 1999 to support the timely implementation of the resulting amendment and the return to power from the Unit 1 outage.

5. References

1. Nuclear Regulatory Commission, Code of Federal Regulations - Energy, Title 10, Chapter 1, Part 50, Section 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors, January 4, 1974, Amended September 16, 1988.

2. Lee, N.; Rupprecht, S. D.; Tauche, W. D.; Schwarz, W. R.; "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10054-P-A, August 1985.

3. Thompson, C. M., et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Revision 1, 1995.

4. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code", WCAP-10079-P-A, August 1985.

5. Bordelon, F. M., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary), 1974.

Table 1

Small Break LOCA Input and Initial Conditions

Core Power[1], MWt	3,800	
Total Peaking Factor	2.70	
Axial Peaking Factor	1.62	
Reactor Coolant Pressure, psia	2,300[2]	
Thermal Design Flow (gpm/loop)	98,000	
Steam Generator Tube Plugging Level, %	10	
AFW Flow, gpm[3]	Loop A	0
	Loop B	500
	Loop C	500
	Loop D	0
AFW Temperature, °F	212	
MSSV setpressures, psig[4]	Valve 1	1285
	Valve 2	1295
	Valve 3	1305
	Valve 4	1315
	Valve 5	1325
Steam Generator PORV setpoint (automatic), psig[5]	1225	
Upper Head Temperature	T_{COLD}	
SI in the Broken Loop	Modeled	
NOTRUMP Loop Modeling	4 loops (no lumping)	
Safety-Grade SG PORVs	Credited	
Operator Action (secondary cooldown)	Within 45 minutes	
Break Size (inches)	1.5, 2, 3	

1 Two percent is added to this power to account for calorimetric error.
2 This value bounds the pressure of 2250 psia plus 46 psi uncertainty assumed in the UFSAR.
3 Corresponds to the LSF scenario leading to the limiting PCT (See Section 3.3.3).
4 Value is adjusted in accident analysis to include 3% accumulation, and 3% uncertainty for each valve. Data represents each of five valves on each of four steam generators.
5 Value is adjusted in accident analysis to include 4.58% uncertainty. While there is no "accumulation" for this PORV, an additional 4.9% was added to allow for full flow.

Table 2

**Δ94 SGs SMALL BREAK
TIME SEQUENCE OF EVENTS**

	Time (seconds)
Event	2 in Lo-Lo†
Start (Accident Initiation)	0.0
Reactor Trip Signal, sec	52.9
Top of Core Uncovered, sec	2025
Accumulator Injection Begins, sec	N/A
Peak Clad Temp Occurs, sec	2657.6
Top of Core Covered, sec	3291

† Lo-Lo - Low loop temperature (582.3°F - 6°F uncertainty); Low MFW temperature (390°F).

Table 3

Δ94 SGs SMALL BREAK ANALYSIS RESULTS

Event	2 in. Lo-Lo†
Peak clad temperature, °F	1654
Elevation, ft.	13.25
Max. local ZR/H20 reaction, %	0.88
Elevation, ft.	13.25
Total ZR/H_2O reaction	<1.0
Hot rod burst time, sec.	N/A
Elevation, ft.	N/A

† Lo-Lo - Low loop temperature (582.3°F - 6°F uncertainty); Low MFW temperature (390°F).

FIGURE 1
Small Break LOCA Pressurizer Pressure
2" break RCS Temperature of 582.3° F
MFW Temperature of 390° F

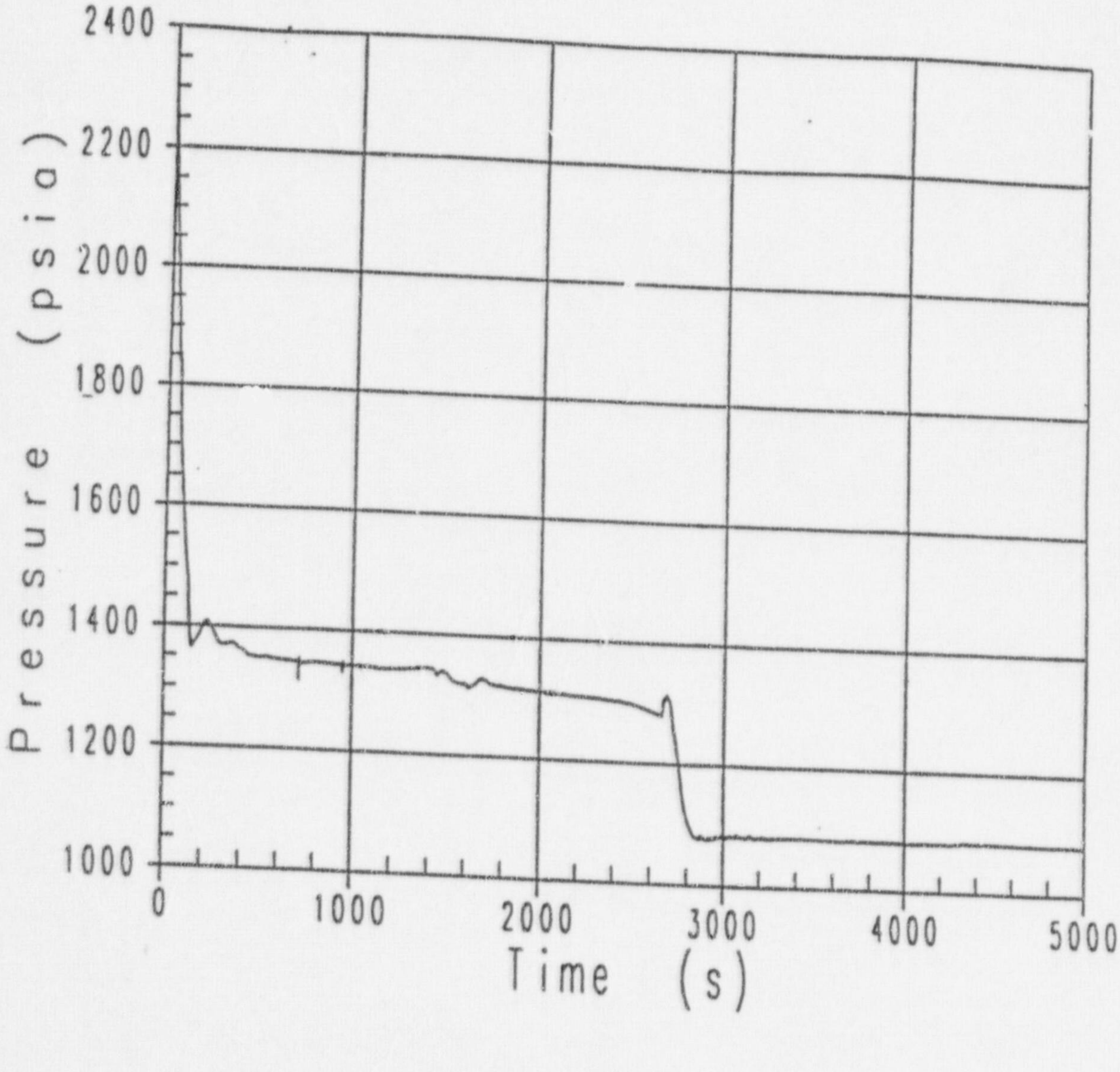


FIGURE 2
Small Break LOCA Core Mixture Level
2" break RCS Temperature of 582.3° F
MFW Temperature of 390° F

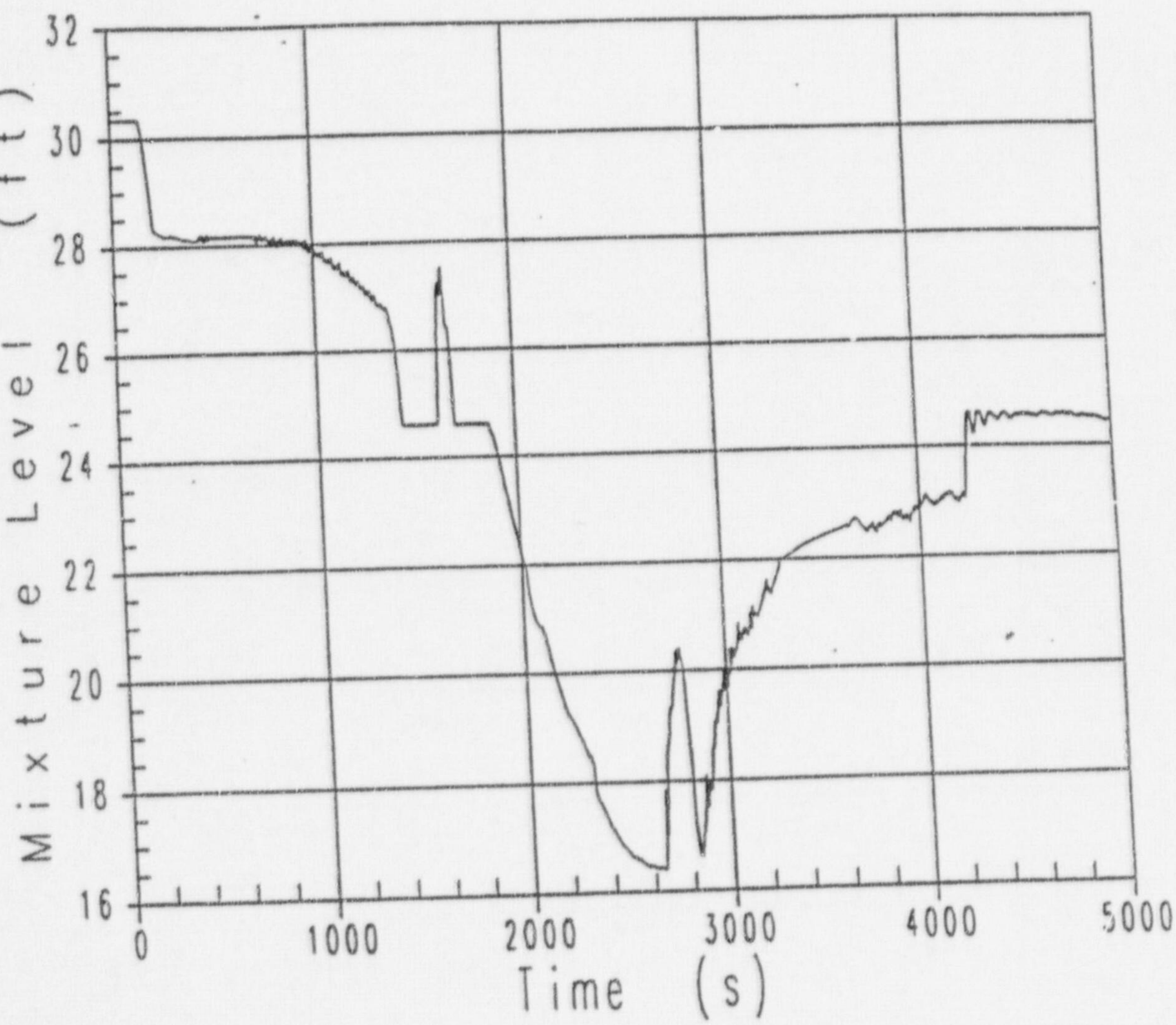
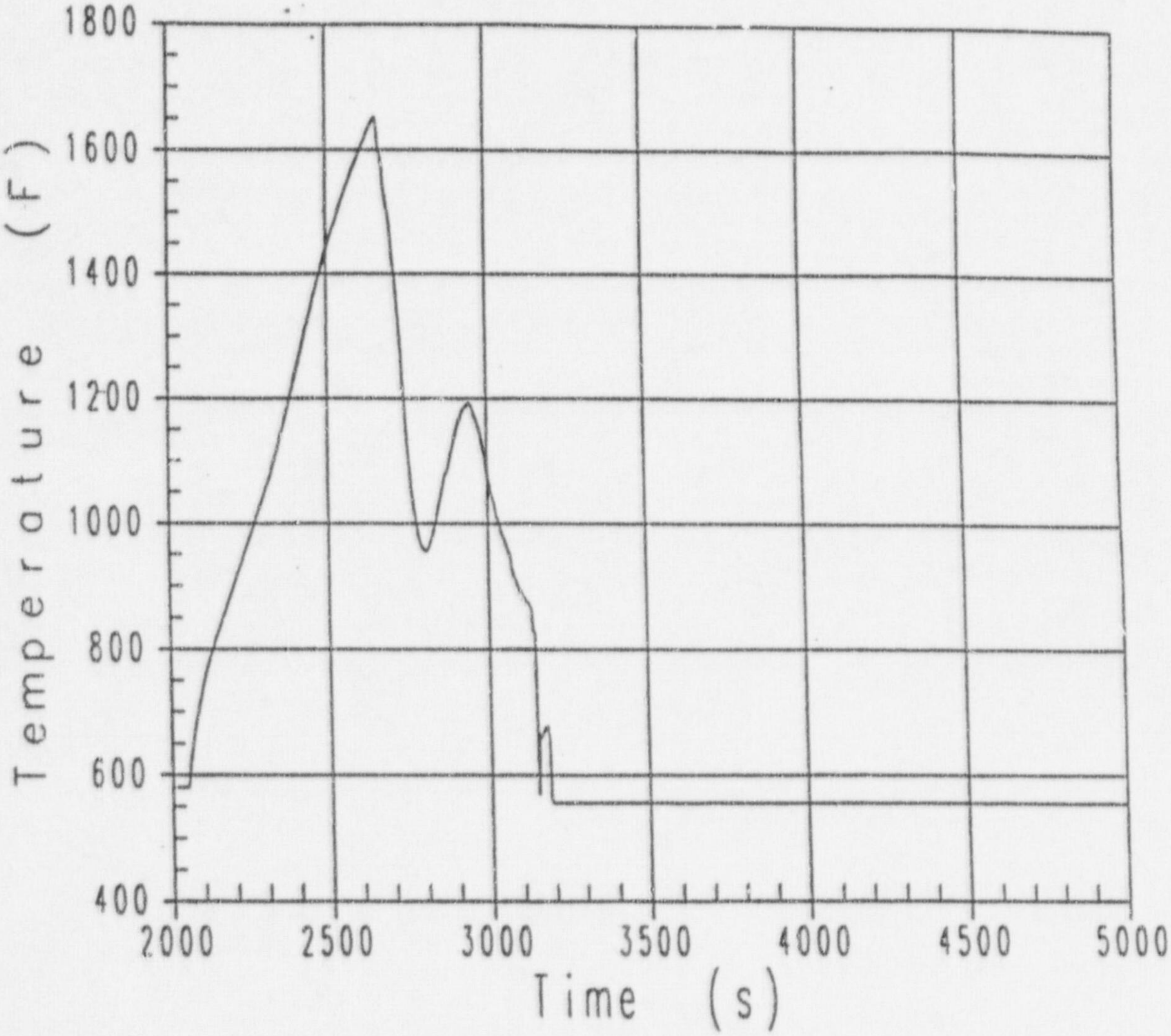


FIGURE 3
Small Break LOCA Peak Clad Temperature
2" break RCS Temperature of 582.3° F
MFW Temperature of 390° F



ATTACHMENT 3

IN 97-78 ASSESSMENT

ATTACHMENT 3
IN 97-78 ASSESSMENT

IN Guideline 1

The specific operator actions required.

STP Assessment of IN Guideline 1

The operator action credited in the small break LOCA analysis is to lower the SG PORV setpoints within 45 minutes after accident initiation. The purpose of the action is to provide a more rapid cooldown of the primary side by depressurizing the secondary side at least to 1000 psig during a small break LOCA using the steam dumps first, then the SG PORVs, if steam dumps are unavailable. This operator action is required for the Δ94 SGs and is not required for the current Model E steam generators.

IN Guideline 2

The potentially harsh or inhospitable environmental conditions expected.

STP Assessment of IN Guideline 2

The operator would complete this action from within the Control Room. The design of the control room permits safe occupancy during abnormal conditions as described in STP evaluation for 10 CFR 50, Appendix A, General Design Criterion 19 (See UFSAR Section 3.1.2.2.10.1).

IN Guideline 3

A general discussion of the ingress/egress paths taken by the operators to accomplish functions.

STP Assessment of IN Guideline 3

The operator would complete this action from within the control room and ingress/egress paths are not necessary.

IN Guideline 4

The procedural guidance for required actions.

STP Assessment of IN Guideline 4

The operator action to lower the safety-grade SG PORV setpoints will be added to the STP Emergency Operating Procedures.

IN Guideline 5

The specific operator training necessary to carry out actions; including operator qualifications required to carry out actions.

STP Assessment of IN Guideline 5

Operator training on the purpose and required sequence will be conducted prior to steam generator replacement. The training will consist of classroom training for all licensed operators followed by simulator training using a small break LOCA scenario. All requalification crews will receive this training, and completion of the operator action within the required time will be evaluated as a critical task. If any crew fails to perform the task within the required time, it will be a basis for crew failure. The training will also become a part of the initial license operator training.

IN Guideline 6

Any additional support personnel and/or equipment required by the operators to carry out actions.

STP Assessment of IN Guideline 6

Currently, the STP EOPs have an operator action for a steam generator tube rupture (SGTR) similar to the operator action for the small break LOCA addressed by this proposed license amendment. The operator action for the SGTR is to raise the safety-grade SG PORV setpoints. The operator action credited in the small break LOCA analysis for the Δ94 SGs is to lower the safety-grade SG PORV setpoints. The purpose of the action is to provide a more rapid cooldown of the primary side by depressurizing the secondary side during a small break LOCA using the steam dumps first, then the SG PORVs, if steam dumps are unavailable. No additional support personnel and/or equipment is required by the operators to perform this action.

IN Guideline 7

A description of information required by the control room staff to determine whether such operator action is required, including qualified instrumentation used to diagnose the situation and to verify that the required action has successfully been taken. (In accordance with Regulatory

Guide 1.97, Revision 3, qualification of the instrumentation relied upon by the operators may be an important review issue. RG 1.97 defines Type A variables as "those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their functions for design basis accident events.")

STP Assessment of IN Guideline 7

Currently, the STP EOPs have an operator action for a steam generator tube rupture (SGTR) similar to the operator action for the small break LOCA addressed by this proposed license amendment. The operator action for the SGTR is to raise the safety-grade SG PORV setpoints. The operator action credited in the small break LOCA analysis for the Δ94 SGs is to lower the safety-grade SG PORV setpoints. The purpose of the action is to provide a more rapid cooldown of the primary side by depressurizing the secondary side during a small break LOCA using the steam dumps first, then the SG PORVs, if steam dumps are unavailable. The same indications to verify the operator action was taken successfully would be used in either accident, which is steam line pressure. These indications conform to RG 1.97 as described in the STP UFSAR, Table 7.5-1.

IN Guideline 8

The ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery.

STP Assessment of IN Guideline 8

The operators at STP are highly trained and proficient in their duties. The operator training includes an understanding of the expected system responses in order to ensure recovery from equipment failures or operator errors. Recent simulator exercises have demonstrated the ability of the STP operators to complete the required operator actions as written in the EOPs and to diagnose system responses.

STPNOC considers it unlikely the highly trained operator crews, using written procedures, 3-way communication and system response verification, would fail to complete the required operator actions. Operator training on the purpose and required sequence will be conducted prior to steam generator replacement.

The proposed operator action is required to be completed at or prior to 45 minutes after the accident. Consequently, STPNOC believes that sufficient time is available for the operators to complete the required actions, verify system responses, diagnose any errors in performing the required operator actions, and to complete any recovery actions necessary.

IN Guideline 9

Consideration of the risk significance of the proposed operator actions.

STP Assessment of IN Guideline 9

STPNOC has submitted the STP Level 2 Probabilistic Safety Assessment and Individual Plant Examination (PRA) in letter dated August 28, 1992 (ST-HL-AE-04193). The current STP PRA models the operator action to depressurize the RCS for a small break LOCA, including depressurizing the secondary side using the SG PORVs if the steam dumps are not available. The likelihood of failure of this operator action is modeled in the STP PRA top event OD (Operator Depressurize). The frequency of failure is based on sequence specific operator interviews and is 4.4×10^{-2} on demand. Since the operator action is currently modeled, the risk to core damage is unchanged.

ATTACHMENT 4

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

FOR THE

REPLACEMENT STEAM GENERATOR SMALL BREAK LOSS OF COOLANT ACCIDENT REQUIRED OPERATOR ACTIONS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

STP has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment does not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

PROPOSED CHANGE

STPNOC proposes to revise the STP UFSAR, as shown by the draft in Attachment 5, to reflect the revised small break LOCA analysis for the Δ94 SGs. The proposed license amendment herein addresses the operator action to reduce the SG PORV setpoint consistent with the revised small break LOCA analysis for the Δ94 SGs.

NO SIGNIFICANT HAZARDS ANALYSIS

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed operator action associated with the re-analysis of the Δ94 SGs will not result in a significant increase in the probability of an accident previously evaluated. The initiators of any design basis accident are not affected by this operator action. The operator action would facilitate the automatic mitigation capability of the SG PORVs, and would not initiate the mitigating safety function. The operator action will be incorporated into the EOPs and would not be performed until after the initiation of an accident. The automatic actuation of the SG PORVs is not a new design feature. The effects of inadvertent opening of a single steam dump, relief or safety valve are currently analyzed as described in Section 15.1.4 of the UFSAR. Consequently, there is no significant impact on any previously evaluated accident probabilities.

The proposed operator action associated with the re-analysis of the Δ94 SGs does not result in a significant increase in the consequences of any accidents previously evaluated. The operator action will not adversely affect the integrated ability of the plant systems to perform their intended safety functions to mitigate the consequences of a small break LOCA, or any other accident previously evaluated. In fact, the re-analysis has demonstrated that the use of the operator action reduces the consequences of a small break LOCA in that the Peak Cladding Temperature for the most limiting small break LOCA transient is reduced and continues to be substantially below the acceptance limit of 10 CFR 50.46.

The operator action does not affect the integrity of any fission product barrier such that their function in the control of radiological consequences is not affected. The radiological consequences for the small break LOCA presented in the UFSAR remain unchanged as a result of the proposed operator action.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of any physical changes to the existing facility. The operator action does not represent a different initiator for any design basis accident and does not create new design basis scenarios. Small break LOCA mitigation, utilizing a combination of automatic and manual actions, is already part of the STP licensing basis. Written procedures address those operator actions required for small break LOCA mitigation. The current STP EOPs have an operator action for a steam generator tube rupture (SGTR) similar to the operator action for the small break LOCA addressed by this proposed license amendment. The operator action for the SGTR is to raise the safety-grade SG PORV setpoints. The operator action credited in the small break LOCA analysis for the Δ94 SGs is to lower the safety-grade SG PORV setpoints. The purpose of the action is to provide a more rapid cooldown of the primary side by depressurizing the secondary side during a small break LOCA using the steam dumps first, then the SG PORVs, if steam dumps are unavailable. The inadvertent operation of a single steam dump, relief or safety valve is currently addressed in UFSAR Section 15.1.4.

The proposed amendment does not alter any original design specification, such as seismic requirements, electrical separation requirements and environmental qualification, and is not the result of any physical changes to the facility. In addition, the proposed amendment does not result in exposure of additional equipment used in accident mitigation to an adverse environment beyond that currently identified in the UFSAR.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

The proposed operator action does not involve a significant reduction in the margin of safety. The plant systems required for the mitigation of any design basis accidents will continue to be able to perform their safety function. In fact, the re-analysis has demonstrated that the use of the operator action reduces the consequences of a small break LOCA in that the Peak Cladding Temperature for the most limiting small break LOCA transient is reduced and continues to be substantially below the acceptance criteria of 10 CFR 50.46.

ATTACHMENT 5

UFSAR PAGES

RELATED TO

REPLACEMENT STEAM GENERATOR
SMALL BREAK
LOSS OF COOLANT ACCIDENT

The following UFSAR section is an additional section to the existing UFSAR and to be inserted after the existing UFSAR section 15.6.5.

15.6.6 Loss Of Coolant Accidents (Δ94 Replacement Steam Generator)

This section describes the LOCA analysis in support of the Δ94 Replacement Steam Generator (RSG), and parallels the information documented in Section 15.6.5 for the existing Model E Steam Generator design. Only that information that is unique for the Δ94 RSG LOCA analyses is included here. All other information related to general descriptions and bounding analyses refers to Section 15.6.5.

For large break LOCA, the short-term consequences are bounded by the analysis for the Model E Steam Generator as presented in the body of Section 15.6.5.

15.6.6.1 Identification of Causes and Frequency Classification. Refer to Section 15.6.5.1.

15.6.6.2 Sequence of Events and Systems Operations. Refer to Section 15.6.5.2, noting that the time for realignment of the ECCS to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel has been calculated to be greater than 5.5 hours, for the Δ94 steam generators, after initiation of the LOCA. Also note that the block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event, for the Δ94 steam generators, is provided in Figure 15.6-73.

15.6.6.3 Environmental Consequences. Refer to Section 15.6.5.3.

15.6.6.4 Core and System Performance. Refer to Section 15.6.5.4.

15.6.6.4.1 Mathematical Model: The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10 CFR 50 (Ref. 15.6-2).

Large Break LOCA Evaluation Model

Refer to Section 15.6.5.4.1 - Large Break LOCA Evaluation Model

Small Break LOCA Evaluation Model

Refer to Section 15.6.5.4.1

The STPEGS small break LOCA analysis for the Δ94 RSGs is fundamentally the same as the small break LOCA described in Section 15.6.5.4.1. One additional unique modeling features is:

- Credit for operator action to reduce the SG PORV setpoint (Ref: NRC SE for amendment)

The operator action credited in the small break LOCA analysis is to lower the SG PORV setpoints at least to 1000 psig within 45 minutes after accident initiation. The purpose of the action is to provide a more rapid cooldown of the primary side by depressurizing secondary side during a small break LOCA using the steam dumps first, then the SG PORVs, if steam dumps are unavailable. The SBLOCA analysis only takes credit for the SG PORVs, since the Steam Dumps are not safety-grade. The operator actions to lower secondary side pressure using either the SG PORV or the steam dumps are achievable from the plant control room. Consistent with the intent of the Emergency Operating Procedures (EOPs), the operator action credited in the small break LOCA analysis does not initiate a mitigating safety function, but facilitates the automatic mitigation capability of the SG PORVs.

15.6.6.4.2 Input Parameters and Initial Conditions: Refer to Section 15.6.5.4.2 for information about large break LOCA.

Table 15.6-15 lists important input parameters and initial conditions used in the Δ94 RSG small break LOCA analysis.

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS cold leg temperature consistent with the STP Δ94 RSG design.

The analyses evaluated a range of operating temperatures (582.3°F$\leq T_{AVG} \leq$ 593.0°F) plus RCS temperature uncertainties of ±6°F, and a range of Main Feedwater (MFW) temperatures (390.0°F $\leq T_{MFW} \leq$ 440.0°F).

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Refs. 15.6-17 through 15.6-19). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time the postulated LOCA occurs and include such items as the core peaking factors and the performance of the ECCS system. Decay heat generated throughout the transient is also conservatively calculated.

15.6.6.4.3 Results:

Large Break Results

Refer to Section 15.6.5.4.3 - Large Break Results

Small Break Results

A small break LOCA analysis was performed for STP in support of:

- Steam Generator Replacement,
- Conversion to T_{COLD} Upper Head, and
- Operator action to reduce the SG PORV setpoint.

Due to the asymmetries presented by the unique features of the STP design, multiple combinations of limiting single failures were considered These included common mode failures (emergency diesel generator failure) that could lead to various failures of pumped ECCS, AFW , and SG PORVs. Sensitivity studies of these multiple scenarios were performed to establish the limiting scenario in terms of small break LOCA PCT results. The limiting scenario was found to be a postulated actuation signal failure leading to the unavailability of the emergency bus for loops A and D. The Loop A and D SG PORVs are also assumed inoperable with the loss of Train A emergency power because of the ultimate discharge of the associated nitrogen accumulators. The failure of the SG PORVs to open was included in the limiting scenario. A failure of the SG PORV to close was concluded not to represent a degradation to the mitigation capability for SBLOCA. The results of this limiting scenario are presented here.

Based upon the results of the LOCA sensitivity studies (Ref. 15.6-18), the limiting small break was found to be less than a 10-inch-diameter rupture of the RCS cold leg. Therefore, a range of small break analyses are presented which establishes the most limiting break size. From these calculations, the 2-inch equivalent diameter break was found to be limiting. The low end of the loop operating temperature range and the low end of the MFW operating range were determined to provide the most limiting small break LOCA results.

The results of these analyses are summarized in Tables 15.6-16 and 15.6-17.

Figures 15.6-59 through 15.6-72 present the principal parameters of interest for the small break ECCS analyses. For all cases presented in the UFSAR the following transient parameters are presented:

1. RCS pressure
2. Core mixture height
3. Peak clad temperature (if core uncovery is predicted to occur)
4. Core power after reactor trip (common to all transients)
5. Small break safety injection flow rate versus RCS pressure (common to all transients)
6. Small break power shape (common to all transients)

For the limiting 2-inch, low loop temperature, low MFW temperature case, the following additional transient parameters are presented:

7. Core steam flow rate

8. Hot spot fluid temperature

9. Rod film heat transfer coefficient

The results presented in Table 15.6-17 bound both Zirc-4 fuel and fuel with ZIRLO cladding, instrument tubes, thimbles, and grids.

The maximum calculated peak clad temperature for all small breaks analyzed are shown in Table 15.6-17. These results are within all acceptance criteria limits of 10 CFR 50.46.

REFERENCES

Section 15.6:

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15.6-2 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 10CFR50.

15.6-3 "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG-75/014, October 1975.

15.6-4 Bordelon, F. M., Massie, H. W., and Borden, T. A. "Westinghouse ECCS Evaluation Model-Summary", WCAP-8339 (Nonproprietary), July 1974.

15.6-5 Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant", WCAP-8302, June 1974 (Proprietary) and WCAP-8306, June 1974 (Nonproprietary).

15.6-6 Kelly, R. D., et al., "Calculated Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)", WCAP-8170, June 1974 (Proprietary) and WCAP-8171, June 1974 (Nonproprietary).

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15.6-8 Young, M. Y. et. al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A Rev. 2, (Proprietary), 1981.

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15.6-10 "Westinghouse ECCS Evaluation Model, 1981 Version", WCAP-9220-P-A, Rev. 1 (Proprietary), WCAP-9221-A, Rev. 1 (Nonproprietary), February 1982.

15.6-11 Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassallo of the Nuclear Regulatory Commission, Letter Number NS-CE-924 dated January 23, 1976.

15.6-11a Not used.

15.6-11b Not used.

15.6-11c Not used.

REFERENCES (Continued)

15.6-11d Not used.

15.6-11e Not used.

15.6-11f Not used.

15.6-11g Not used.

15.6-11h Letter, E. P. Rahe (Westinghouse) to James R. Miller (USNRC), NS-EPRS-2679, November 9, 1982.

15.6-12 Not used.

15.6-13 Not used.

15.6-13a Lee, N., Rupprecht, S. D., Tauche, W. D., Schwarz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10054-P-A, August 1985.

15.6-13b Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code", WCAP-10079-P-A, August 1985.

15.6-13c Frantz, E. R., Huang, P. H., Ofstun, R. P., "Comparison of the TREAT and NOTRUMP Small Break LOCA Transient Results", WCAP-11232, September 1986.

15.6-14 Skwarek, R., Johnson, S., Meyer, P., "Westinghouse Emergency Core Cooling System Small Break October 1975 Model", WCAP-8970, April 1977 (Proprietary).

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15.6-16a Letter, E. P. Rahe (Westinghouse) to R. L. Tedesco (USNRC), Letter Number NS-EPR-2538, December 22, 1981.

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15.6-26 Rupprecht, S. D., et. al., "W Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A with (Proprietary) and WCAP-11372-A (Non-Proprietary)

15.6-27 Bordelon, F. M., "LOCATA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary), 1974.

TABLE 15.6-15

INPUT PARAMETERS USED IN THE Δ94 RSG SBLOCA ANALYSIS

Core Power[1], MWt		3,800
Total Peaking Factor		2.70
Axial Peaking Factor		1.62
Power Shape		See Figure 15.6-72
Fuel Assembly Array		17 x 17 XL, V5H
Safety Injection Pumped Flow		See Figure 15.6-71
Initial Loop Flow, lb/sec		9,841
RCS Temperature,[2] °F		$582.3°F \leq T_{AVG} \leq 593.0°F$
Reactor Coolant Pressure, psia		2,300[3]
Steam Pressure, psia		1107 / 904[4]
Steam Generator Tube Plugging Level, %		10
Main Feedwater Temperature, °F		$390.0°F \leq T_{MFW} \leq 440.0°F$
AFW Flow, gpm[5]	Loop A	0
	Loop B	500
	Loop C	500
	Loop D	0
AFW Temperature, °F		212
MSSV Setpressures, psig[6]	Valve 1	1285
	Valve 2	1295
	Valve 3	1305
	Valve 4	1315
	Valve 5	1325
Steam Generator PORV setpressure (automatic), psig[7]		1225

1 Two percent is added to this power to account for calorimetric error.
2 Analysis incorporates effects of ±6°F uncertainty.
3 This value bounds the actual pressure of 2250 psia plus 46 psi uncertainty.
4 Corresponds to the range of MFW temperatures (390°F / 440°F, respectively).
5 Corresponds to the LSF scenario leading to the limiting PCT.
6 Value is adjusted in the analysis to include 3% accumulation, and 3% uncertainty for each valve. Data represents each of five valves on each of four steam generators.
7 Value is adjusted in the analysis to include 4.58% uncertainty. While there is no "accumulation" for this PORV, an additional 4.9% was added to allow for full flow.

TABLE 15.6-16

Δ94 RSG SMALL BREAK
TIME SEQUENCE OF EVENTS

	Time (seconds)		
Event (Equivalent Break Size at Lo-Lo* Conditions)	3 in.	2 in.	1.5 in.
Start (Accident Initiation)	0.0	0.0	0.0
Reactor Trip Signal, sec	21.9	52.9	100.1
Top of Core Uncovered, sec	615	2025	N/A
Accumulator Injection Begins, sec	N/A	N/A	N/A
Peak Clad Temp Occurs, sec	1186.3	2657.6	N/A
Top of Core Covered, sec	1833	3291	N/A

*Lo-Lo - Low loop temperature (582.3°F - 6°F uncertainty); Low MFW temperature (390°F).

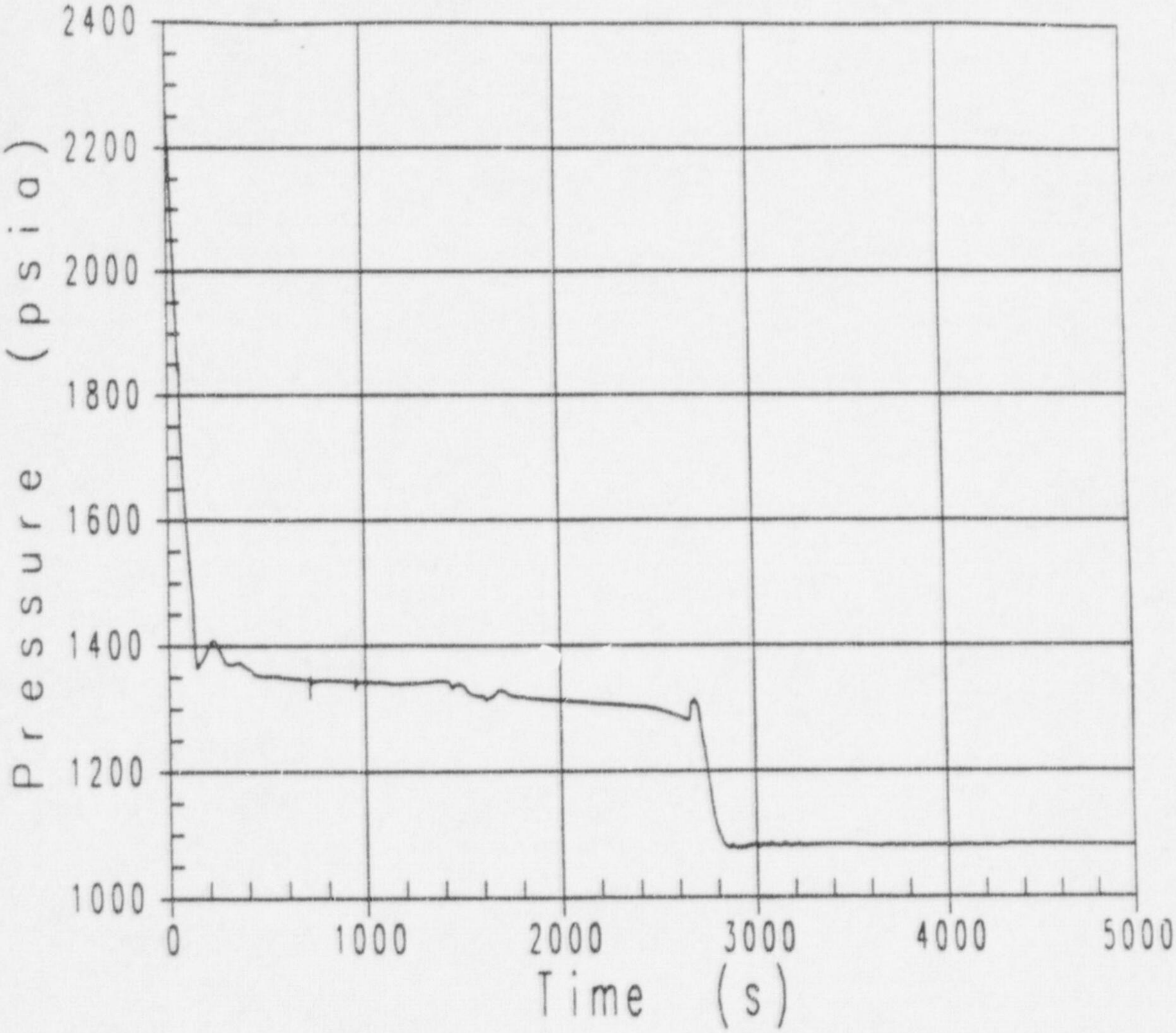
TABLE 15.6-17

Δ94 RSG SMALL BREAK ANALYSIS RESULTS

Event (Equivalent Break Size at Lo-Lo* Conditions)	3 in.	2 in.	1.5 in.
Peak clad temperature, °F	1481	1654	NCU**
Elevation, ft.	13.25	13.25	N/A
Max. local ZR/H20 reaction, %	0.54	0.88	N/A
Elevation, ft.	13.25	13.25	N/A
Total ZR/H_2O reaction	<1.0	<1.0	<1.0
Hot rod burst time, sec.	N/A	N/A	N/A
Elevation, ft.	N/A	N/A	N/A

*Lo-Lo - Low loop temperature (582.3°F - 6°F uncertainty); Low MFW temperature (390°F).

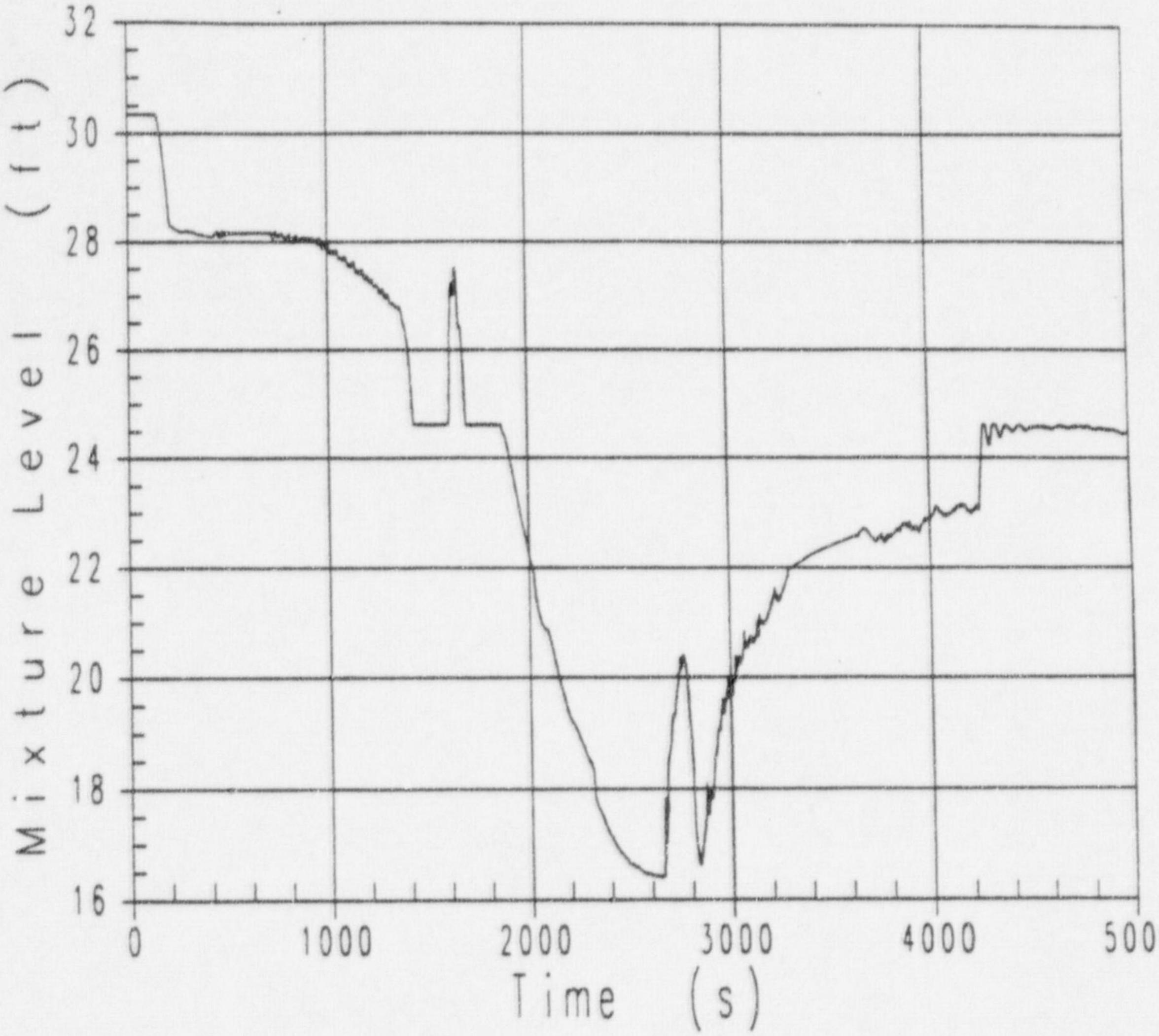
**No Core Uncovery was observed for this case.



SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

RCS DEPRESSURIZATION TRANSIENT
(2-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-59 Revision 7

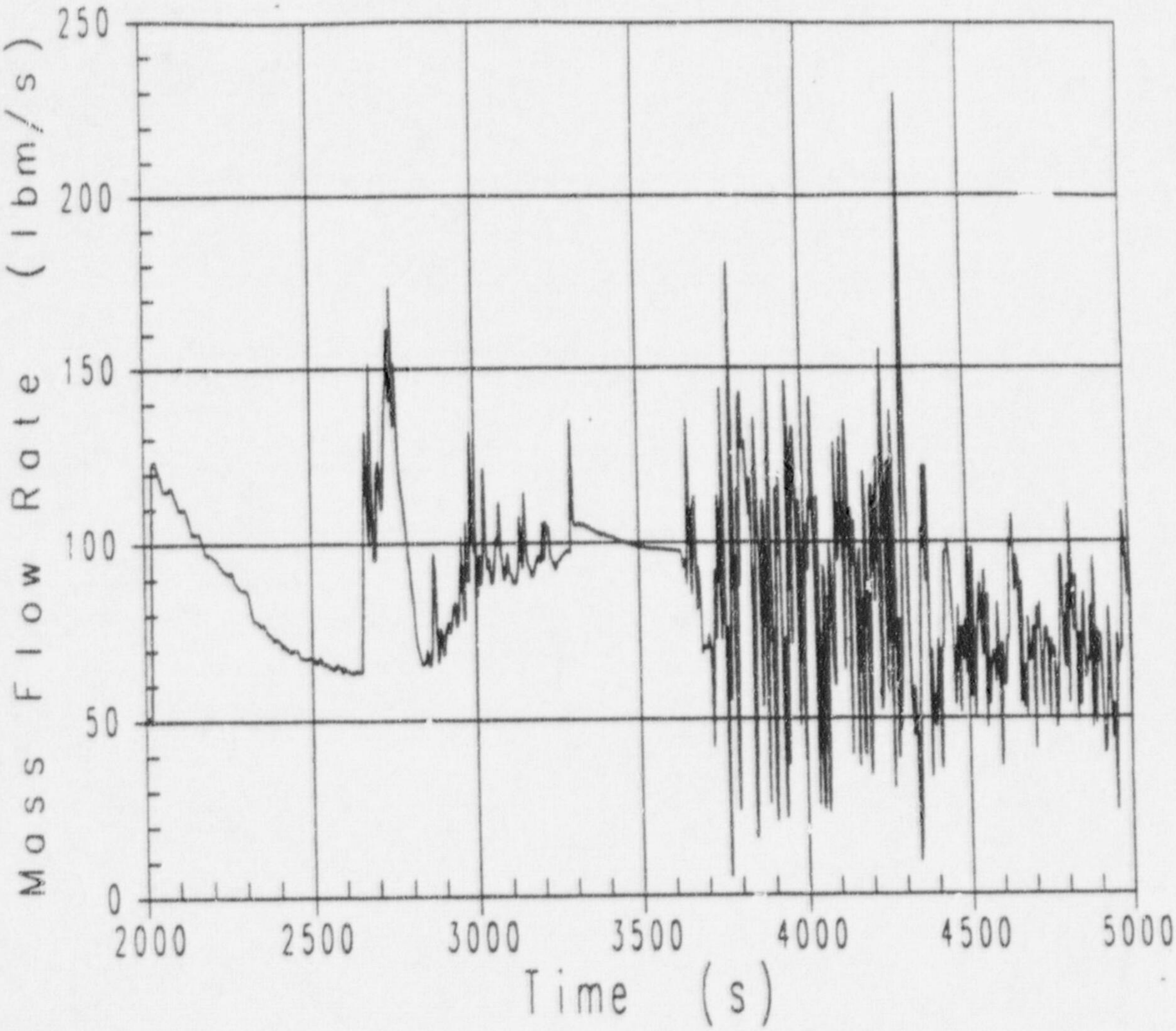


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

CORE MIXTURE HEIGHT

(2-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-60 Revision 7



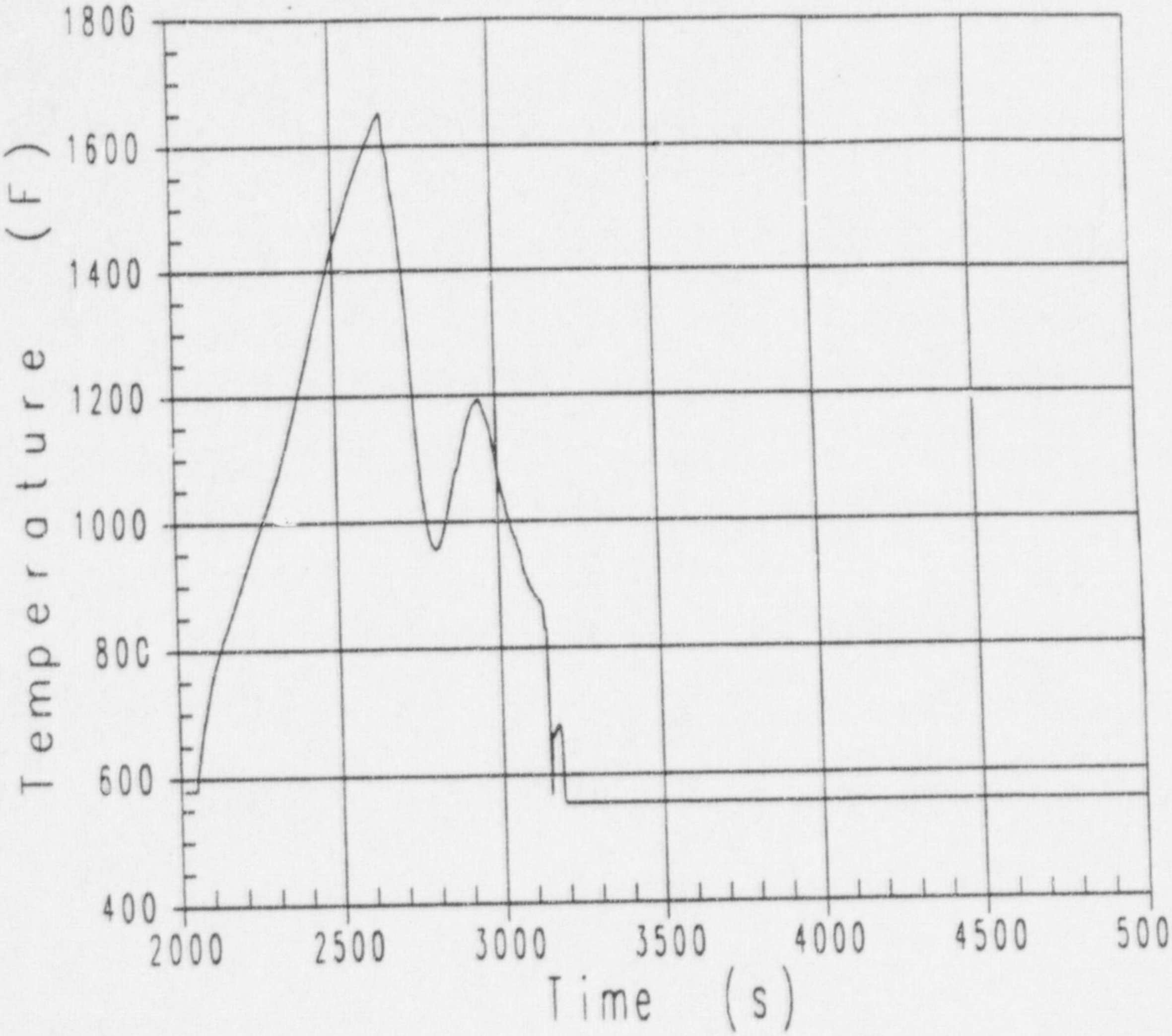
SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

STEAM FLOW

(2-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-61

Revision 7

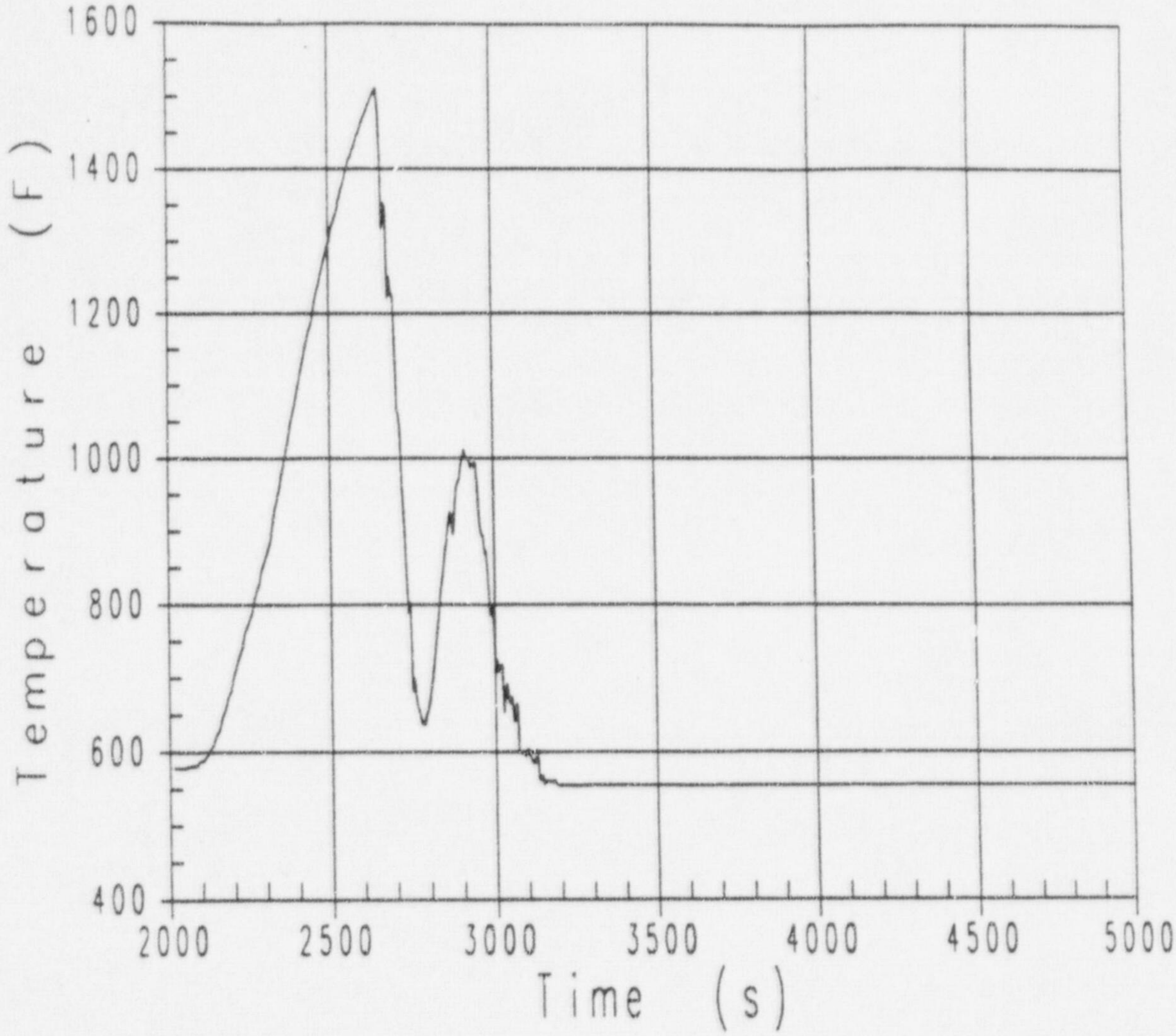


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

CLAD TEMPERATURE TRANSIENT

(2-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-62 Revision 7

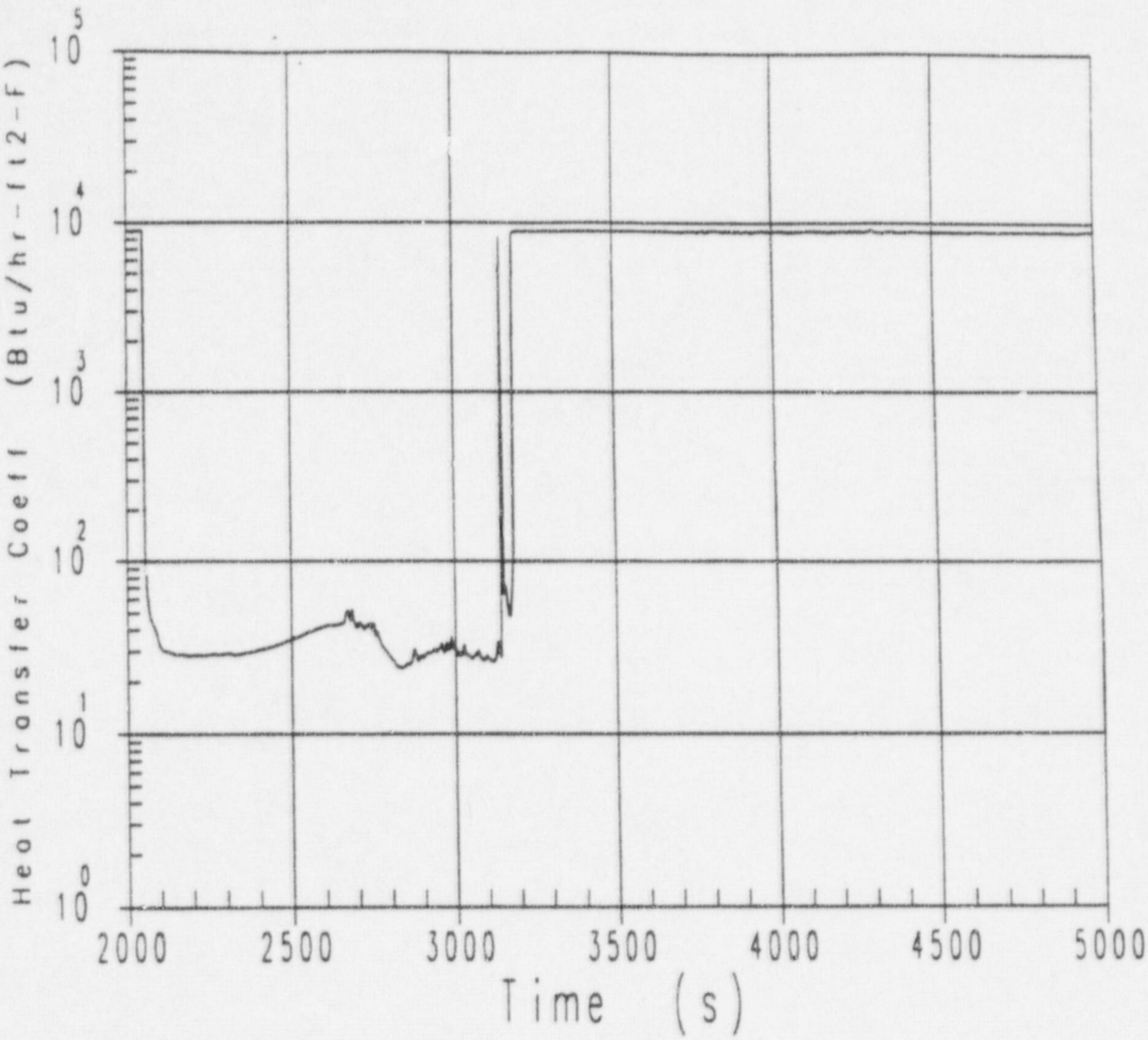


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

HOT SPOT FLUID TEMPERATURE

(2-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-63 Revision 7

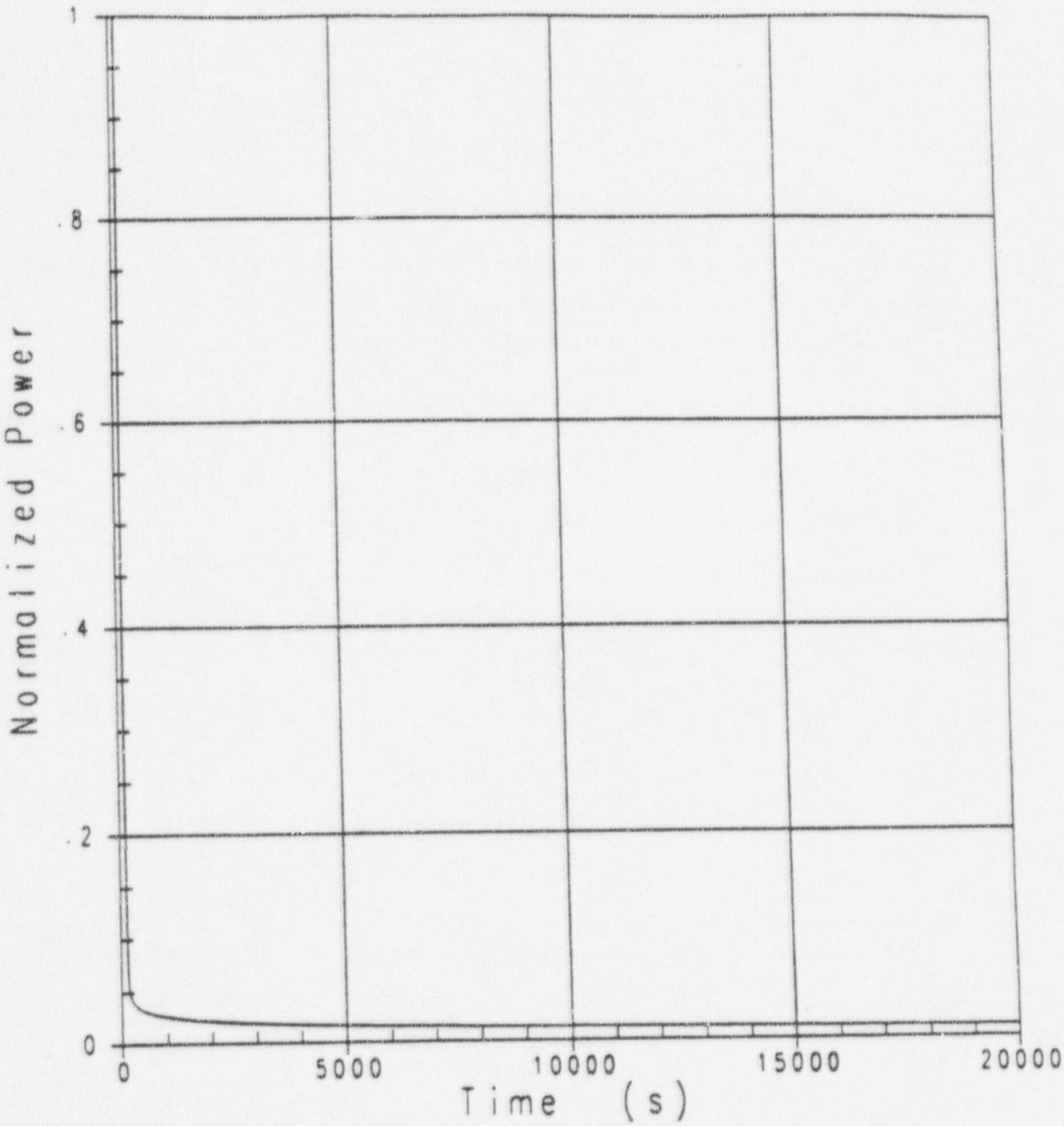


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

ROD FILM HEAT TRANSFER COEFFICIENT

(2-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-64 Revision 7

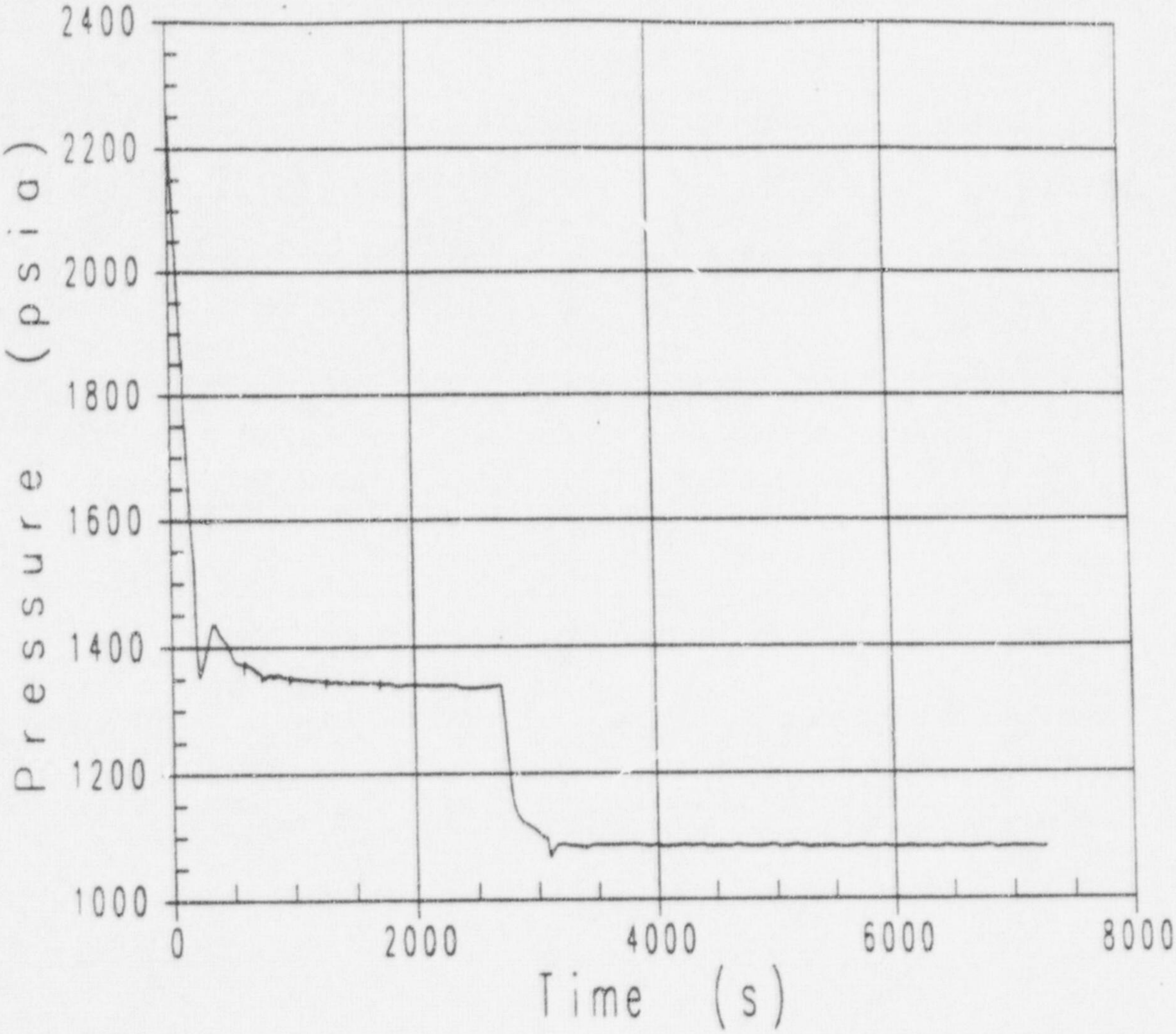


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

CORE POWER AFTER REACTOR TRIP

Common to all cases

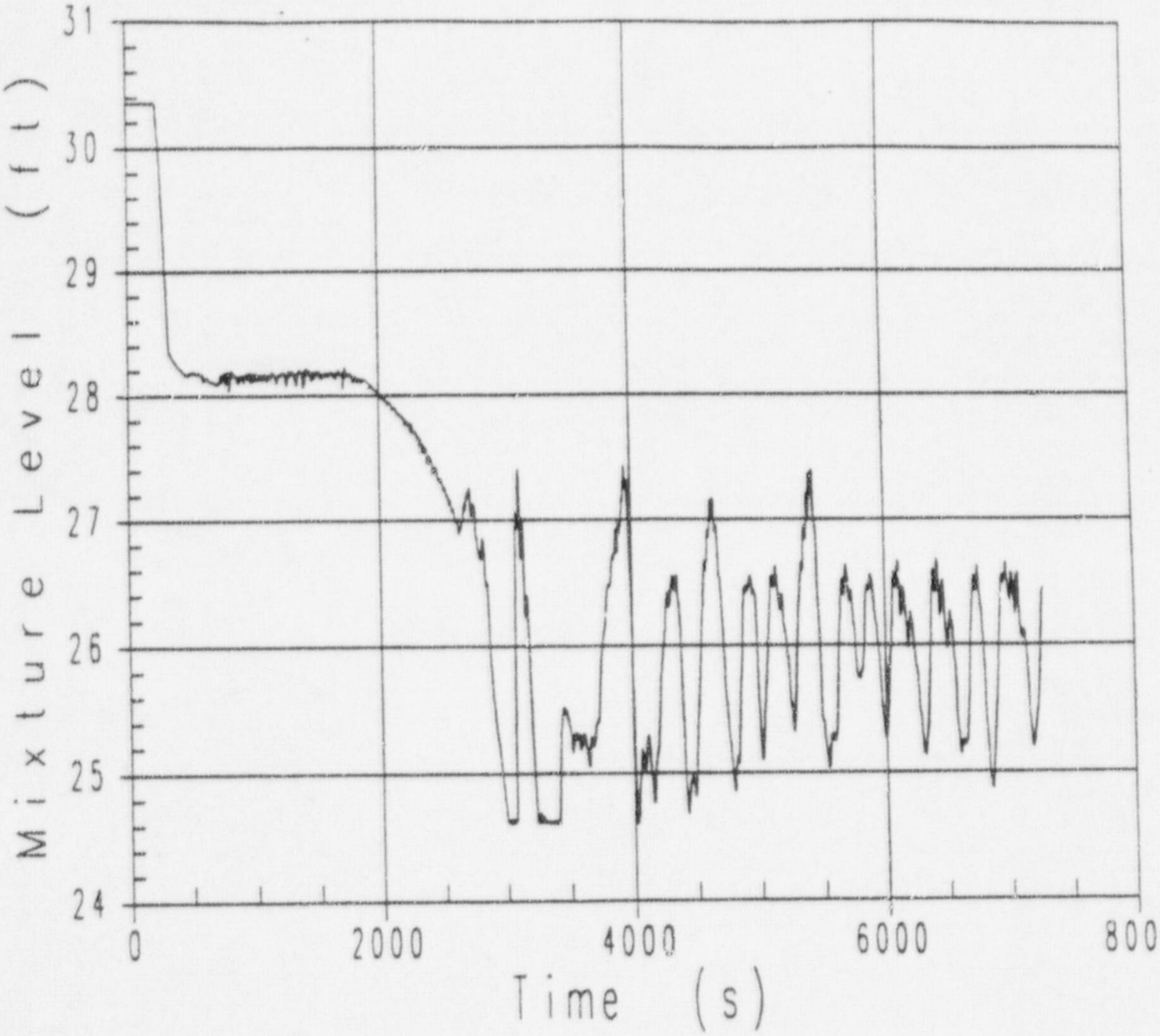
Figure 15.6-65 Revision 7



SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

RCS DEPRESSURIZATION TRANSIENT
(1.5-INCH BREAK Low loop temperature. Low MFW temperature)

Figure 15.6-66 Revision 7

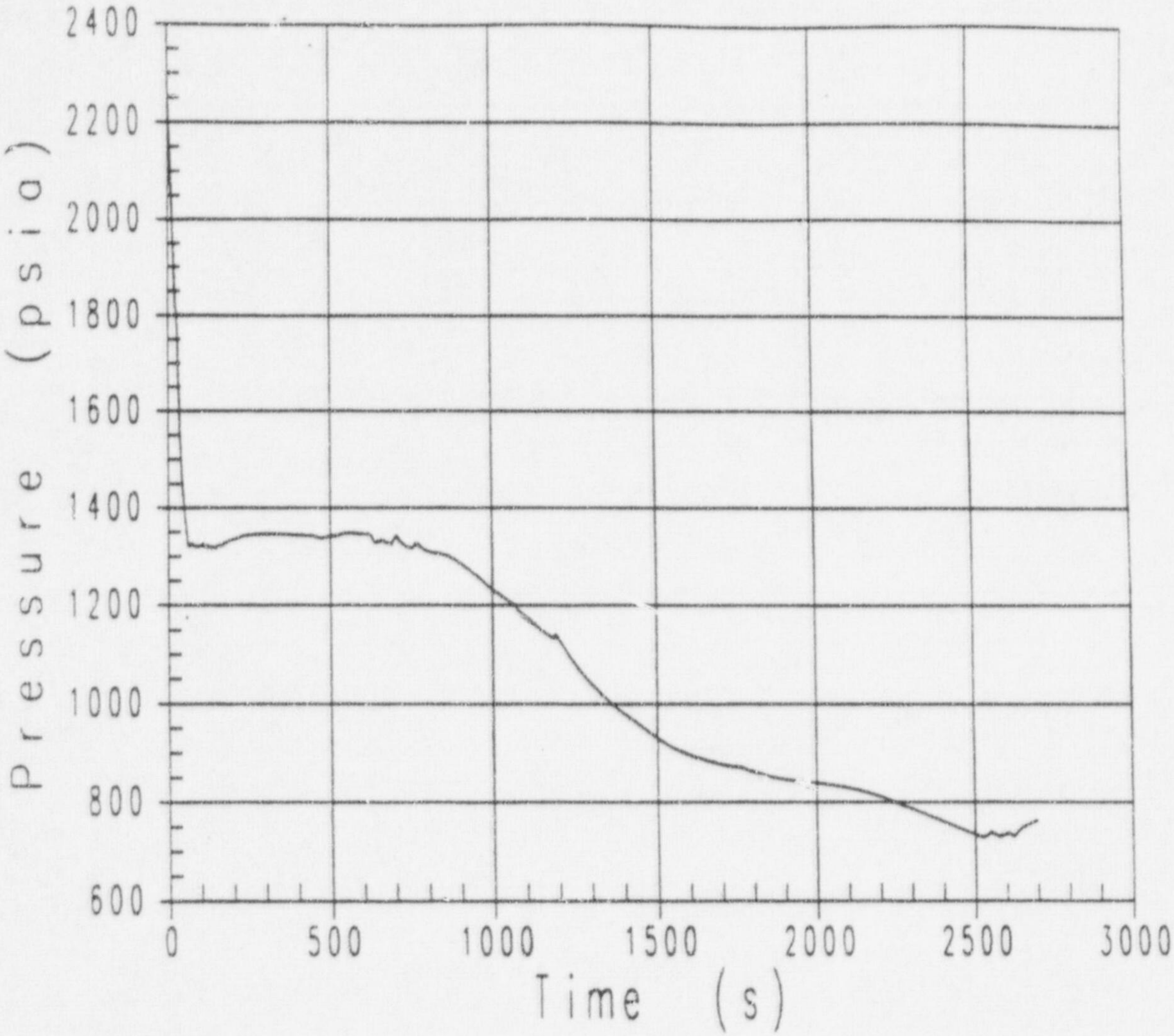


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

CORE MIXTURE HEIGHT

(1.5-INCH BREAK Low loop temperature, Low MFW temperature)

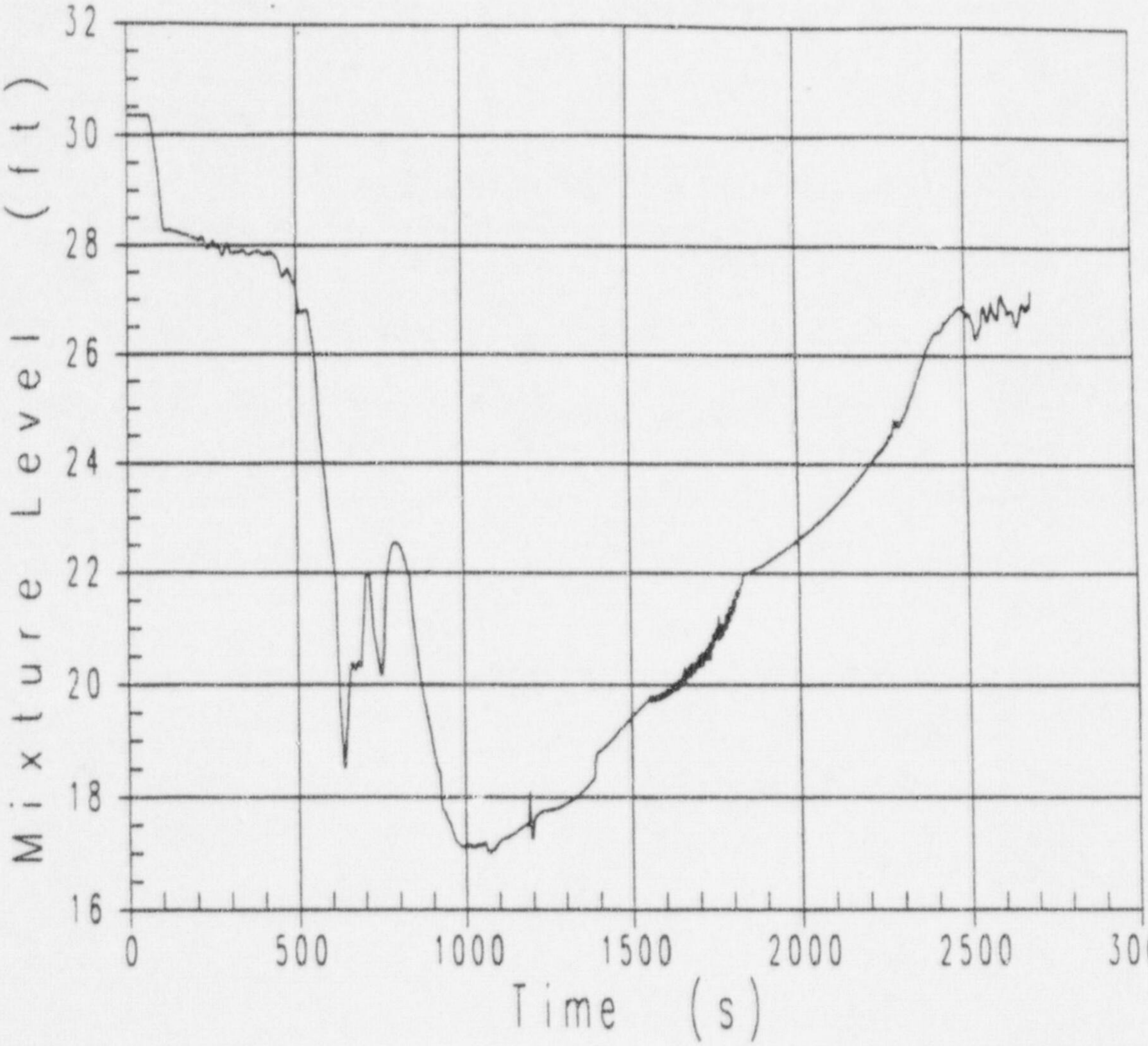
Figure 15.6-67 Revision 7



SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

RCS DEPRESSURIZATION TRANSIENT
(3-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-68 Revision 7

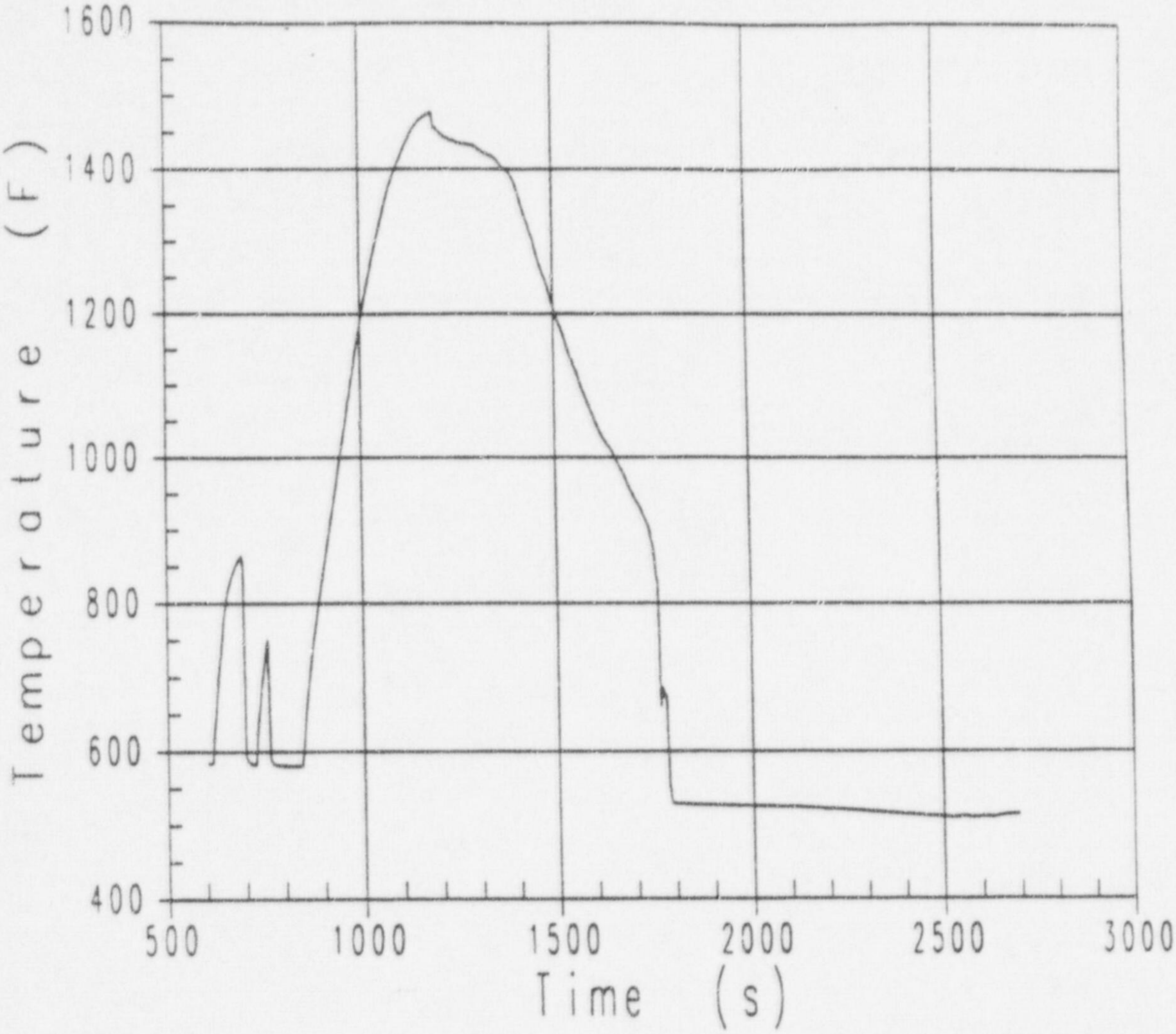


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

CORE MIXTURE HEIGHT

(3-INCH BREAK Low loop temperature, Low MFW temperature)

Figure 15.6-69 Revision 7

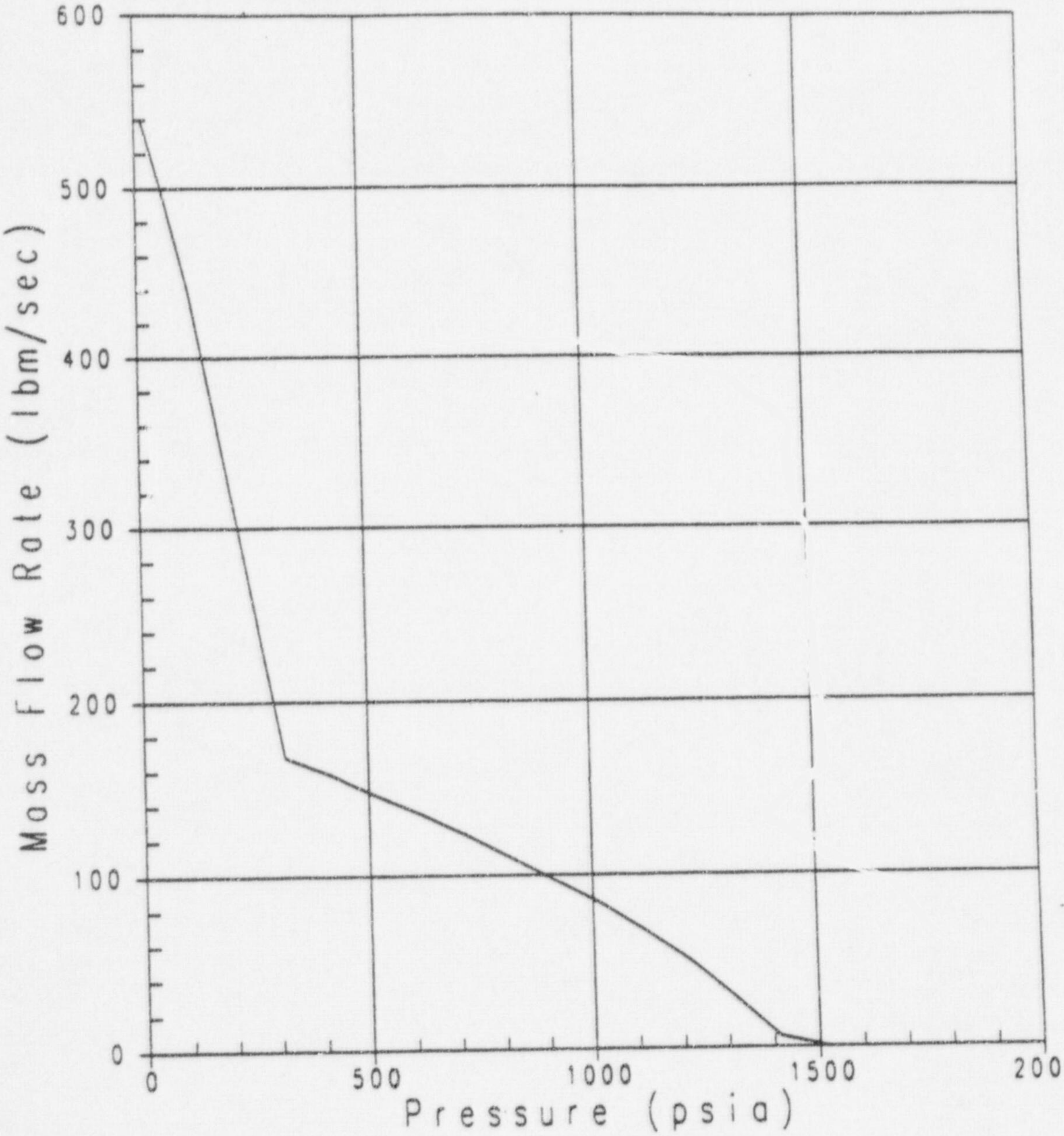


SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

CLAD TEMPERATURE TRANSIENT

(3-INCH BREAK Low loop temperature, Low MFW temperature)

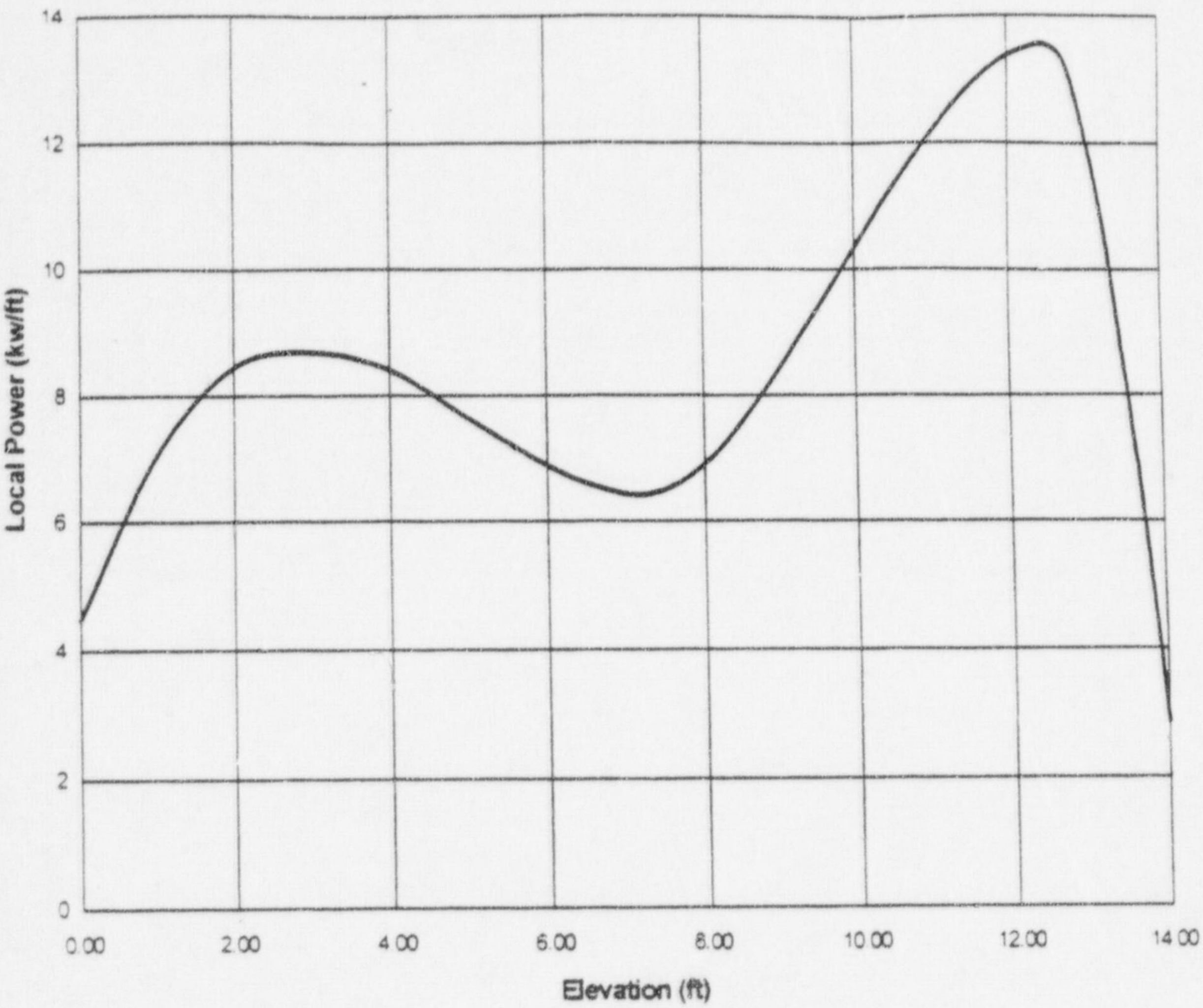
Figure 15.6-70 Revision 7



SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

SMALL BREAK SAFETY INJECTION FLOW RATE VERSUS RCS PRESSURE

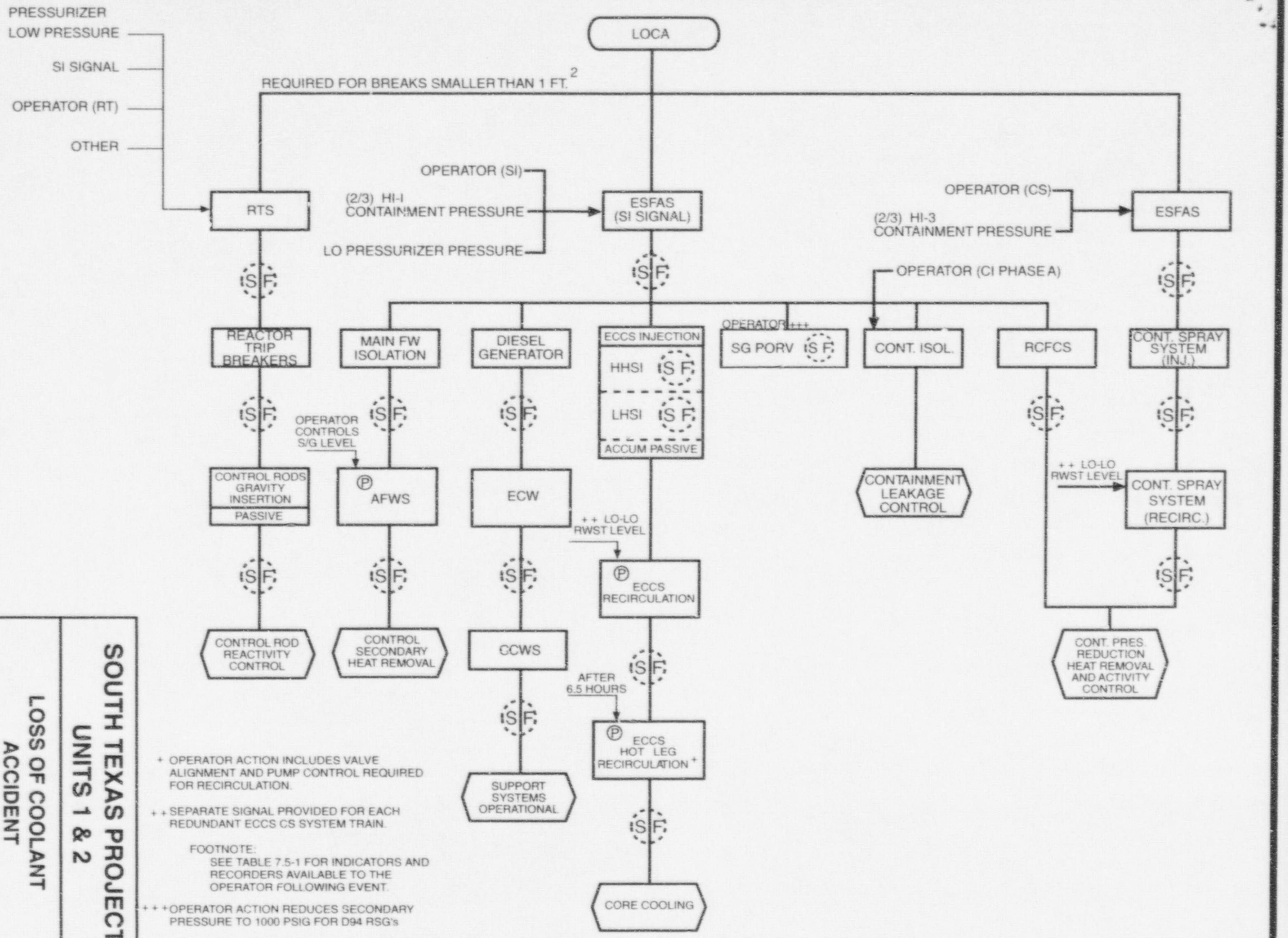
Figure 15.6-71 Revision 7



SOUTH TEXAS PROJECT
UNIT 1 - Δ94 RSG

SMALL BREAK POWER SHAPE

Figure 15.0-72 Revision 7



\+ OPERATOR ACTION INCLUDES VALVE ALIGNMENT AND PUMP CONTROL REQUIRED FOR RECIRCULATION.

++ SEPARATE SIGNAL PROVIDED FOR EACH REDUNDANT ECCS CS SYSTEM TRAIN.

FOOTNOTE:
SEE TABLE 7.5-1 FOR INDICATORS AND RECORDERS AVAILABLE TO THE OPERATOR FOLLOWING EVENT.

+++ OPERATOR ACTION REDUCES SECONDARY PRESSURE TO 1000 PSIG FOR D94 RSG's

SOUTH TEXAS PROJECT
UNITS 1 & 2

LOSS OF COOLANT ACCIDENT

Figure 15.6-73 **Revision 7**