MARKED-UP TECHNICAL SPECIFICATIONS PAGES
(NUREG 1399)
PAGE 6-25

ATTACHMENT 3 TO TXX-92323
PAGE 1 OF 3

TS-92-025



WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET 6
CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 6
(W Proprietary).
(Methodology for Specification 3.2.2-Heat Flux Hot Channel 6
Factor (W(z) surveillance requirements for $F_Q$
Methodology).)

Delete

WCAP-8200, "WFLASH, A FORTRAN-IV COMPUTER PROGRAM FOR 6
SIMULATION OF TRANSIENTS IN A MULTI-LOOP PWR," Revision 2,
June 1974 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot 6
Channel Factor.)

WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model, 6
February 1978 Version," February 1978 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot 6
Channel Factor.)

INSERT A

The core operating limits shall be determined so that all applicable 6
limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or 6
supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 In addition to the applicable reporting requirement of Title 10, Code of Federal Regulations, special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

TS-92-025

INSERT A

WCAP-10079-P-A, "NOTRUMP, A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985, (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE", August 1985, W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

WCAP-11145-P-A, "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL GENERIC STUDY WITH THE NOTRUMP CODE", October 1986, W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

WESTINGHOUSE LETTER, WPT-14670

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTR:C STATION
UNIT NU BER 1
SMALL BREAK LOCA USING W-FLASH PEAK CLAD TEMPERATURE (PCT)

JULY 13, 1992

ENCLOSURE 1 TO TXX-92323
(TOTAL  GES = 3)

DNB

Westinghouse Electric Corporation

Energy Systems

WPT-14670
ET-NSL-OPL-II-92-..9

Box 355
Pittsburgh Pennsylvania 15230-0355

July 13, 1992

Mr. W. J. Cahill, Jr., Executive Vice President
Nuclear Engineering & Operations
TU Electric Company
P. O. Box 1002
Glen Rose, Texas 76043

Attention: W. Choe

S.O. No. TBX-4708

Ref: 1. WPT-13635
2. WPT-14479

(No Response Required)

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC STATION
UNIT NUMBER 1
SMALL BREAK LOCA USING W-FLASH PEAK CLAD TEMPERATURE (PCT)

Dear Mr. Cahill:

As recently discussed with Mr. Whee Choe of your organization, the current Comanche Peak Unit 1 small break LOCA analysis has been evaluated relative to open issues previously provided via Reference 1. Our investigation of PI-91-005, "Small Break LOCA Burst and Blockage Considerations", has resulted in a Peak Clad Temperature (PCT) over the 2200°F criteria of 10CFR50.46, when applied to the original Unit 1 analysis which utilizes the W-FLASH evaluation model. However, Westinghouse does not consider this condition to be reportable to the NRC as we have recently supplied TU Electric (Reference 2) an engineering assessment based on application of the NOTRUMP Small Break LOCA methodology which demonstrates compliance with the 2200°F criteria.

Please find attached the results of our evaluation in this matter.

If there are any questions, please contact Craig Thompson on 412/374-4409 or Roy Owoc on 412/374-4037.

Very truly yours,

C. Benton for

J. L. Vota, Manager
Comanche Peak Projects

R. H. Owoc

Attachment

# COMANCHE PEAK STEAM ELECTRIC STATION UNIT NO.1 SMALL BREAK LOCA LICENSING BASIS PEAK CLADDING TEMPERATURE OVER THE 2200°F 10CFR50.46 CRITERIA

## INTRODUCTION

Westinghouse provided TU Electric with text (Reference 1) as part of the annual reporting requirement of 10CFR50.46. Attachment 2 to Reference (1) also presented several open items which Westinghouse was investigating that were considered to be too new to require reporting as a permanent change to the Westinghouse ECCS Evaluation model. The investigation for "SMALL BREAK LOCA BURST AND BLOCKAGE CONSIDERATIONS", with consideration for the most limiting time in life, has been completed, and Westinghouse has determined that this concern, when applied to CPSES-1 current licensing basis (W-Flash SBLOCA EM), results in a Peak Cladding Temperature (PCT) over the 2200°F criteria of 10CFR50.46. Westinghouse does not consider this condition to be reportable to the NRC as a substantial safety hazard, since Westinghouse has recently supplied TU Electric with an engineering assessment (Reference 2) based on application of the NOTRUMP Small Break LOCA evaluation model to CPSES-1, which demonstrated compliance with the 2200°F criteria.

## TECHNICAL DISCUSSION

While References (1) and (2) have already provided a detailed technical discussion, an abbreviated discussion is provided for clarity. Reference (1) pointed out that the small break LOCA burst model could go outside the intended range and that such extrapolation could result in premature burst for high pressure rods or failure to predict burst when burst should occur for low pressure rods. When this condition was corrected, burst occurred for lower pressure rods at higher temperatures, which caused the Zirconium water reaction to occur very rapidly, leading to higher calculated Peak Cladding Temperatures.

The CPSES-1 small break LOCA licensing basis analysis was performed with the W-FLASH Small Break LOCA Evaluation Model and fuel parameters based on 500 psig backfill pressure at beginning of life conditions. The Cycle 2 reload introduced fuel at 275 psig. When both fuel conditions were evaluated, using the corrected burst model, an increase in PCT occurred. This effect, when combined with the series of previous 10CFR50.59 safety evaluations which increased the small break PCT to 2133.65°F, have resulted in an increase in PCT above the 2200°F criteria for the CPSES-1 small break LOCA licensing basis.

## CONCLUSION

A new small break LOCA analysis, using the NOTRUMP small break evaluation model, has been performed for CPSES-1. This new analysis, using all changes previously evaluated under the provision of 10CFR50.59, calculated a low PCT showing a large amount of margin to the 10CFR50.46 requirement of 2200°F. This result indicates that NOTRUMP calculates improved core cooling when compared to the older W-FLASH model. While the W-FLASH analysis has been evaluated to have a PCT over the 2200°F criteria, the NOTRUMP analysis ameliorates any concern with regard to safe operation or for compliance with 10CFR50.46 criteria.

Further, the NOTRUMP analysis has gone through the Westinghouse Appendix B quality assurance program and is therefore considered acceptable as a licensing basis analysis. This included a comparison of the current CPSES-1 17x17 Standard Fuel THRIVE data with the values used in the 3 inch NOTRUMP analysis. Differences were limited to pressure drops in the downcomer and core regions. Because the values for pressure drops used in the NOTRUMP analysis are higher in the downcomer and only 0.1% lower in the core, these differences would have negligible effect on the calculated PCT.

REFERENCES

1. WPT-13635, J. L. Vota (W) to Mr. W. J. Cahill, Jr. (TU), "Comanche Peak Steam Electric Station, ECCS Evaluation Model Changes", June 20, 1991.
2. WPT-14479, J. L. Vota (W) to Mr. W. J. Cahill, Jr. (TU), "Comanche Peak Steam Electric Station Unit Number 1, NOTRUMP Small Break LOCA Analysis - Engineering Assessment in Support of Continued Operation", April 15, 1992.

WESTINGHOUSE LETTER, WPT-14387

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC STATION
UNIT 2
SMALL BREAK LOCA ECCS REANALYSIS

FEBRUARY 26, 1992

ENCLOSURE 2 TO TXX-92323
(TOTAL PAGES - 28)

Drat
DNB

WPT-14387

915.1

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230 0355

February 26, 1992
ET-NSL-OPL-II-92-102

Mr. W. J. Cahill, Jr., Executive Vice President
Nuclear Engineering & Operations
TU Electric Company
P. O. Box 1002
Glen Rose, Texas 76043

S.O. No. TCX-4708

Ref: 1) CPSES-9122081
2) WPT-14131
3) CPSES-9130542

Attention: A. Tajbakhsh

No Response Required

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC STATION
UNIT 2
SMALL BREAK LOCA ECCS REANALYSIS

Dear Mr. Cahill:

In response to your request of Reference 1, and in accordance with our commitment of Reference 2, please find attached the Comanche Peak Unit 2 Small Break LOCA ECCS reanalysis.

The Small Break LOCA ECCS analysis for Unit 2 was performed at a core power level of 3411 Mwt. Other pertinent analysis assumptions include 5 percent steam generator tube plugging level, 17x17 Optimized Fuel Assembly (OFA) fuel design, and 275 psig fuel rod helium backfill pressure. The analysis was performed with the NRC-approved Westinghouse ECCS Small Break Evaluation Model which utilizes NOTRUMP and is described in WCAP-10081-A.

Some of the assumptions used in the analysis were provided by TU Electric in the partially filled out Accident Analysis Checklist (ACC) per above Reference 3. TU Electric should assure that these assumptions remain valid.

The Unit 2 Small Break LOCA section FSAR updates are provided in Attachment A which include the results of the 2-, 3-, and 4-inch break analysis. The 3-inch is the worst case break resulting in a Peak Clad Temperature (PCT) of 1434°F. These results demonstrate conformance with 10 CFR 50.46 requirements for Small Break LOCA ECCS Analysis for Comanche Peak Unit 2.

This closes open item 10179-3.

If there are any questions on the above, please contact Mr. Roy Owoc at 412/374-4037.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

C. Bentonlo

J. L. Vota, Manager
Comanche Peak Projects

RHO/lg

Attachment

| | | |
|---|---|---|
| cc: | W. J. Cahill, Jr. | 1L, 1A |
| | CCS | 1L, 1A, 1AR |
| | S. C. Wood | 1L, 1A |
| | VETIP Coordinator | 1L, 1A |
| | L. Terry | 1L, 1A |
| | J. B. Roberts | 1L, 1A |
| | T. A. Hope | 1L, 1A |
| | W. G. Guldemond | 1L, 1A |
| | W. Choe | 1L, 1A |
| | A. Tajbakhsh | 1L, 1A |
| | D. Bize | 1L, 1A |

ATTACHMENT A

Comanche Peak Steam Electric Station Unit No. 2

FSAR Updates for Small Break LOCA

## 15.6.5.3 CORE AND SYSTEM PERFORMANCE

### 15.6.5.3.1 MATHEMATICAL MODEL

Small Break LOCA Evaluation Model

The Comanche Peak Steam Electric Station Unit No. 2 small break LOCA analysis was performed using the Westinghouse ECCS Small Break Evaluation model[14] which utilizes the NOTRUMP[12,13] and LOCTA-IV[8] computer codes. These computer codes are used to perform the analysis of Loss-Of-Coolant Accidents due to small breaks in the Reactor Coolant System (RCS). The NOTRUMP computer code, approved for this use by the Nuclear Regulatory Commission (NRC), is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of the flow through the reactor core and break. This code is a state-of-the-art one-dimensional general network code incorporating a number of advanced features. Among these new features are the utilization of nonequilibrium thermal calculation in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stack fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

In NOTRUMP[12,13], the RCS is subdivided into fluid filled control volumes (fluid nodes) and metal nodes interconnected by flowpaths and heat transfer links. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied to these nodes. The broken loop is modeled explicitly, and the intact loops are lumped into a second loop. A detailed description of the NOTRUMP code is provided in References 12 and 13.

In the NOTRUMP model[14], the reactor core is represented as a vertical stack of heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables the explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a Loss-of-coolant accident.

Clad thermal analysis are performed with the LOCTA-IV, Reference 8, computer code which uses as input the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input. For all computations, the NOTRUMP and LOCTA-IV calculations were terminated slightly after the time the core mixture level returned to the top of the core following core uncovery.

A schematic representation of the computer code interfaces is given in Figures 15.6-5 and 15.6-6.

15.6.5.3.3 RESULTS

Small Break Results

As noted previously, the calculated peak clad temperature resulting from a small break LOCA is less than calculated for a large break. A range of small break analyses are presented which establishes the limiting break size as 3 inches. The results of these analyses are summarized in Tables 15.6-1 and 15.6-7.

Figures 15.6-34 through 15.6-47 present the principal parameters of interest for the small break ECCS analyses. For all cases analyzed the following transient parameters are presented:

a. RCS pressure. (Figure 15.6-34, 15.6-41, 15.6-42)

b. Core mixture height. (Figure 15.6-35, 15.6-43, 15.6-44)

c. Hot spot clad temperature. (Figure 15.6-36, 15.6-45, 15.6-46)

d. Core Power after trip. (Figure 15.6-37)

e. Pumped safety injection. (Figure 15.6-47)

For the limiting 3 inch break, the following additional transient parameters are presented.

a. Core steam flow rate. (Figure 15.6-38)

b. Core heat transfer coefficient. (Figure 15.6-39)

c. Hot spot fluid temperature. (Figure 15.6-40)

Peak clad temperature for the limiting break (3-inch) was 1433.8°F. The maximum local zirconium oxidation was 0.60% and the core wide oxidation was less than the 1% criteria. These results indicate that a coolable geometry was maintained for small break LOCAs and therefore, long-term core cooling is assured by continued operation of the ECCS. These results are well below all Acceptance Criteria limits of 10CFR50.46 and in all cases are not limiting when compared to the results presented for large breaks.

TABLE 15.6-1
(sheet 3 of 4)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

| Accident | Event | Time (sec) |
|---|---|---|
| 3. DECLG $C_D$ = 0.4 (Min SI) | Start | 0.0 |
| | Reactor trip signal | 0.53 |
| | Safety injection signal | 1.62 |
| | Accumulator injection begins | 19.6 |
| | End-of-bypass | 35.73 |
| | End-of-blowdown | 35.73 |
| | Pump injection begins | 26.62 |
| | Bottom of core recovery | 48.25 |
| | Accumulator empty | 54.08 |
| Small break LOCA | | |
| 1. 2 inch | Start | 0.0 |
| | Reactor trip signal | 62.9 |
| | Safety injection signal | 73.9 |
| | Top of core uncovered | 2381.2 |
| | Accumulator injection begins | N/A |
| | Peak clad temperature occurs | 4062.6 |
| | Top of core covered | 5512.5 |
| 2. 3 inch | Start | 0.0 |
| | Reactor trip signal | 21.6 |
| | Safety injection signal | 31.6 |
| | Top of core uncovered | 990.5 |
| | Accumulator injection begins | 1999.8 |
| | Peak clad temperature occurs | 1841.8 |
| | Top of core covered | 3263.9 |

TABLE 15.6-1
(sheet 4 of 4)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

| Accident | Event | Time (sec) |
| --- | --- | --- |
| 3. 4 inch | Start | 0.0 |
| | Reactor trip signal | 12.7 |
| | Safety injection signal | 21.6 |
| | Top of core uncovered | 623.5 |
| | Accumulator injection begins | 897.6 |
| | Peak clad temperature occurs | 948.0 |
| | Top of core covered | 1342.2 |

TABLE 15.6-5

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

| | | |
|---|---|---|
| Licensed core power[a], (MWt) | 3411 | |
| Peak linear power, includes 102 % factor (KW/ft) | 12.87 | |
| Total peaking factor, $F_Q$ | 2.32[b] | |
| Axial peaking factor, $F_Z$ | 1.497 | |
| Power shape | | |
| Large break | Chopped cosine | |
| Small break | See Figure 15.6-48 | |
| Fuel assembly array | Optimized 17x17 | |
| Accumulator water volume, nominal ($ft^3$/accum) | 850 | |
| Accumulator tank volume, nominal ($ft^3$/accum) | 1350 | |
| Accumulator gas pressure, minimum (psia) | 600 | |
| Safety injection pumped flow | See Figures 15.6-21 and 15.6-47 | |
| Containment parameters | See Sec 6.2 | |
| Initial loop flow (lb/sec) | 9868 | |
| | Large | Small |
| Vessel inlet temperature ($^oF$) | 558.3 | 564.1 |
| Vessel outlet temperature ($^oF$) | 618.7 | 623.3 |
| Average reactor coolant pressure (psia) | 2280 | 2280 |
| Steam pressure (psia) | 994.7 | 1000 |
| Steam generator tube plugging level (%) | 0 | 5 |

---

(a) Two percent is to be added to this power to account for calorimetric error.

(b) "Envelope" for small break.

TABLE 15.6-7

SMALL BREAK LOCA RESULTS FUEL CLADDING DATA

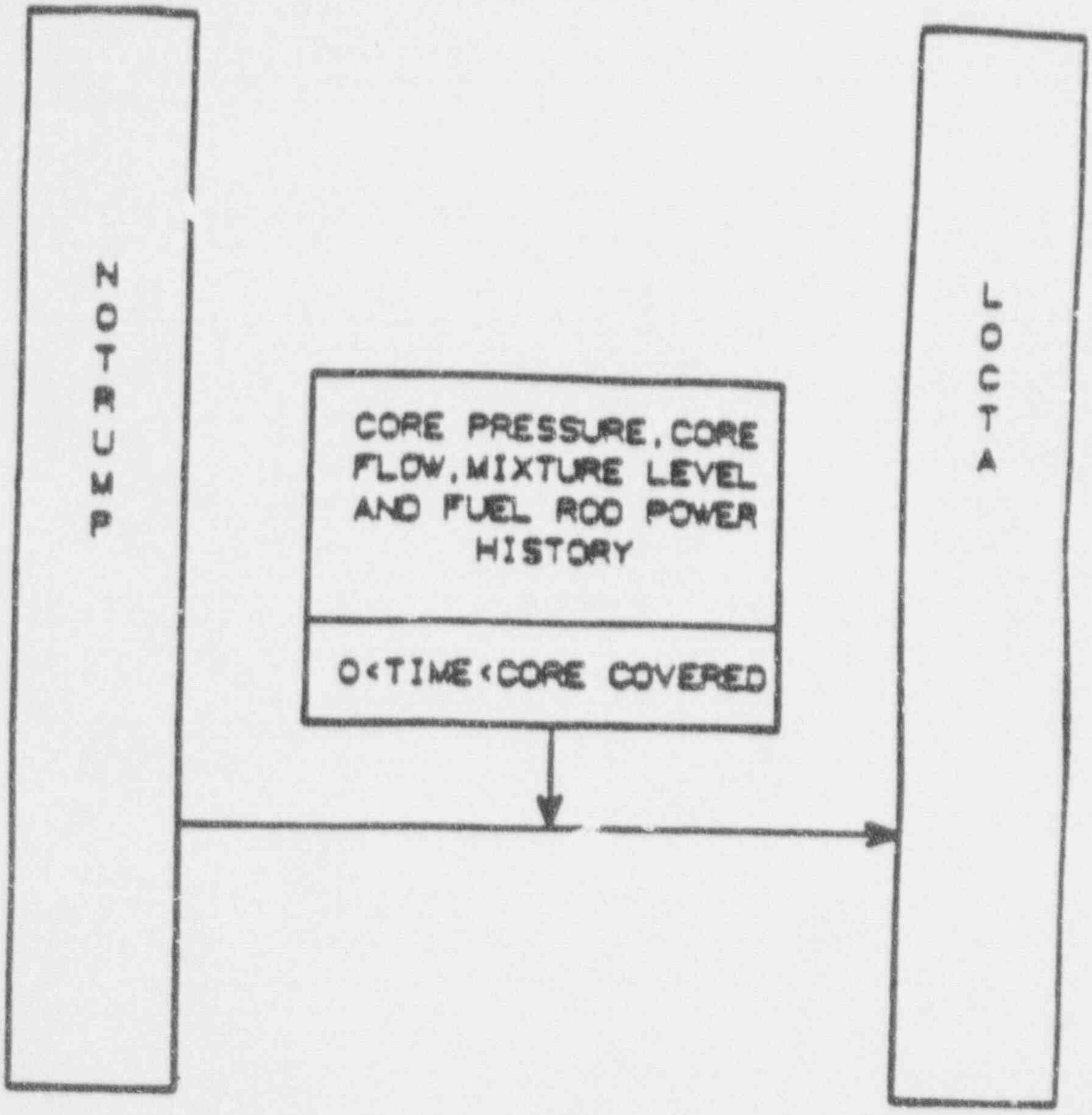
| Results | 2 inch | 3 inch | 4 inch |
|---|---|---|---|
| Peak clad temperature ($^{o}F$) | 1005.3 | 1433.8 | 1290.9 |
| Peak clad temperature location (ft) | 11.5 | 11.75 | 11.5 |
| Local $Zr/H_2O$ reaction, maximum (%) | 0.05 | 0.60 | 0.11 |
| Local $Zr/H_2O$ reaction location (ft) | 11.5 | 11.75 | 11.5 |
| Total $Zr/H_2O$ reaction (%) | <1.0 | <1.0 | <1.0 |
| Hot rod burst time (sec) | N/A | N/A | N/A |
| Hot rod burst location (ft) | N/A | N/A | N/A |

15.6.7 REFERENCES

12. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-100079-P-A (Proprietary), and WCAP-10080-P-A (Non-Proprietary), August 1985.

13. Rupprecht, S. D., et al, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," WCAP-11145-P-A (Proprietary), and WCAP-11373-A (Non-Proprietary), October 1986.

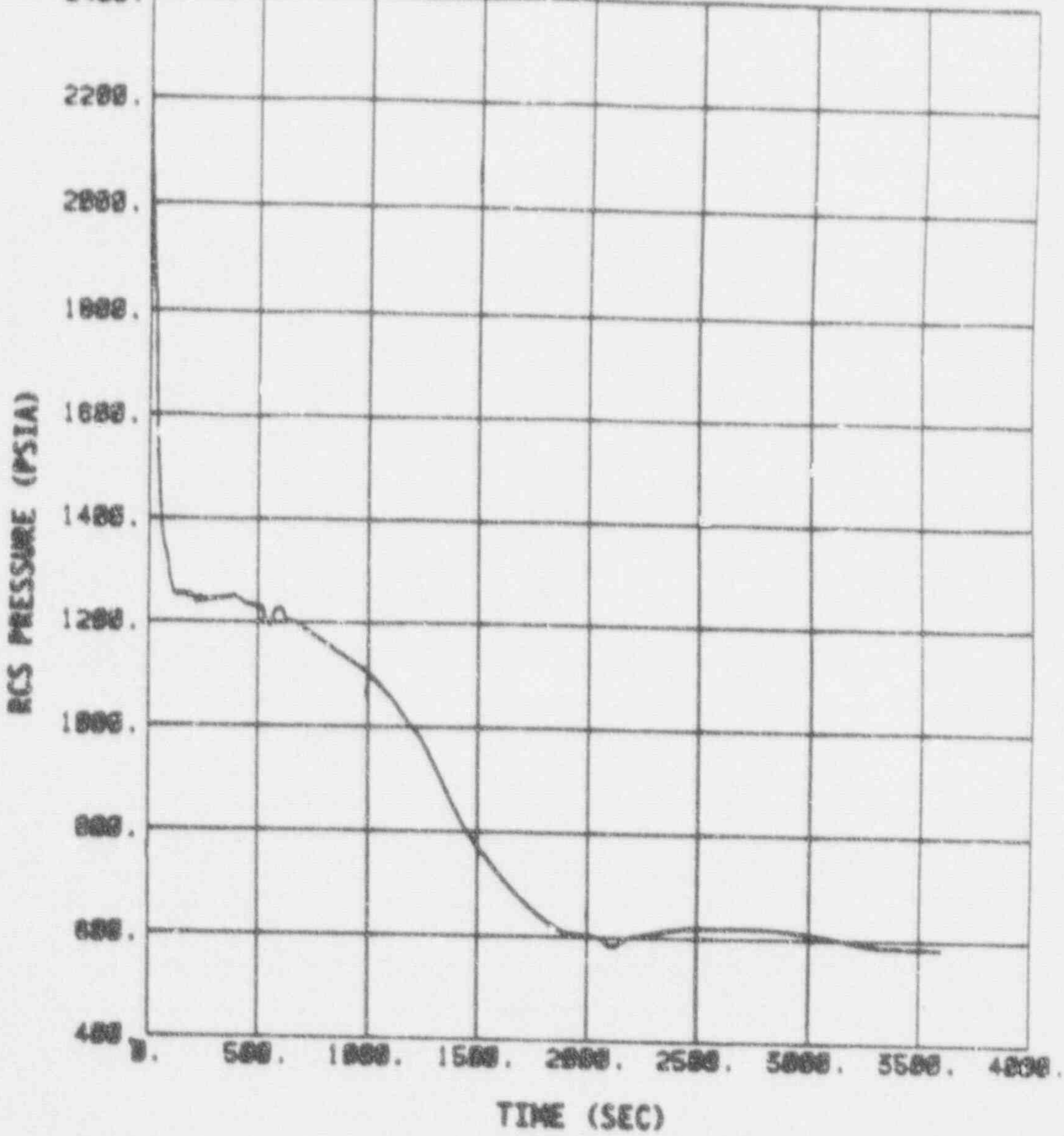
14. Lee, N., et al, "Westinghouse Small Break LOCA ECCS Evaluation Model using the NOTRUMP Code," WCAP-10054-P-A (Proprietary), and WCAP-10081-A (Non-Proprietary), August 1985.




**COMANCHE PEAK S.E.S.**
**FINAL SAFETY ANALYSIS REPORT**
**UNIT 2**

Code Interface Description for Small Break Model

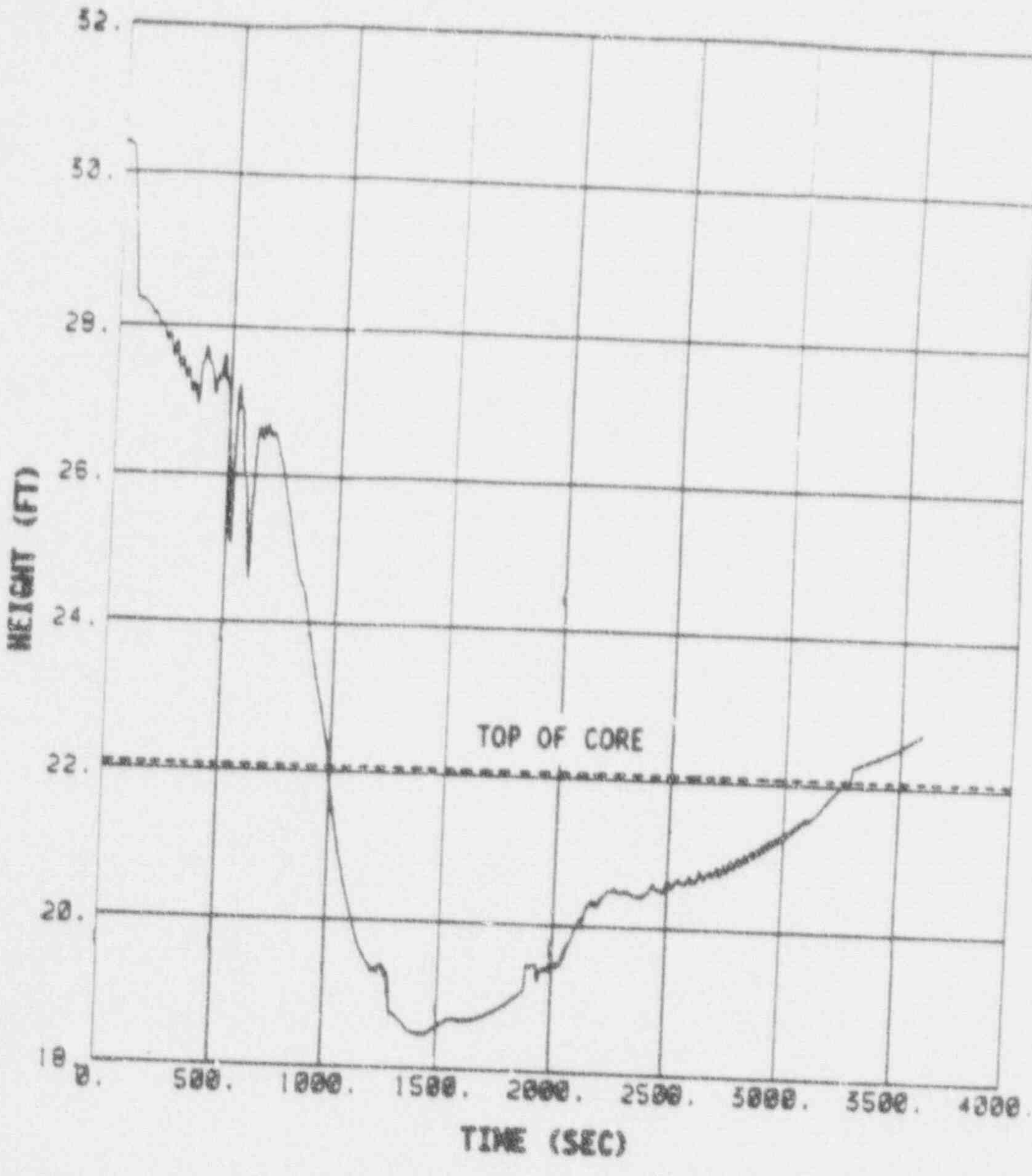
FIGURE 15.6-6




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

RCS Depressurization Transient
(3 Inch Break)

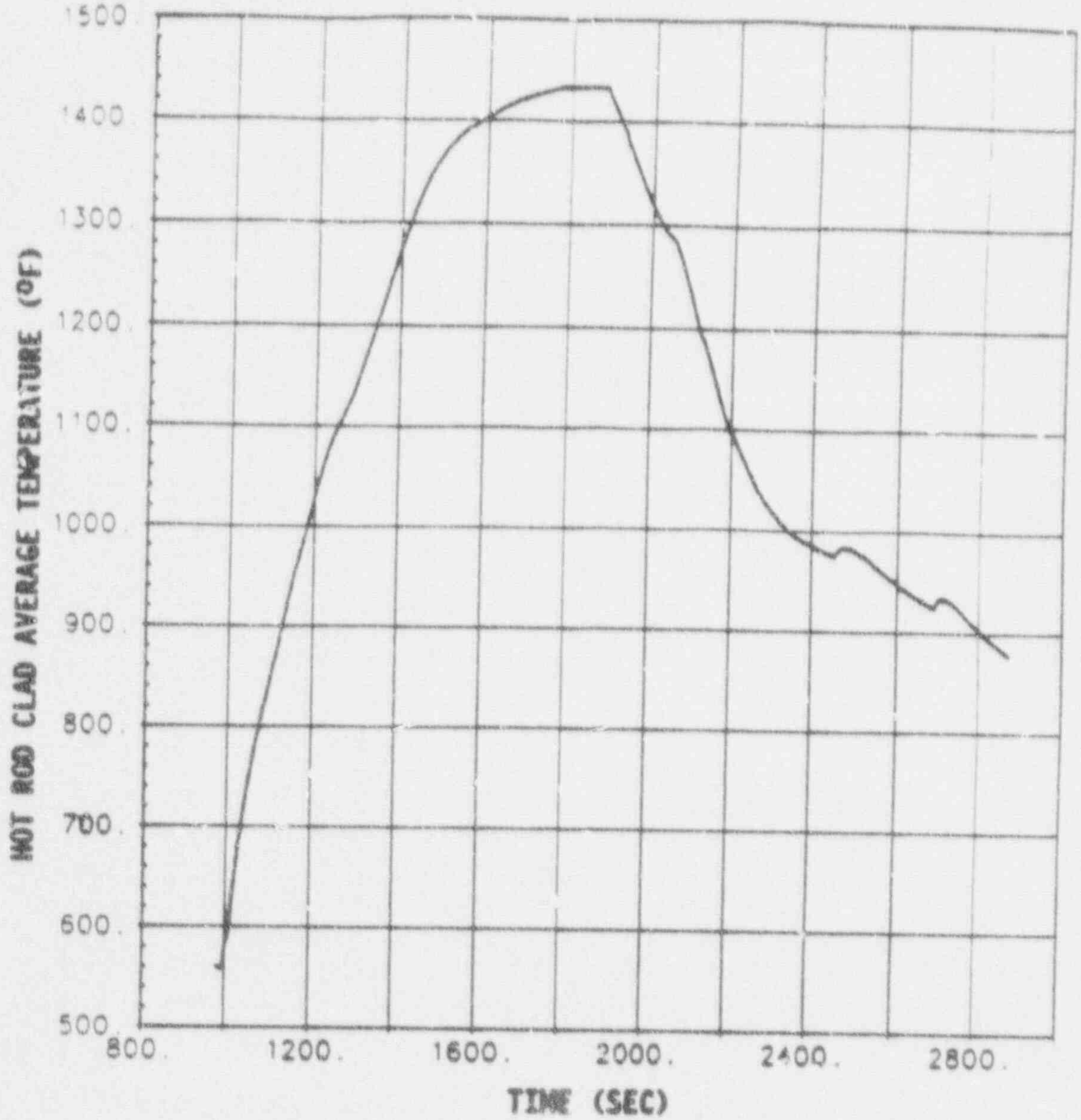
FIGURE 15.6-34




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Mixture Height
(3 Inch Break)

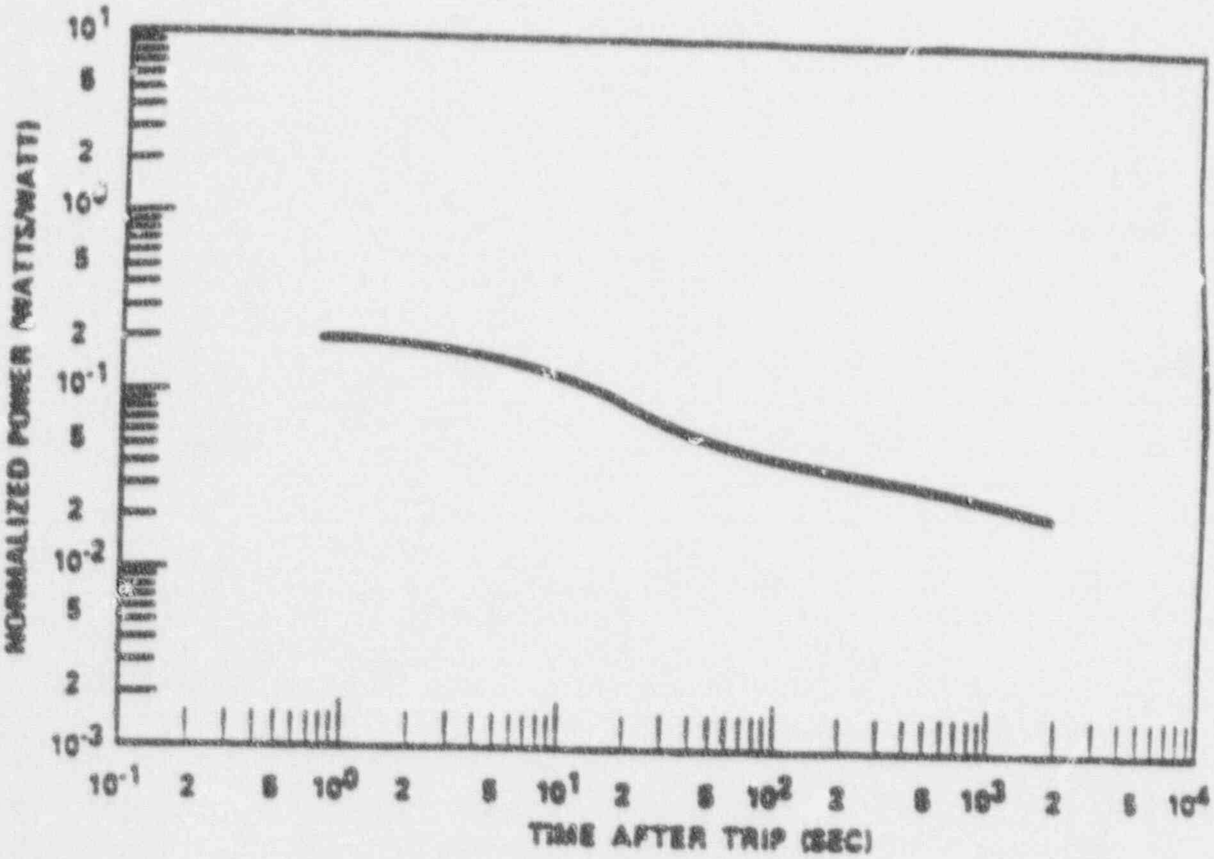
FIGURE 15.6-35




**COMANCHE PEAK S.E.S.**
**FINAL SAFETY ANALYSIS REPORT**
**UNIT 2**

Clad Temperature Transient
(3 Inch Break)

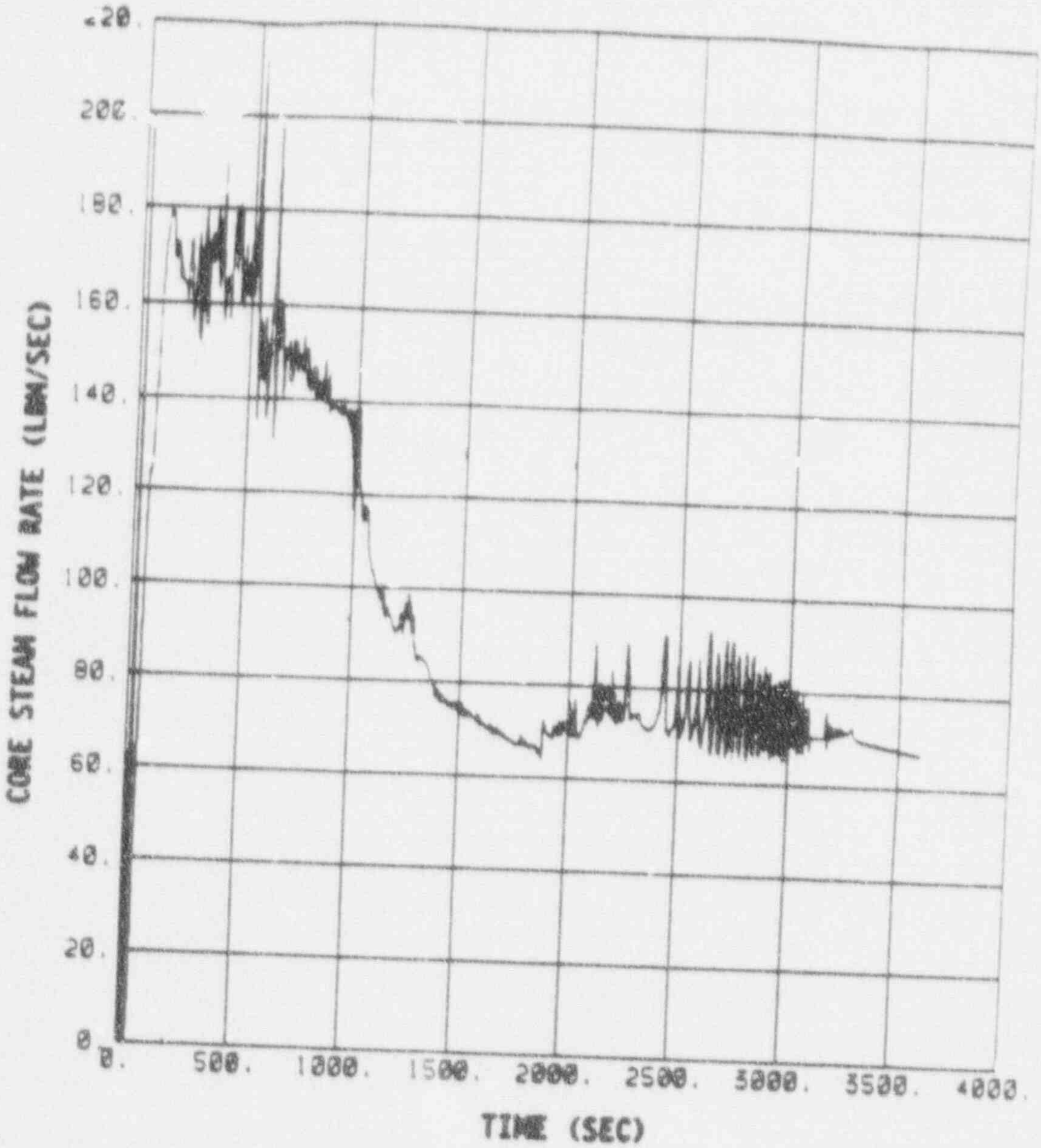
FIGURE 15.6-36



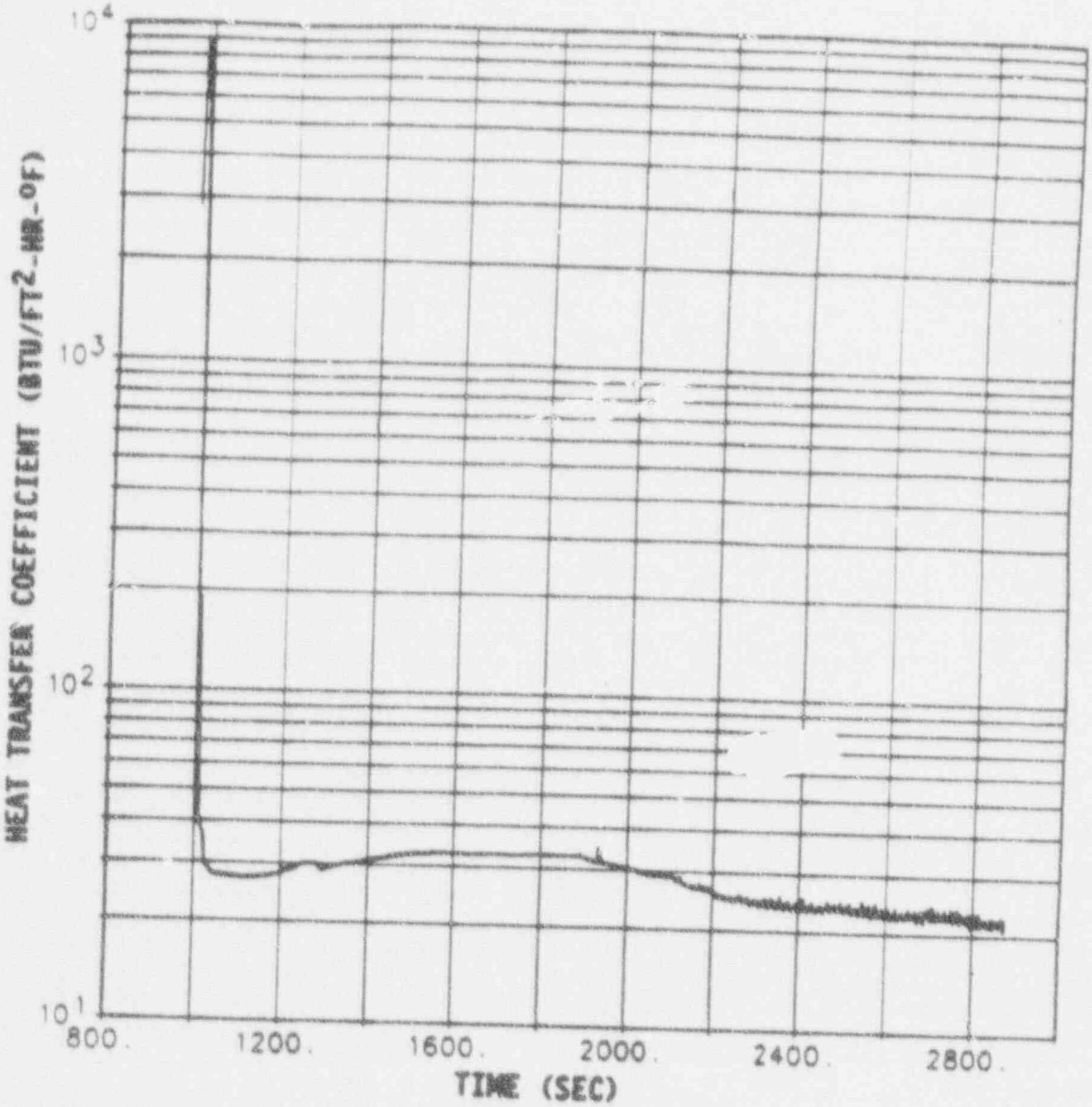

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Power After Reactor Trip

FIGURE 15.6-37



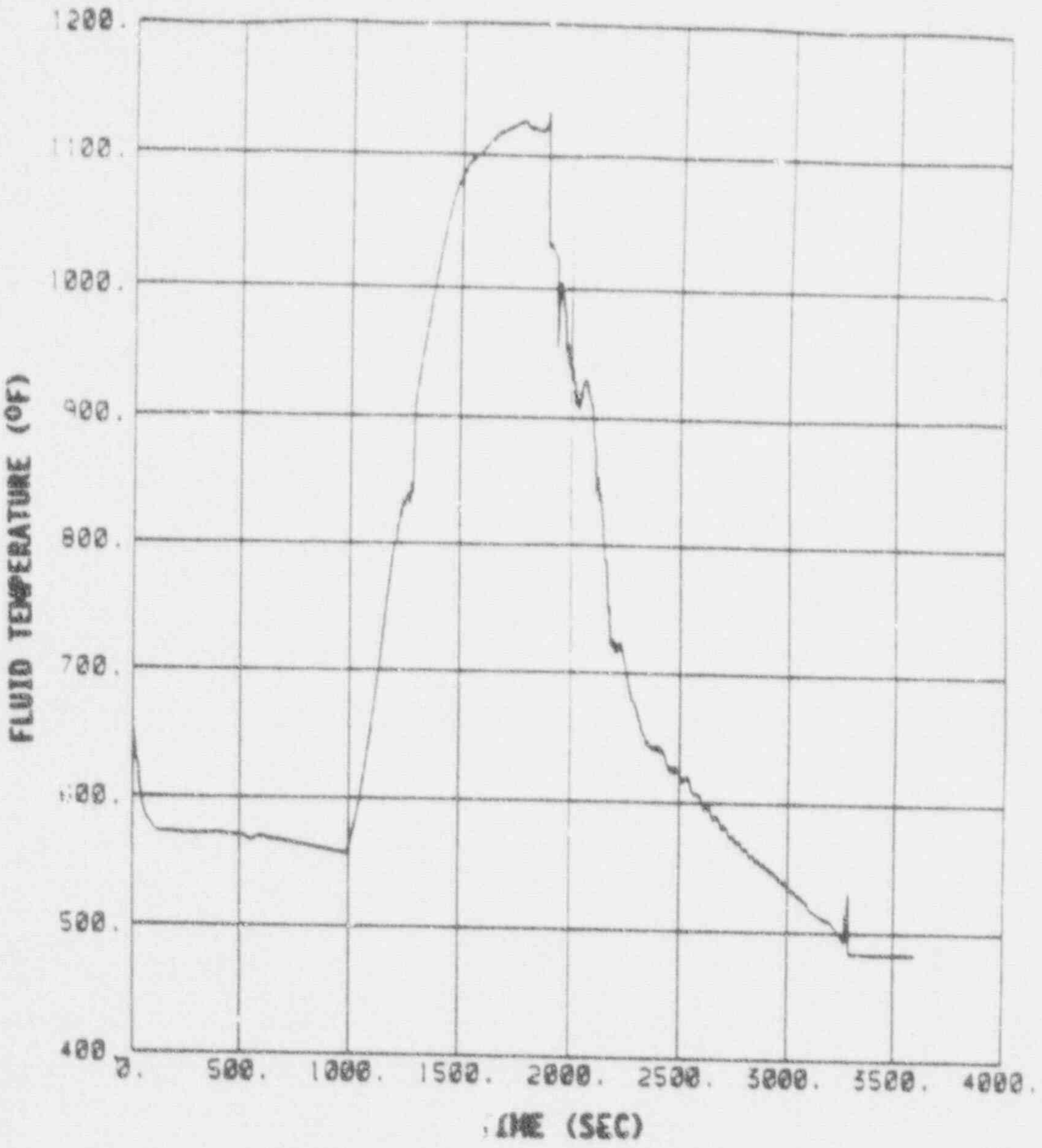

| COMANCHE PEAK S.E.S. FINAL SAFETY ANALYSIS REPORT UNIT 2 |
|---|
| Steam Flow (3 Inch Break) |
| FIGURE 15.6-38 |




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Rod Film Heat Transfer Coefficient
(3 Inch Break)

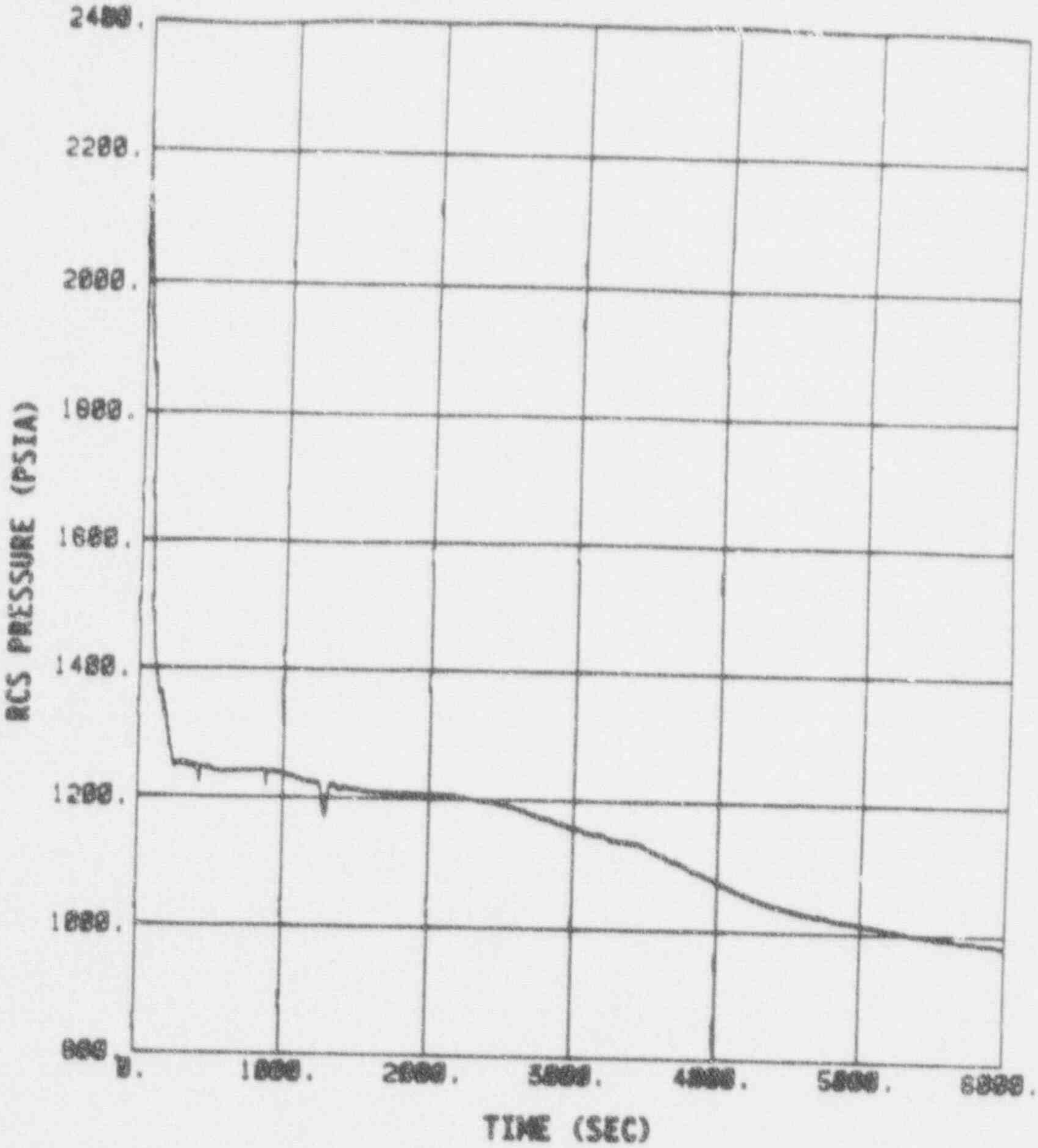
FIGURE 15.6-39




**COMANCHE PEAK S.E.S.**
**FINAL SAFETY ANALYSIS REPORT**
**UNIT 2**

Hot Spot Fluid Temperature
(3 Inch Break)

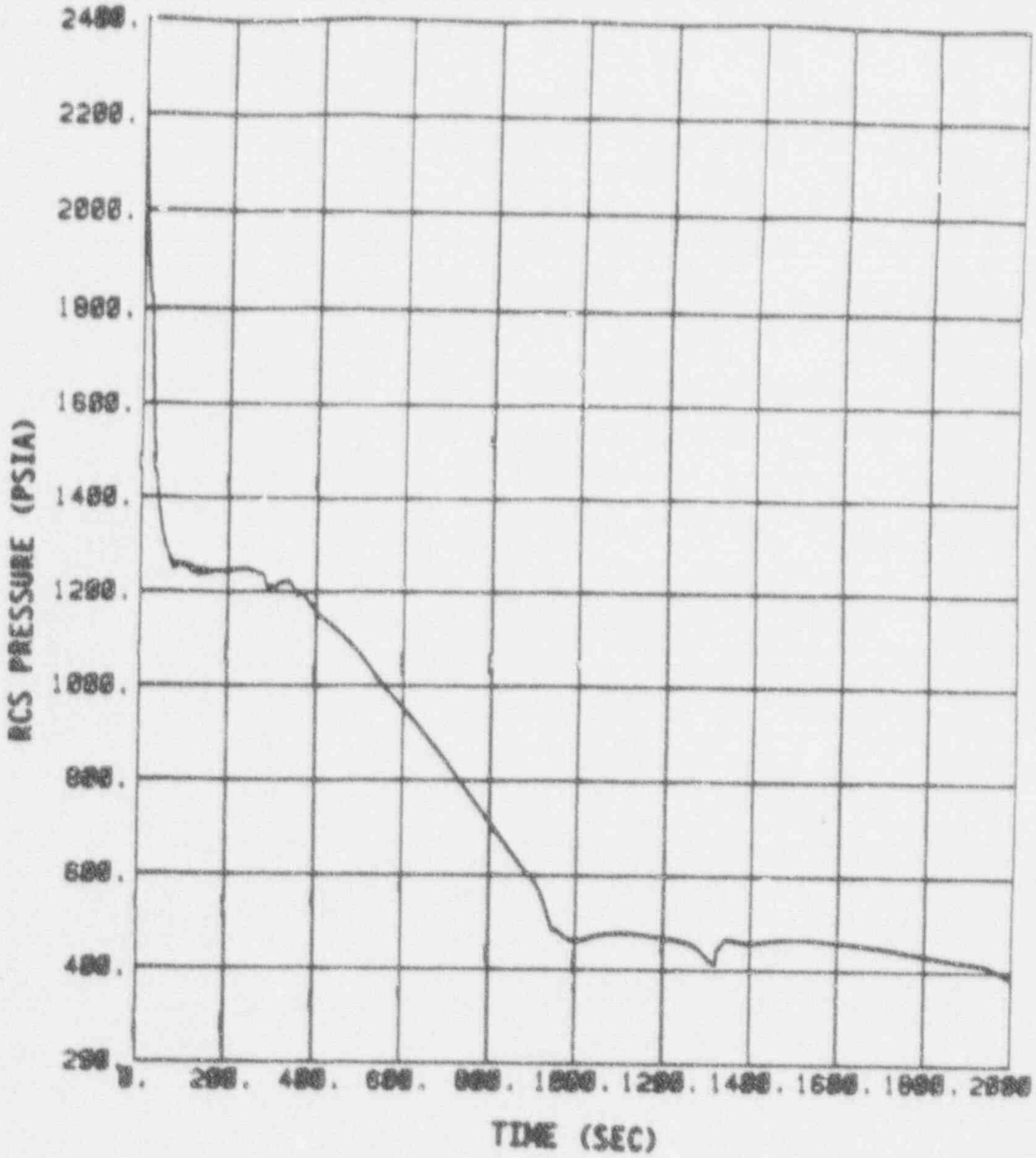
FIGURE 15.6-40




**COMANCHE PEAK S.E.S.**
**FINAL SAFETY ANALYSIS REPORT**
**UNIT 2**

RCS Depressureization Transient
(2 Inch Break)

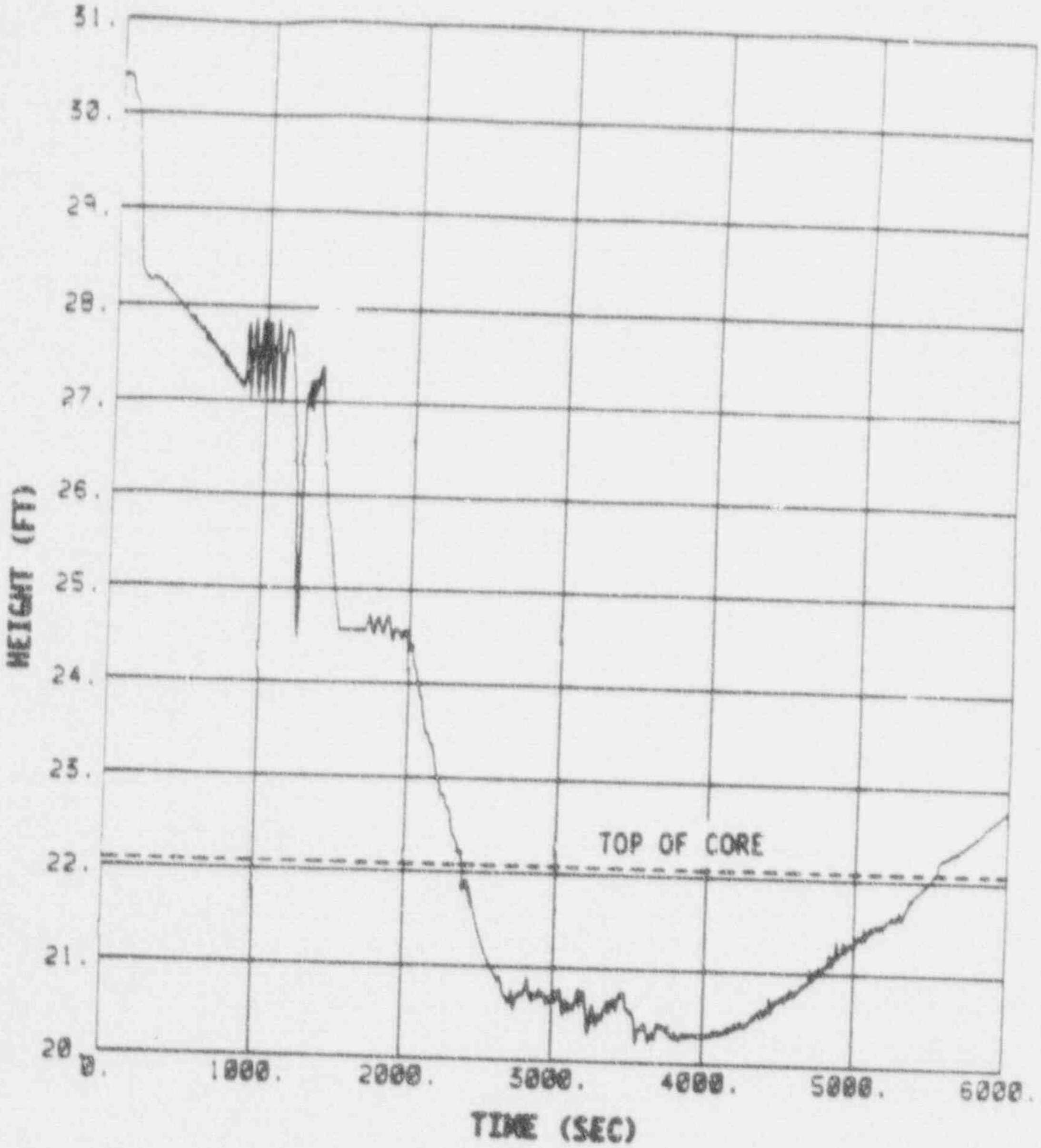
FIGURE 15.6-41




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

RCS Depressurization Transient
(4 Inch Break)

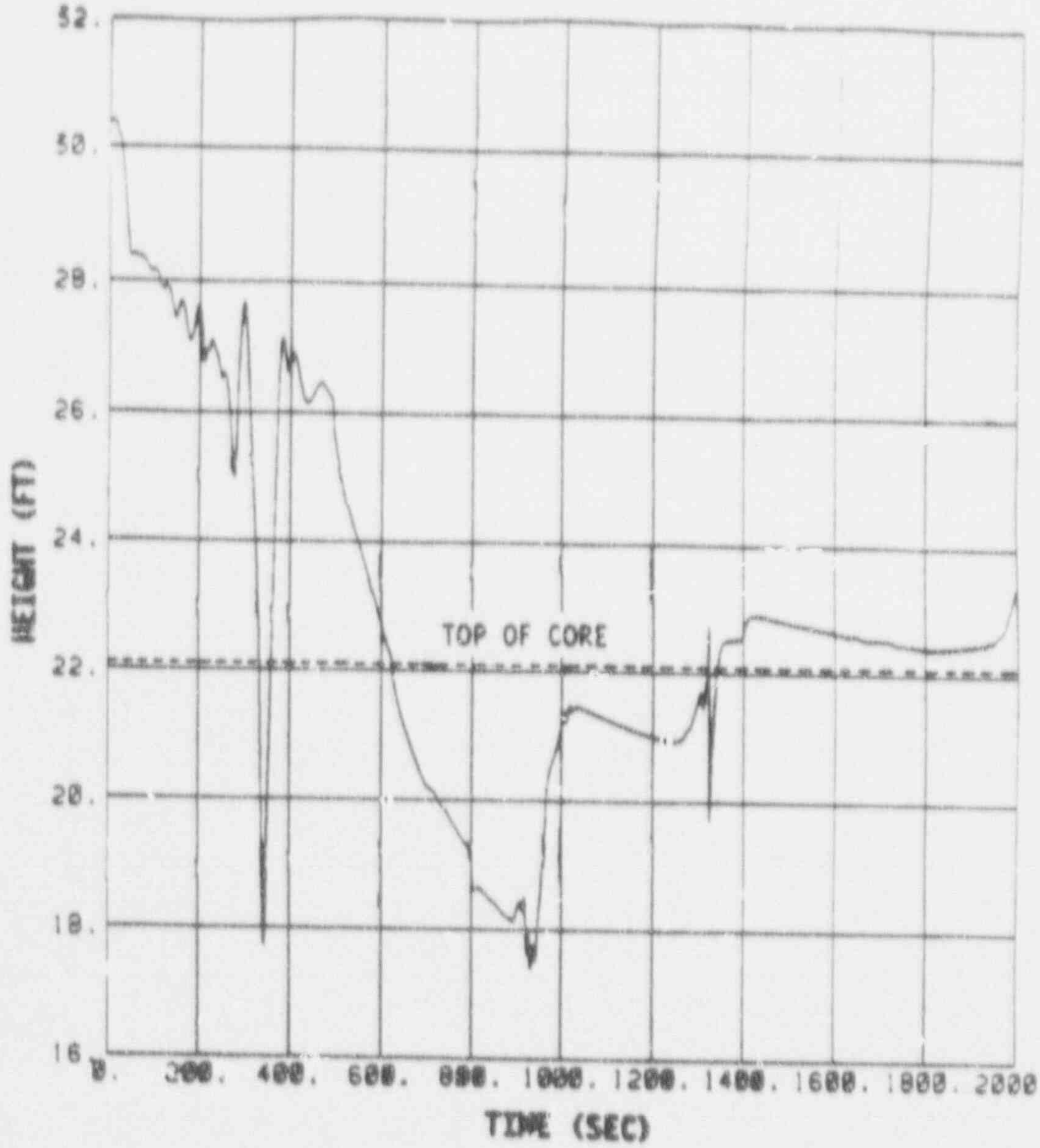
FIGURE 15.6-42




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Mixture Height
(2 Inch Break)

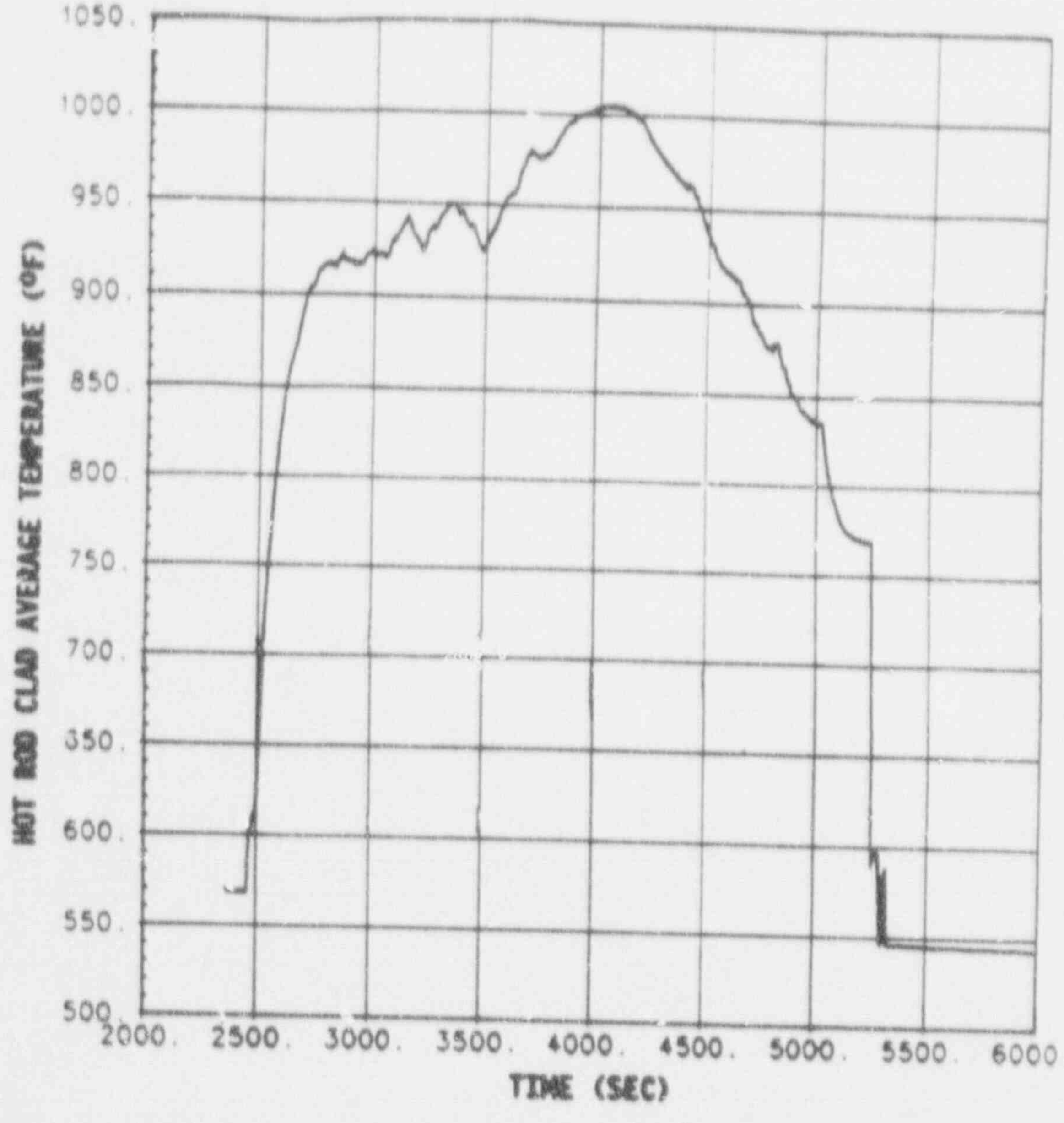
FIGURE 15.6-43



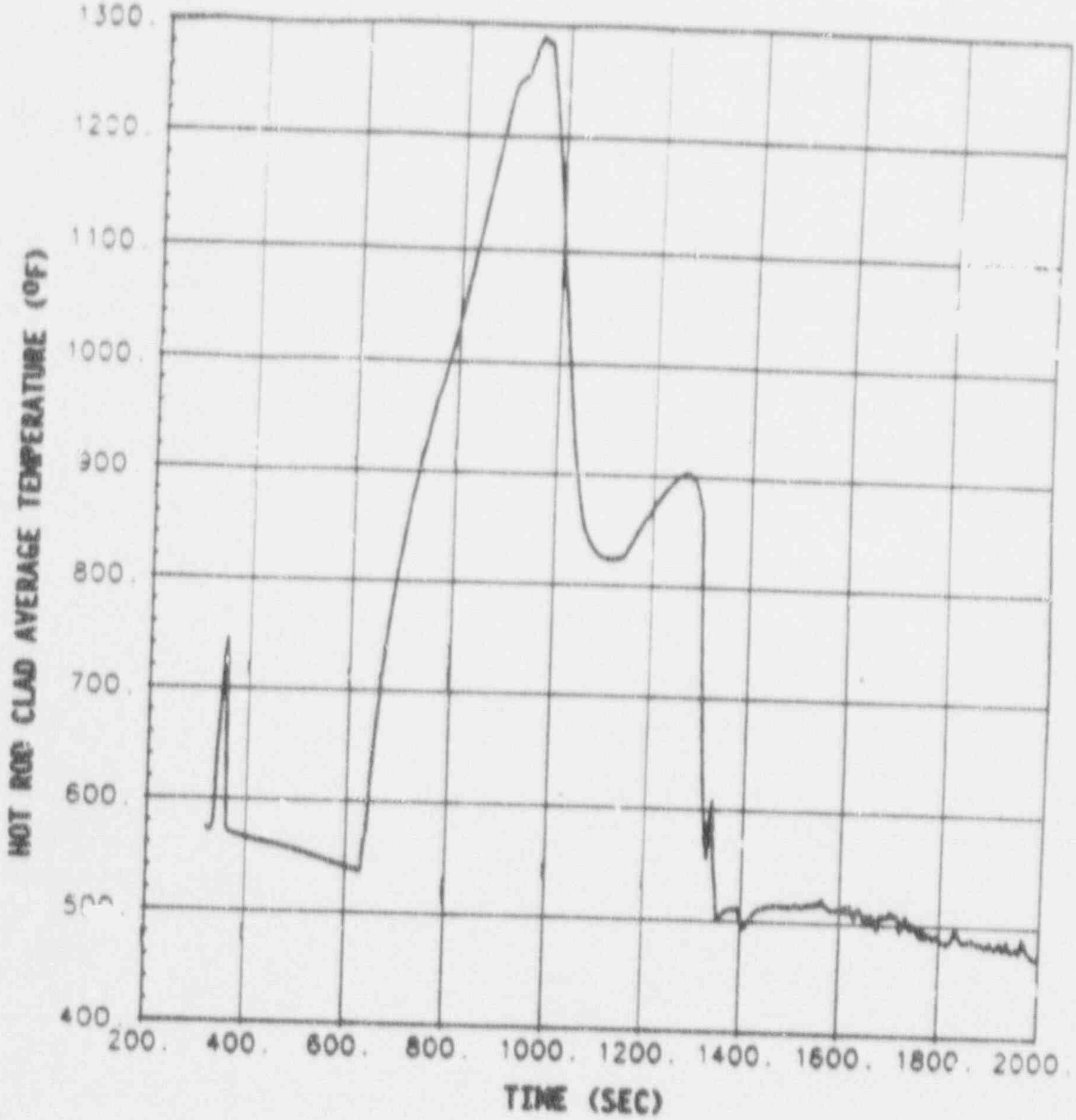

COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Mixture Height
(4 Inch Break)

FIGURE 15.6-44



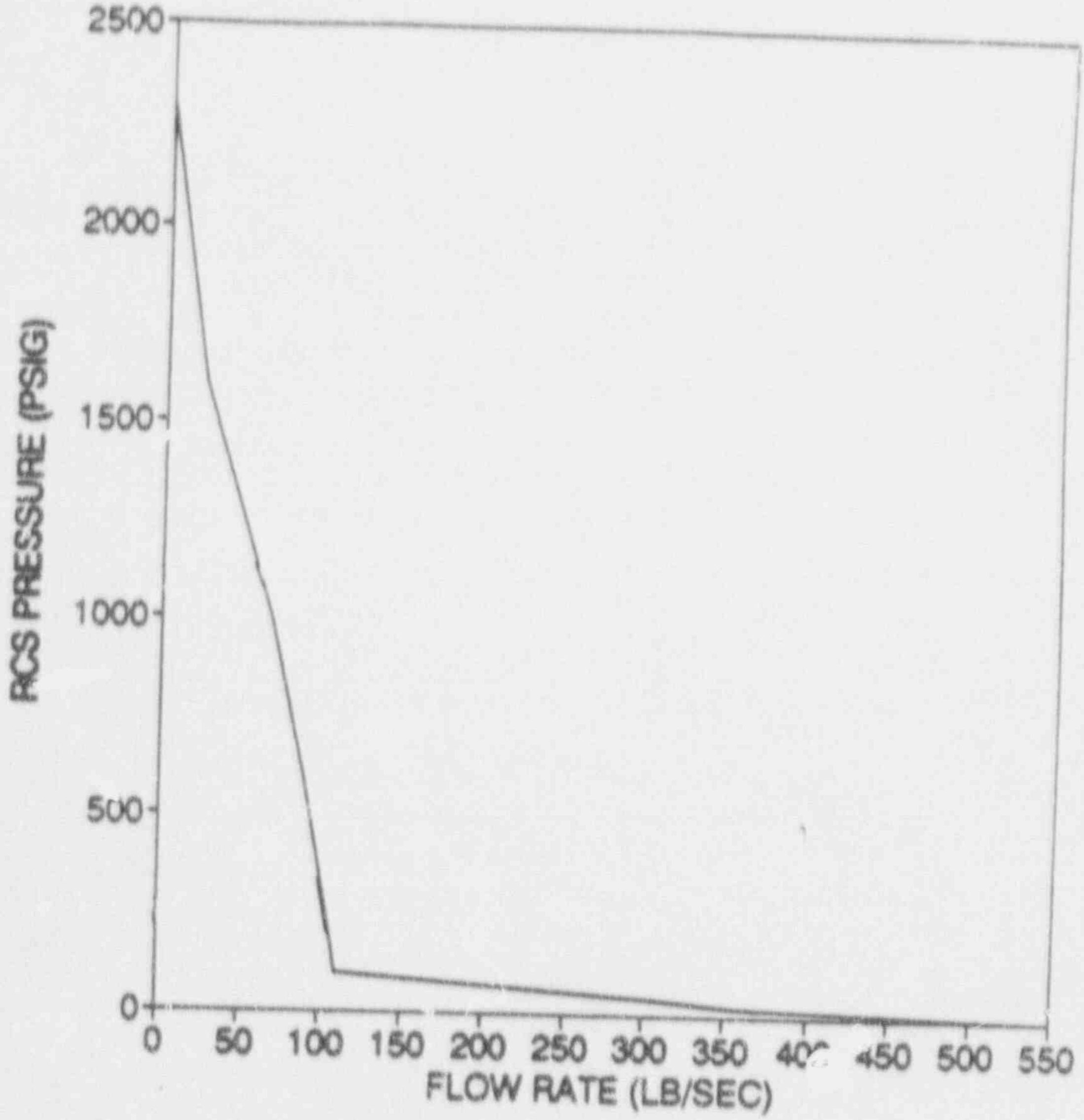

| COMANCHE PEAK S.E.S. FINAL SAFETY ANALYSIS REPORT UNIT 2 |
|---|
| Clad Temperature Trans' (2 Inch Break) |
| FIGURE 15.6-45 |




**COMANCHE PEAK S.E.S.**
**FINAL SAFETY ANALYSIS REPORT**
**UNIT 2**

Clad Temperature Transient
(4 Inch Break)

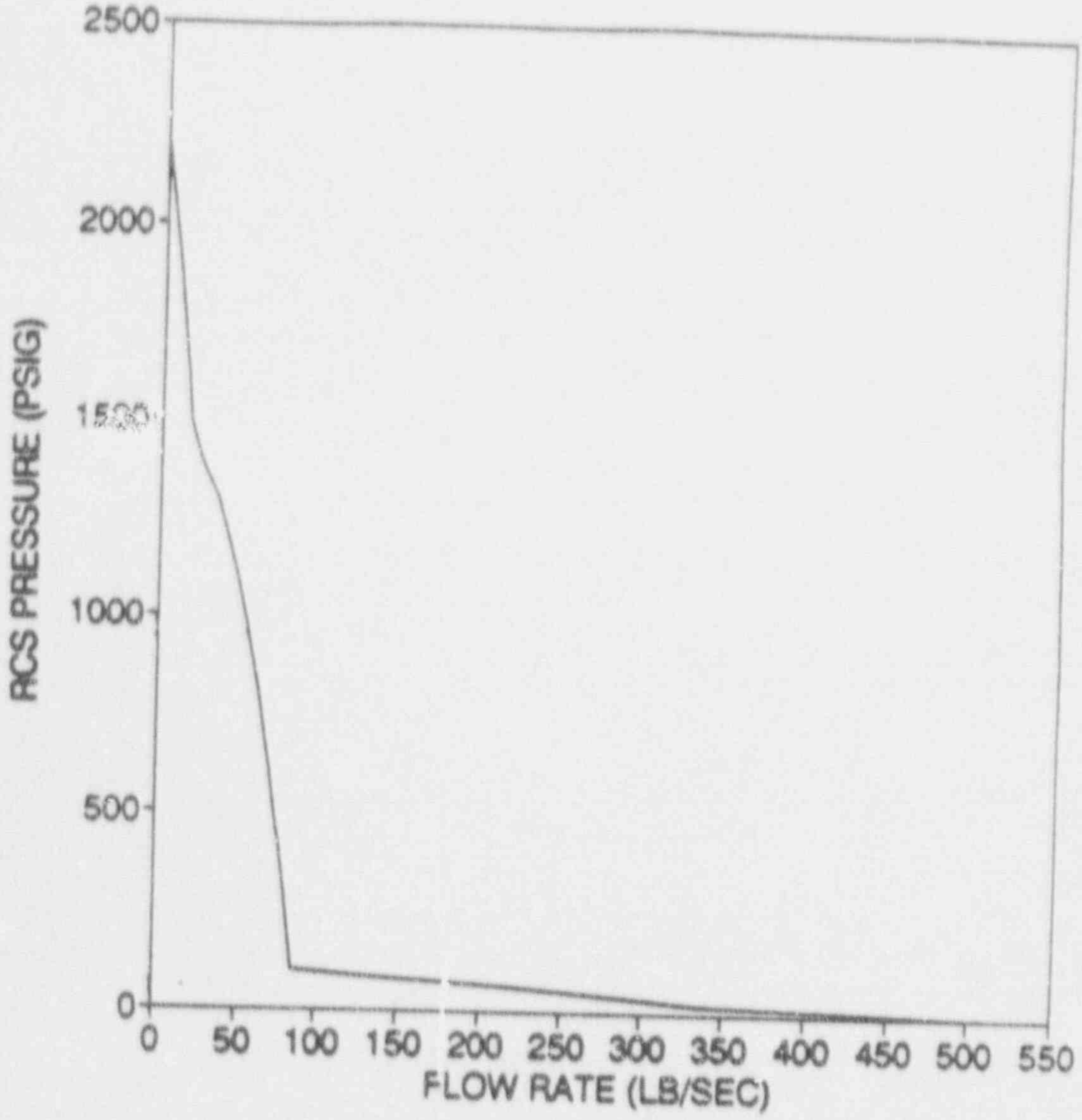
FIGURE 15.6-46




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Large Break
Safety Injection Flow Rate

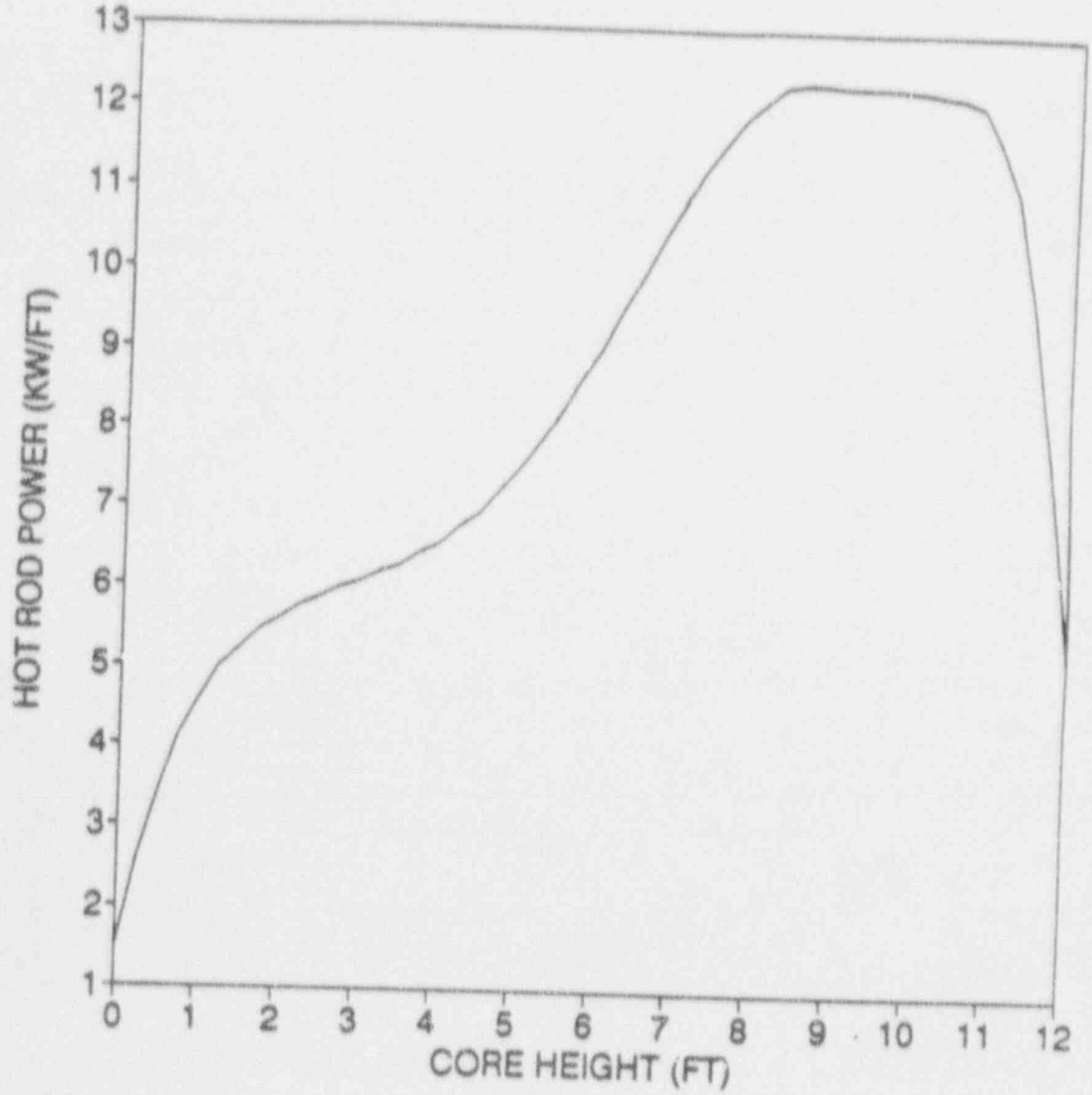
FIGURE 15.6-47a




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Small Break
Safety Injection Flow Rate

FIGURE 15.6-47b




COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Small Break
Power Distribution

FIGURE 15.[illegible]

WESTINGHOUSE LETTER, WPT-14479

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC STATION
UNIT NUMBER 1
NOTRUMP SMALL BREAK LOCA ANALYSIS - ENGINEERING ASSESSMENT
IN SUPPORT OF CONTINUED OPERATION

APRIL 15, 1992

ENCLOSURE 3 TO TXX-92323
(TOTAL PAGES = 12)

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WPT-14479
ET-NSL-OPL-II-92-185

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230 0355

April 15, 1992

Mr. W. J. Cahill, Jr., Executive Vice President
Nuclear Engineering & Operations
TU Electric Company
P. O. Box 1002
Glen Rose, Texas 76043

S.O. No. TBX-4708

Ref: 1. WPT-14387

Attention: W. Choe

(No Response Required)

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC STATION
UNIT NUMBER 1
NOTRUMP SMALL BREAK LOCA ANALYSIS - ENGINEERING ASSESSMENT
IN SUPPORT OF CONTINUED OPERATION

Dear Mr. Cahill:

As discussed with Mr. Whee Choe of TU Electric, a single small break LOCA analysis was performed for Comanche Peak Unit 1 using the NOTRUMP model. This analysis is based on the NOTRUMP analysis performed for Unit 2 and transmitted via Reference 1 above.

Attached please find an Engineering Assessment based on this Unit 1 analysis. TU Electric may use this assessment in support of a Justification for Continued Operation (JCO) of Comanche Peak. The need for a JCO could arise in the event the existing W-Flash analysis Peak Clad Temperature (PCT) exceeds the 2200°F acceptance criteria due to penalities associated with new safety issues and/or plant changes resulting in 10CF50.59 Safety Evaluations.

If there are any questions on the above or attached please contact Craig Thompson on 412/374-4409 or Roy Owoc on 412/374-4037.

This letter closes Westinghouse open item No. 10404-2.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

SHyde/for

R. H. Owoc

J. L. Vota, Manager
Comanche Peak Projects

# COMANCHE PEAK STEAM ELECTRIC STATION UNIT NO.1

## NOTRUMP SMALL BREAK LOCA ANALYSIS

## ENGINEERING ASSESSMENT IN SUPPORT OF CONTINUED OPERATION

### BACKGROUND

Westinghouse (Ref:1) transmitted the results of a 10CFR50.59 safety evaluation to remove the LOCA analysis credit for the turbine driven auxiliary feedwater pump from the current licensing basis small break LOCA analysis. This evaluation increased the small break LOCA PCT to 2133.65°F. Discussions with TU Electric regarding current Westinghouse open Potential Items (PIs), in particular the item on Small Break LOCA Burst and Blockage Consideration, reported to TU Electric in Reference 2, could, when fully resolved result in the current small break exceeding the 2200°F criteria. Westinghouse/TU Electric agreed to reanalyze CPSES-1 with the newer NOTRUMP evaluation model as a means to support continued operation. Westinghouse would provide TU Electric with an engineering assessment which TU Electric can use to support continued operation. Since application of the NOTRUMP small break methodology to CPSES-1 has not received NRC approval, application of the NOTRUMP methodology is considered outside the licensing basis for CPSES-1.

### Small Break LOCA Engineering Assessment

The small break LOCA analysis of record for Comanche Peak Unit 1 was performed using the WFLASH model (Ref:3). The limiting break size was a four inch diameter cold leg break which predicted a peak clad temperature of 1787.5°F. Safety evaluations that have been performed against this analysis are listed in Table 2. The cumulative result of these safety evaluations is a final PCT of 2133.65°F. This result when combined with currently open Potential Issues which affect small break LOCA analysis could result in a PCT above the 2200°F 10CFR50.46 Criteria. In order to have a basis for continued operation, in the event that unacceptable results would be obtained for the current W-FLASH analysis, the Comanche Peak Steam Electric Station Unit No.2 NOTRUMP small break LOCA analysis (Ref:4) was used as a basis to perform a CPSES-1 NOTRUMP small break LOCA analysis. The NOTRUMP (Ref: 5 & 6) small break LOCA code has received NRC approval for use in a licensing amendment in support of small break LOCA analyses performed under the requirements of Appendix K to 10 CFR part 50. The most limiting break identified in the Reference 4 analysis, the 3-inch cold leg break, was repeated by changing appropriate input to model the CPSES-1 core having 17X17 Standard Fuel (Fuel Rod O.D. of 0.374 inches) and changes necessary to model the CPSES-1 model D4 steam generator, since CPSES-2 has a model D5 design.

The results of the CPSES-1 NOTRUMP small break analysis were a PCT of 1418.4°F, and a local maximum zirconium water oxidation of 0.55%. These results are such that the additional 10CFR50.46 criteria for core wide oxidation, coolable geometry and long-term core cooling are not called into question.

The CPSES-1 current licensing basis analysis, using W-FLASH, had shown the 4-inch cold leg break to be limiting. However, the 3-inch cold leg break was analyzed for CPSES-1 using NOTRUMP since the Reference 4 analysis has shown this break to be more limiting for CPSES-2. Traditionally, analyses using the NOTRUMP code have shown smaller breaks to be more limiting when compared to W-Flash results for the same plant. Therefore a shift to a break smaller than the current WFLASH 4-inch break was not unexpected when CPSES-2 was analyzed. Further, based on the Reference 4 analysis, which has shown a greater difference in calculated PCT between the 2-inch and 4-inch breaks when compared to the 3-inch break, than calculated for the difference between the CPSES-1 & 2 cores, the CPSES-1 single break analysis (3-inch cold leg break) is justified and a spectrum of breaks is, in Westinghouse's judgement, not required in support of this engineering assessment of continued operation.

Since use of the NOTRUMP small break LOCA evaluation model has not been approved for use on CPSES-1, via the licensing amendment process, the above single break analysis for CPSES-1 is considered to be outside the licensing basis for CPSES-1.

<u>Large Break LOCA, LOCA Hydraulic Forcing Functions, Post-LOCA Subcriticality Requirement, and Switchover of the ECCS to hot leg recirculation to prevent potential boron precipitation.</u>

The remaining LOCA licensing requirements listed above are unaffected by changes in small break LOCA analysis, or choice of small break LOCA evaluation model. Therefore, an evaluation of these licensing requirements is not provided with this engineering assessment.

<u>Conclusion</u>

A new small break LOCA analysis, using the NOTRUMP small break evaluation model, has been performed for CPSES-1. This new analysis, using all changes previously evaluated under the provision of 10CFR50.59 calculated a low PCT showing a large amount of margin to the 10CFR50.46 requirement of 2200°F. This result indicates that NOTRUMP calculates improved core cooling when compared to the older W-FLASH model. Should the W-FLASH analysis be evaluated to have a PCT over the 2200°F criteria, the NOTRUMP result ameliorates any concern with regard to safe operation. In the event that the W-FLASH analysis for CPSES-1 is evaluated to have a PCT above 2200°F, the NOTRUMP analysis can be used as a basis for continued operation of CPSES-1.

## 6.0 REFERENCES

1) Letter WPT-14390, J. L. Vota (W) to W. J. Cahill, Jr. (TUE), "Comanche Peak Steam Electric Station Unit Number 1, Safety Evaluation to Remove LOCA Analysis Credit for the Turbine Driven Auxiliary Feedwater Pump", March 9, 1992.

2) Letter WPT-12933, J. L. Vota (W) to W. J. Cahill, Jr. (TUE), "Comanche Peak Steam Electric Station, ECCS Evaluation Model Changes", June 20, 1991

3) WCAP-8200 Rev.2 (PROPRIETARY) AND WCAP-8261 Rev.1 (NON-PROPRIETARY), "WFLASH - A Fortran-IV Computer Program for Simulation of Transients in a Multi-Loop PWR", June 1974

4) Letter WPT-14387, J. L. Vota (W) to W. J. Cahill, Jr. (TUE), "Comanche Peak Steam Electric Station Unit Number 2, Small Break LOCA ECCS Reanalysis", February 26, 1992

5) WCAP-100079-P-A (PROPRIETARY) and WCAP-10080-A (NON-PROPRIETARY) "NOTRUMP: A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE", AUGUST 1985.

6) WCAP-10054-P-A (PROPRIETARY), "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL USING THE NOTRUMP CODE", AUGUST 1985.

## TABLE 1

Safety Evaluations for the Comanche Peak Unit 1 Large Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 1. | 0.0°F | CWS-TBX-895 | Reduced SI flow would reduce spilling, with no impact on core or downcomer levels during reflood. |
| 2. | 0.0°F | SED-SA-296 | Bottom of core recovery delayed less than 0.02 sec. Later, downcomer filled slightly earlier due to higher flow. Supersedes evaluation number 1. |
| 3. | 6.2°F | SED-SA-340 | Modified steam generator bypass flow. Increase in initial core inlet temperature. |
| 4. | 0.0°F | SED-SA-774 | Revised SI flow tech. spec. Increased SI is a benefit since Comanche Peak 1 is not a max-SI plant. |
| 5. | 10.0°F | SED-SA-884 | Reduced accumulator water volume by 6 cubic feet. |
| 6. | 0.0°F | SED-SA-1048 | Reduced auxiliary feedwater flow. |
| 7. | 0.0°F | SECL-88-706 | Increased the signal processing delay time from 1 sec. to 2 sec. |
| 8. | 0.0°F | SECL-89-210 | Installed heated junction thermocouples and shrouds. |

TABLE 1 cont.

Safety Evaluations for the Comanche Peak Unit 1 Large Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 9. | 18.6°F | SECL-89-594 Rev 1 | Increase in S/G tube plugging. 2.1% area correction and 1% SGTP. |
| 10. | ......... | SECL-89-494 | Steam generator feedwater f[illegible] split. Same as evaluation 3. |
| 11. | 1.0°F | SECL-89-432 | Reduced RHR flow due to delay in isolating the miniflow lines. |
| 12. | 0.0°F | SECL-89-672 | Increased the main steam safety valve blowdown. |
| 13. | 0.0°F | SECL-89-1011 | Increased the upper nitrogen pressure limit for the accumulators. |
| 14. | 0.0°F | SECL-89-964 | Increased the AFW purge volume used to calculate the time to switchover to the lower enthalpy. |
| 15. | 0.0°F | WPT-11168 | Comanche Peak Steam Electric Station Setpoint Study Information. Pressurizer Low Pressure SI at 1700 psig and containment HI-1 at 5.0 psig. |
| 16. | 0.0°F | SECL-90-135 | Automatic AFW Controller Safety Evaluation. |

TABLE 1 cont.

Safety Evaluations for the Comanche Peak Unit 1 Large Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 17. | 0.0°F | SECL-90-195 | Revised Charging Flow Evaluation. |
| 18. | 0.0°F | SECL-90-215 | Reevaluation of the effect on small break LOCA for reductions in Charging SI and HHSI. This evaluation rescinds SECLs 90-135, 195 and SED-SA-296. |
| 19. | 0.0°F | SECL-90-293 | Increased AFW purge volumes due to check valve back leakage. |
| 20. | 12.0°F | | Thimble tube modeling penalty, NRC GENERIC LETTER 86-016. |
| 21. | 0.0°F | SECL-90-329 | Revised Auxiliary Feedwater purge volumes. |
| 22. | 0.0°F | SECL-90-352 | Increased Main Feedwater Isolation time. |
| 23. | 0.0°F | SECL-90-545 | Increased Auxiliary Feedwater flow from 625 gpm to 1225 gpm, entire purge volume assumed to be at 440°F. |
| 24. | 0.0°F | SECL-91-088D | Increased start time for the steam driven turbine auxiliary feedwater pump. The PCT change is based on an assumed total auxiliary feedwater flow rate of 1290 gpm compared to the SECL-90-545 assumption of 1225.5 gpm. |
| 25. | 7.2°F | WPT-13635 | Permanent changes to the ECCS evaluation model. |
| 26. | 0.0°F | SECL-91-367D | ECCS Flow changes to prevent runout of the Charging/SI and HHSI during post-LOCA recirculation. |

TABLE 1 cont.

Safety Evaluations for the Comanche Peak Unit 1 Large Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 27. | 0.0°F | SECL-92-090D | Removal of the credit for the TDAFW delivery from LOCA analysis. |
| 28. | 0.0°F | WPT-xxxxx | Engineering assessment for small break LOCA performed with NOTRUMP. No affect on large break LOCA. |
| | ------------ | -------------- | ------------------------------------ |
| | PCT Penalty | Reference | Evaluation Description |
| | 55.0°F | | Total PCT penalty for 10CFR50.59 changes and permanent ECCS model changes. |
| | 2010.7°F | | Limiting Case PCT |
| | ============ | ============== | ==================================== |
| | 2065.7°F | | Total Limiting Case PCT |

TABLE 2

Safety Evaluations for the Comanche Peak Unit 1 Small Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 1. | 0.0°F | CWS-TBX-895 | New data more conservative because more SI flow delivered before time of PCT. |
| 2. | 88.0°F | SED-SA-296 | 4.4% shortfall is SI flow delivered over time period of interest. Supersedes evaluation number 1. |
| 3. | 0.0°F | SED-SA-774 | Revised SI flow tech. spec. Increased SI is a benefit. |
| 4. | 11.0°F | SED-SA-1048 | Reduced auxiliary feedwater flow from 1410 to 1290 gpm. |
| 5. | 9.0°F | SECL-88-706 | Increased the signal processing delay time from 1 sec. to 2 sec. |
| 6. | 0.0°F | SECL-89-210 | Installed heated junction thermocouples and shrouds. |
| 7. | 0.0°F | SECL-89-594 Rev. 1 | Increase in S/G tube plugging. 2.1% area correction and 1% SGTP. |
| 8. | 0.0°F | SECL-89-494 | Steam generator feedwater flow split. |
| 9. | 0.0°F | SECL-89-432 | Reduced RHR flow due to delay in isolating the miniflow lines. |

TABLE 2 cont.

Safety Evaluations for the Comanche Peak Unit 1 Small Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 10. | 0.0°F | SECL-89-672 | Increased the main steam safety valve blowdown. |
| 11. | 0.0°F | SECL-89-1011 | Increased the upper nitrogen pressure limit for the accumulators. |
| 12. | 53.0°F | SECL-89-964 | Increased the AFW purge volume used to calculate the time to switchover to the lower enthalpy. |
| 13. | 2.0°F | WPT-11168 | Comanche Peak Steam Electric Station Setpoint Study Information. Pressurizer Low Pressure SI at 1700 psig. |
| 14. | 75.5°F | SECL-90-135 | Automatic AFW Controller Safety Evaluation. |
| 15. | 84.0°F | SECL-90-195 | Revised Charging Flow Evaluation. |
| 16. | 121.0°F<br>-88.0°F<br>-75.0°F<br>-84.0°F<br>—————<br>-126.0°F | SECL-90-215 | Reevaluation of the effect on small break LOCA for reductions in Charging SI and HHSI. This evaluation supersedes SECLs 90-135, 195 and SED-SA-296. |

TABLE 2 cont.

Safety Evaluations for the Comanche Peak Unit 1 Small Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 17. | 19.2°F | SECL-90-293 | Increased AFW purge volumes due to check valve back leakage. |
| 18. | 0.5°F<br>-11.0°F | SECL-90-329 | Revised AFW purge volumes. Supersedes evaluation performed in SED-SA-1048 (07/01/85). The 11°F penalty has been removed since the analysis value of 625 gpm is conservative when compared to the CPSES Unit No.1 Aux feed flow of 1290 gpm. |
| 19. | 0.0°F | SECL-90-352 | Increase in the Main Feedwater Isolation time. |
| 20. | -25.0°F | SECL-90-545 | Increase in the Auxiliary Feedwater flow rate for 625 gpm to 1225.5 gpm, entire purge volume assumed to be at 440°F. |
| 21. | 2.0°F | SECL-90-545 | Adjustment to the small break analysis results for the correction to the Zirc/Water error. |
| 22. | 0.0°F | SECL-91-088D | Increased start time for the steam driven turbine auxiliary feedwater pump. The PCT change is based on an assumed total auxiliary feedwater flow rate of 1290 gpm compared to the SECL-90-545 assumption of 1225.5 gpm. |

TABLE 2 cont.

Safety Evaluations for the Comanche Peak Unit 1 Small Break LOCA Analysis

| | PCT Penalty | Reference | Evaluation Description |
|---|---|---|---|
| 23. | 0.00°F | WPT-13635 | Permanent changes to the ECCS evaluation model. |
| 24. | 64.85°F | SECL-91-367D | ECCS Flow changes to prevent runout of the Charging/SI and HHSI during post-LOCA recirculation. |
| 25. | 99.10°F | SECL-92-090D | Removal of the credit for TDAFW delivery from LOCA analysis. |
| | ------------ | ------------ | ---------------------------------- |
| | 346.15°F | | Total PCT penalty for 10CFR50.59 changes and permanent ECCS model changes. |
| | 1787.5°F | | Limiting Case PCT |
| | ============ | ============ | ================================== |
| | 2133.65°F | | Total Limiting Case PCT |
| 26. | -715.25°F | WPT-XXXXX | Engineering assessment for CPSES-1 NOTRUMP small break LOCA analysis. This is a temporary use of PCT margin until the Engineering assessment can be replaced. |
| | ============ | ============ | ================================== |
| | 1418.40°F | | Total Limiting Case PCT using the NOTRUMP methodology. |