Copies for: Hazelton
C. Johnson
Klecker

A-11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 19 1979

MEMORANDUM FOR: V. Noonan, Chief, Engineering Branch, DOR

FROM: T. M. Novak, Chief, Reactor Systems Branch, DSS x27460

SUBJECT: INFORMATION MEMO - VESSEL INTEGRITY FOR SMALL LOCAs

Introduction:

Evidence has arisen recently which suggests that small LOCAs for PWRs may be more limiting with respect to vessel integrity at low temperatures than the normally assumed steam line break or large break LOCA events.

Problem:

An analysis of reactor vessel repressurization fracture mechanics performed by Westinghouse for Alabama Power's Sequoyah Units 1 and 2 revealed that the limiting events were small LOCAs, 4 and 6 square inches in area. Two dimensional flaw analyses of the pressure vessel indicated that following these small LOCAs, the faulted stress limits would be exceeded at 27 and 28 calendar years assuming a load factor of 0.8 (see Table 1). Previously, the staff had considered large break LOCA and steam line break events to be the most limiting events which challenge vessel integrity. For vessels with high copper concentration or marginal welds, a small LOCA calculation may prove to be more limiting than the previously analyzed events which are normally required by the staff.

In the vessel integrity analyses performed for Sequoyah by Westinghouse, material fracture properties were based on a copper content of 0.15 weight percent, a phosphorus content of 0.011 weight percent and an initial RT_{NDT} of 73°F obtained from vessel material certification for the pressure vessel. The fluence used in these analyses was supplied by Westinghouse for a four loop plant similar to Sequoyah. The applicant's acceptance criteria was that flaws less than 0.1156 a/t (1.0 inch) would be arrested within 75 percent of the vessel wall thickness. No credit was given for operator action prior to 10 minutes after the first alarm.

DSS Actions:

DSS intends to pursue this matter on a generic basis through Generic Task A-11, a study on the resistance of reactor vessel materials to brittle fracture.

Contact: Glenn Kelly, NRR
49-27591

FY '79 = 0.1 m-y

1765 117

8001070 255

We request that DOR supply us with a description of any action or actions they intend to pursue to resolve this issue for operating plants. This information will be included in a board notification being prepared by DSS related to vessel integrity.

Thomas M. Novak, Chief
Reactor Systems Branch
Division of Systems Safety

Enclosure:
As stated

cc: R. Mattson
V. Stello
R. Tedesco
T. Novak
S. Israel
G. Mazetis
G. Kelly

SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA AND LSB
VESSEL INTEGRITY ANALYSES
2 DIMENSIONAL FRACTURE
MECHANICS ANALYSIS
Table 1

Added by Amendment 58, December 22, 1978

CASE	PLANT LIFE (YEARS)		
	BS - LG CU = 0.15% P = 0.011% RT_{NDTI} = 40°F	BS - LG CU = 0.13% P = 0.015% RT_{NDTI} = 73°F	WD - CF CU = .38 (.33)% P = 0.021% RT_{NDTI} = -40°F
2 INCH SMALL LOCA	40	40	40
3 INCH SMALL LOCA	40	40	40
4 INCH SMALL LOCA	40	27	40
6 INCH SMALL LOCA	40	28	40
LSB WITH REACTOR COOLANT PUMPS RUNNING	40	40	40
LSB WITH REACTOR COOLANT PUMPS TRIPPED	40	40	40

BS - BASE MATERIAL
WD - WELD MATERIAL

LC - LONGITUDINAL FLAW
CF - CIRCUMFERENTIAL FLAW

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Sequoyah

6.54 During long term cooling following a steamline break, feedwater line break, or small LOCA, the operator must control primary system pressure to preclude overpressurizing the pressure vessel after it has been cooled off.

a. Describe the instructions given the operator to perform long term cooling.

b. Indicate and justify the time frame for performing the required action.

c. List the instrumentation and components needed to perform this action and confirm that these components meet safety grade standards.

d. Discuss the safety concerns during this period and the design margins available. This should include potential adverse hydraulic conditions leading to inadequate cooling or mechanical damage.

e. Provide temperature, pressure, and RCS inventory graphs that would show the important features during this period.

The above discussion should account for the following: 49

a. loss of offsite power

b. operator error or single failure

c. small LOCA's may occur in the cold leg or in the hot leg/ pressurizer.

d. small LOCA's may result in nitrogen blanketing of the steam generators.

e. long term cooling for a small LOCA may depend on alternating forced convection and vaporization depending on the break location and size.

Response:

The response to this question as submitted on the D.C. Cook Unit 2 docket is an appropriate approach to the generic issues which have been raised. See D.C. Cook FSAR amendment 78. 58

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To address the issue of reactor vessel repressurization a fracture mechanics study on the integrity of the Sequoyah Units 1 and 2 reactor vessel beltline under faulted conditions was performed. The faulted conditions evaluated were the large steamline break (LSB) and the small loss-of-coolant accident (LOCA). These analyses supplement previous studies done for normal, upset, emergency, and faulted conditions as described in FSAR section 5.2.

The LSB transients used for this analysis are generic transients for a UHI four loop plant that have been modified to approximate the impact of pressurizer thick metal heat of the Sequoyah Units. Two LSB transients were evaluated: a case which assumes a loss of offsite power that causes the main reactor coolant pumps to stop (pumps tripped case) and a case where offsite power remains available that allows the main reactor coolant pumps to continue operation (pumps running case). The RCS response for the LSB is shown in Figures Q6.54-1 through Q6.54-5. The transients used in the analysis of small LOCAs were taken from work for the British performed by Westinghouse in 1974. The RCS responses for the 2, 3, 4, and 6 inch diameter small LOCAs are given in Figures Q6.54-6 through Q6.54-13. These RCS responses were used to determine the temperature, thermal stress, and pressure stress profiles through the vessel wall in the beltline region as a function of time. these profiles were then used in performing the fracture mechanics analyses.

In these analyses the following material properties were used for a longitudinal flaw in the base material:

Copper Content = 0.15 weight percent
Phosphorus Content = 0.011 weight percent
Initial RT'T = 73°F

For a circumferential flaw, the following material properties of the core region circumferential weldment were used:

Copper Content = 0.33 weight percent
Phosphorus Content = 0.021 weight percent
Initial RT'T = -40°F

These properties were obtained from the vessel fabrication material test certification for the Sequoyah Units. The fluence used in these analyses was that calculated for a generic four loop vessel similar to the Sequoyah Units and satisfactorily approximates the fluence levels of these units.

The irradiation damage of the material is correlated by trend curves. These curves were developed by Westinghouse to relate the magnitude of the shift of RT'T to the amount of neutron fluence and are a function of copper content. The final RT'T values are then

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used to calculate the plane strain fracture toughness (K7@) and the reference fracture toughness (K7@) as a function of the fractional depth through the vessel wall. A two dimensional combined flaw analysis that is an approximation of a three dimensional flaw is used. The results of this fracture mechanics analysis are presented in Table Q6.54-1. These results are presented in terms of the maximum number of calendar years (.8 load factor is assumed) the plant will conform to the following criteria:

Minimum critical flaw is greater than 0.1156 a/t (1.0 inch) or flaw arrest is within 75 percent of the vessel wall thickness.

From Table Q6.54-1 it can be seen that for the two dimensional flaw method the vessel integrity can be shown for only about 30 years of plant operating life for two cases. All other cases indicated vessel integrity is assured for at least 40 years. For the 4 and 6 inch small LOCAs, the maximum number of calendar years the plant will conform to the vessel integrity criteria is 27 and 28 years respectively. TVA is reviewing plans to perform a 10 CFR 50 Appendix G analysis of the Sequoyah vessels prior to the one quarter service life surveillance. This more realistic analysis is fully expected to verify reactor vessel integrity for the full 40-year plant life. Based on the anticipated outcome of the upcoming Appendix G analysis and the analyses already performed showing vessel integiry for nearly 30 years, vessel integrity is assured with adequate margin for the first 10 years of its service life.

To provide guidance to the operator to be alert to the potential for vessel repressurization after an accident and also to be able to respond quickly, the plant operating procedures provide explicit instructions. The operator is instructed to be continuously aware of primary system pressure and temperature comparing them to 10 CFR 50 Appendix G pressure-temperature curves for Sequoyah which are provided in the Technical Instructions. The procedures also identify the qualified instruments necessary for this monitoring action.

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SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA AND LSB
VESSEL INTEGRITY ANALYSES
2 DIMENSIONAL FRACTURE
MECHANICS ANALYSIS
TABLE Q6.54-1

Added by Amendment 58, December 22, 1978

CASE	PLANT LIFE (YEARS)		
	BS - LG CU = 0.15% P = 0.011% RT_{NDTI} = 40°F	BS - LG CU = 0.13% P = 0.015% RT_{NDTI} = 73°F	WD - CF CU = .38 (.33)% P = 0.021% RT_{NDTI} = -40°F
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3 INCH SMALL LOCA	40	40	40
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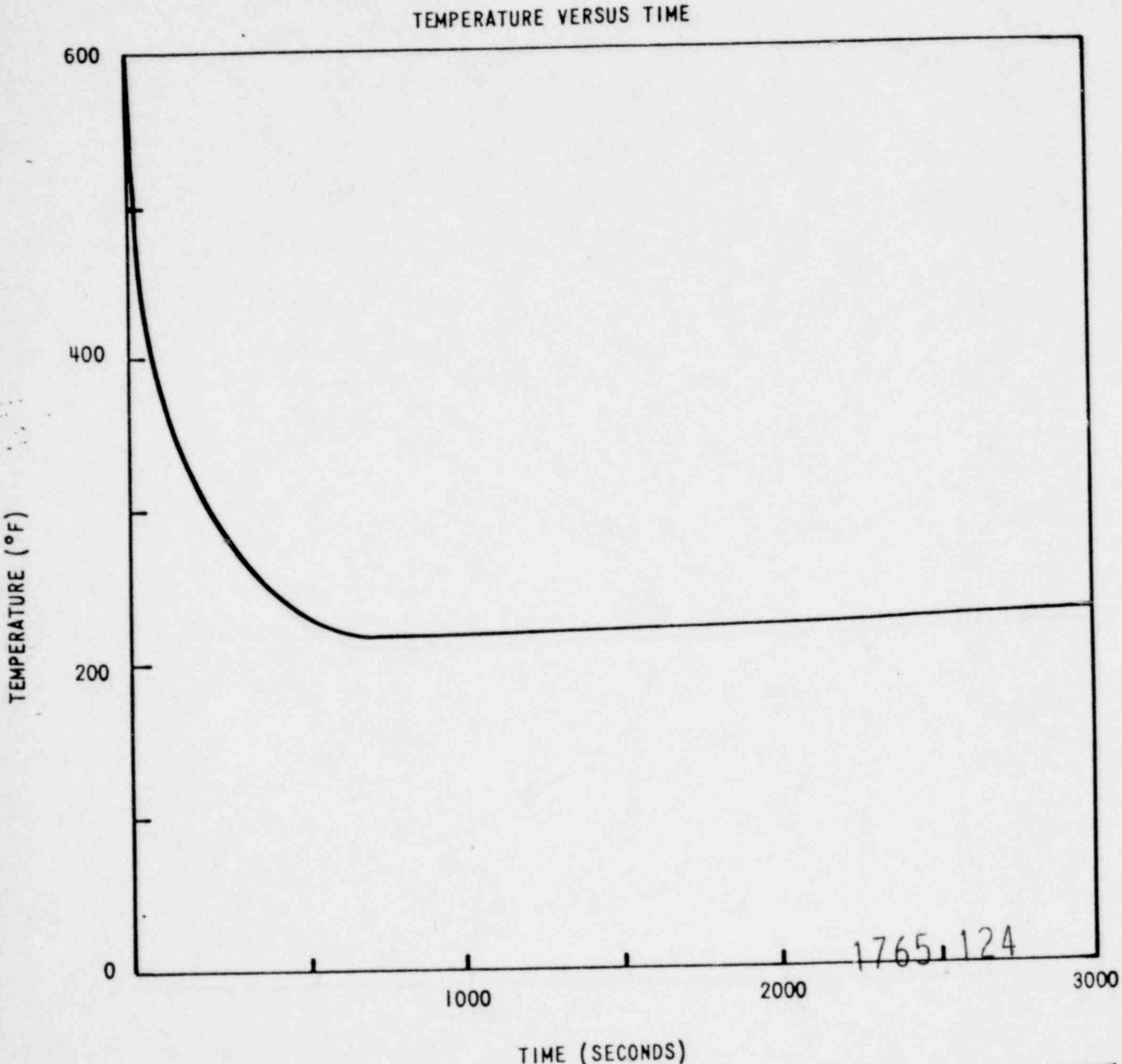
LSB WITH REACTOR COOLANT PUMPS RUNNING	40	40	40
LSB WITH REACTOR COOLANT PUMPS TRIPPED	40	40	40

BS - BASE MATERIAL
WD - WELD MATERIAL

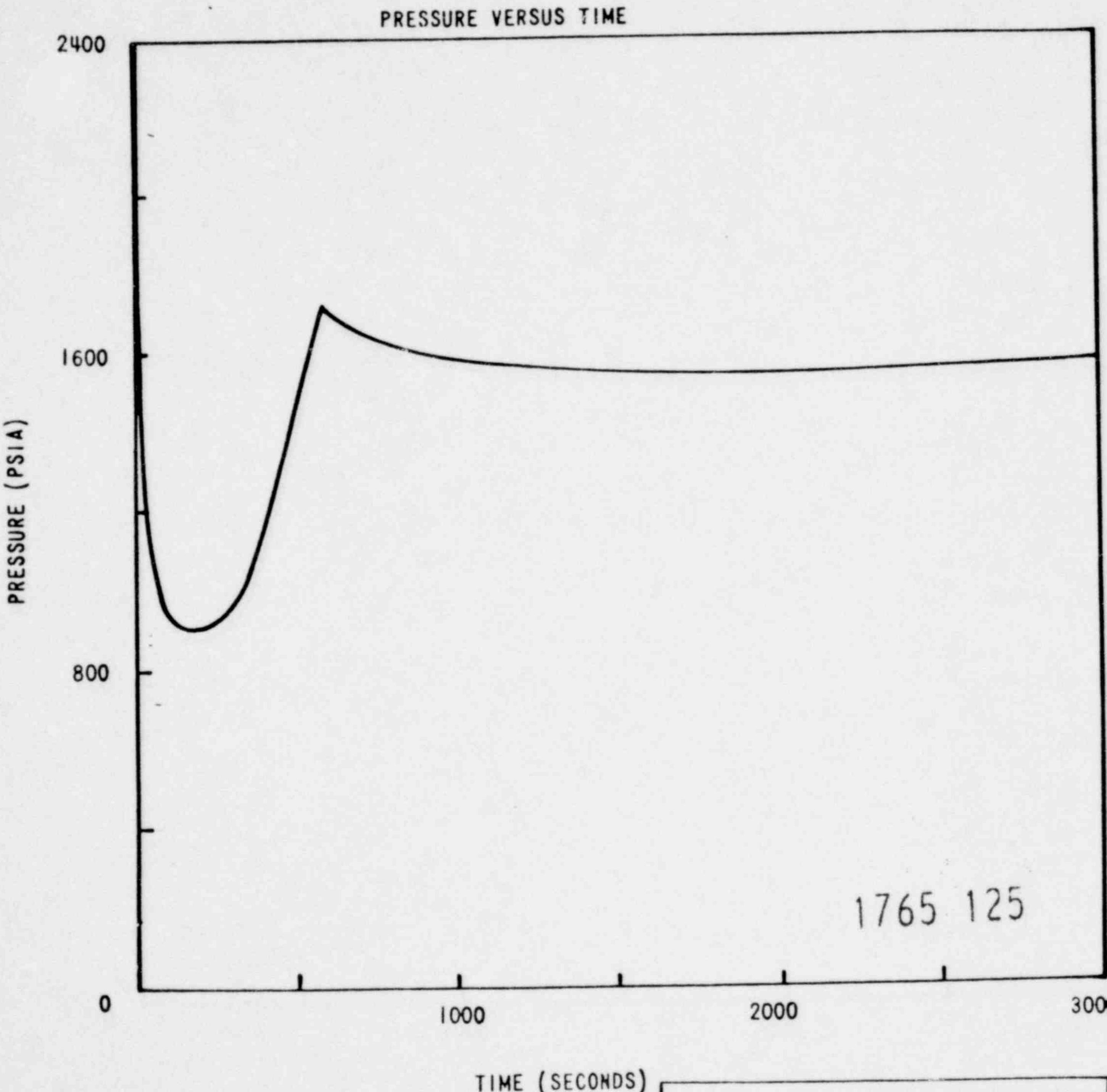
LC - LONGITUDINAL FLAW
CF - CIRCUMFERENTIAL FLAW



SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LARGE STEAMLINE BREAK WITH
REACTOR COOLANT PUMPS RUNNING
FIGURE Q6.54-1

Added by Amendment 58, December 22, 1978



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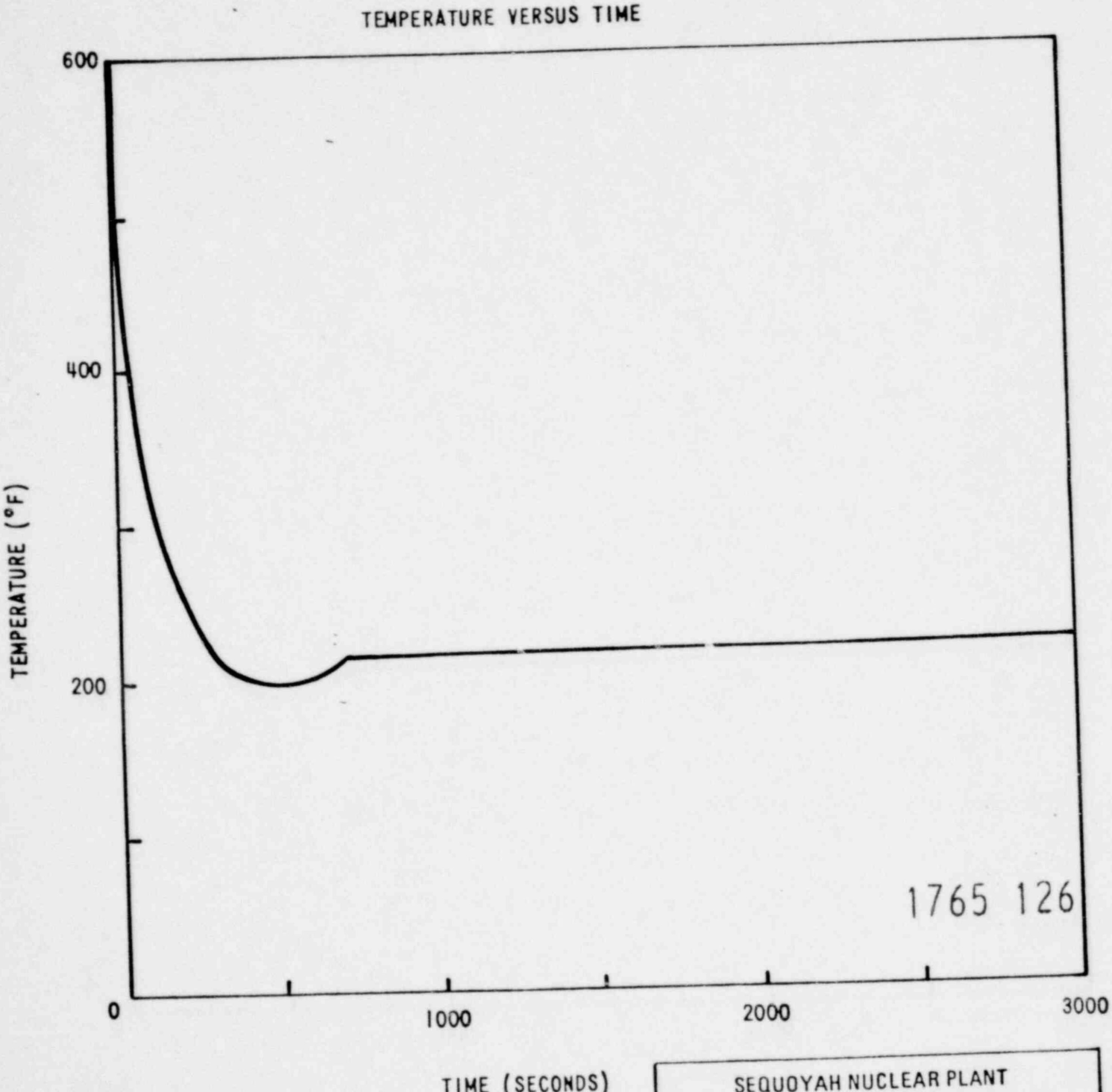
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

LARGE STEAMLINE BREAK WITH
REACTOR COOLANT PUMPS RUNNING
FIGURE Q6.54-2

Added by Amendment 58, December 22, 1978



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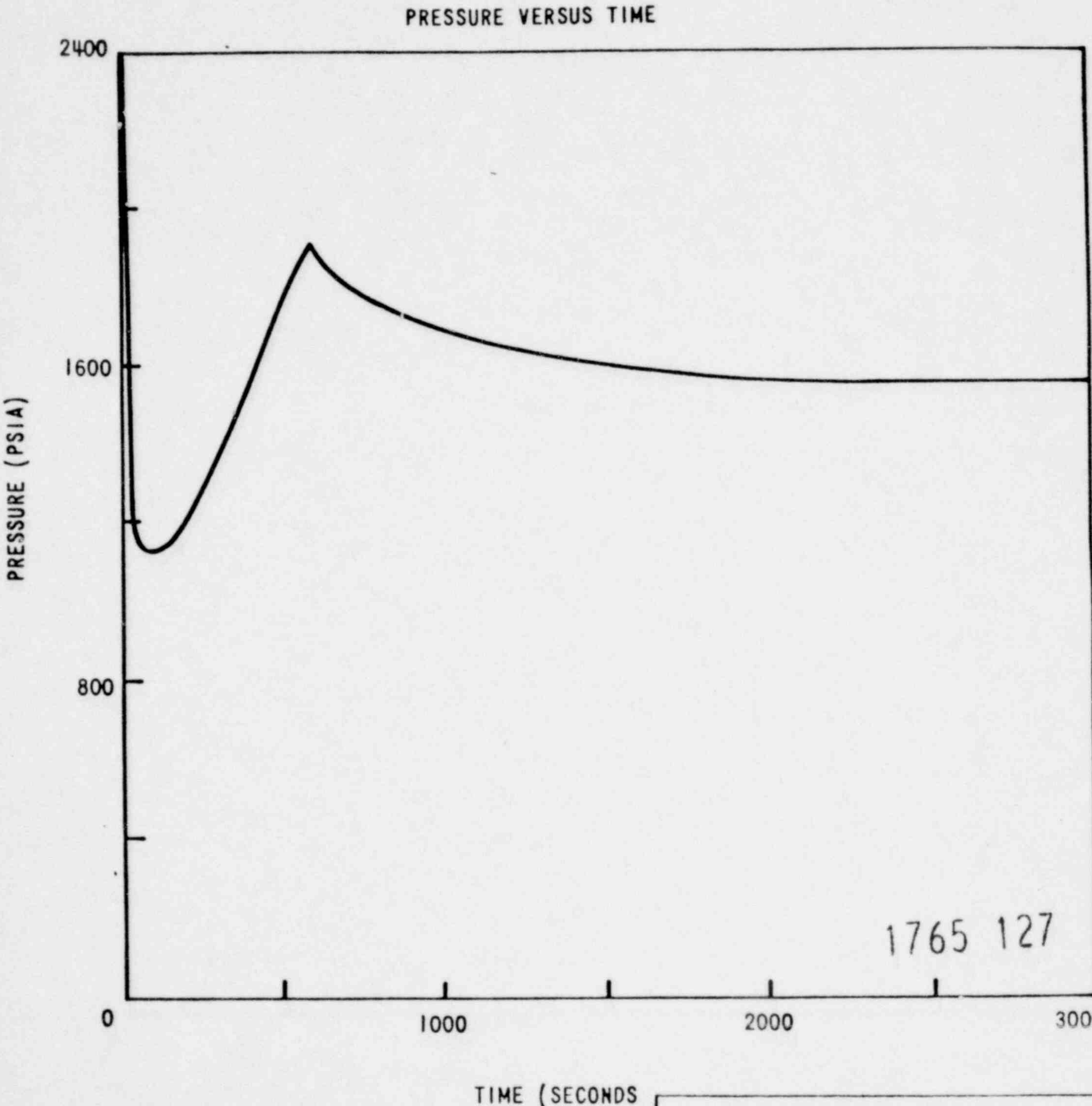
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

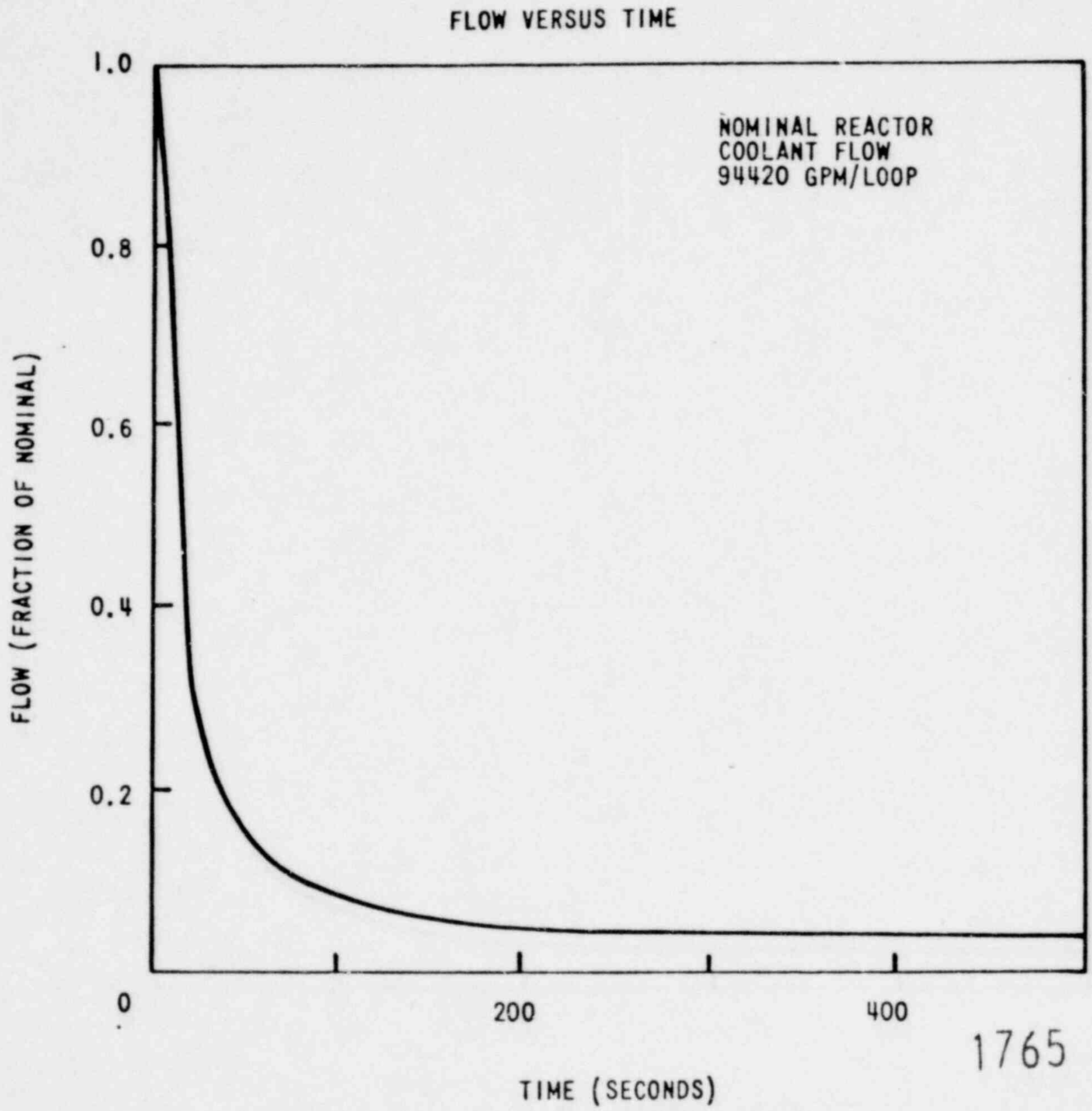
LARGE STEAMLINE BREAK WITH
REACTOR COOLANT PUMPS TRIPPED
FIGURE Q6.54-3

Added by Amendment 58, December 22, 1978



SEQUOYAH NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
LARGE STEAMLINE BREAK WITH REACTOR COOLANT PUMPS TRIPPED FIGURE Q6.54-4

Added by Amendment 58, December 22, 1978



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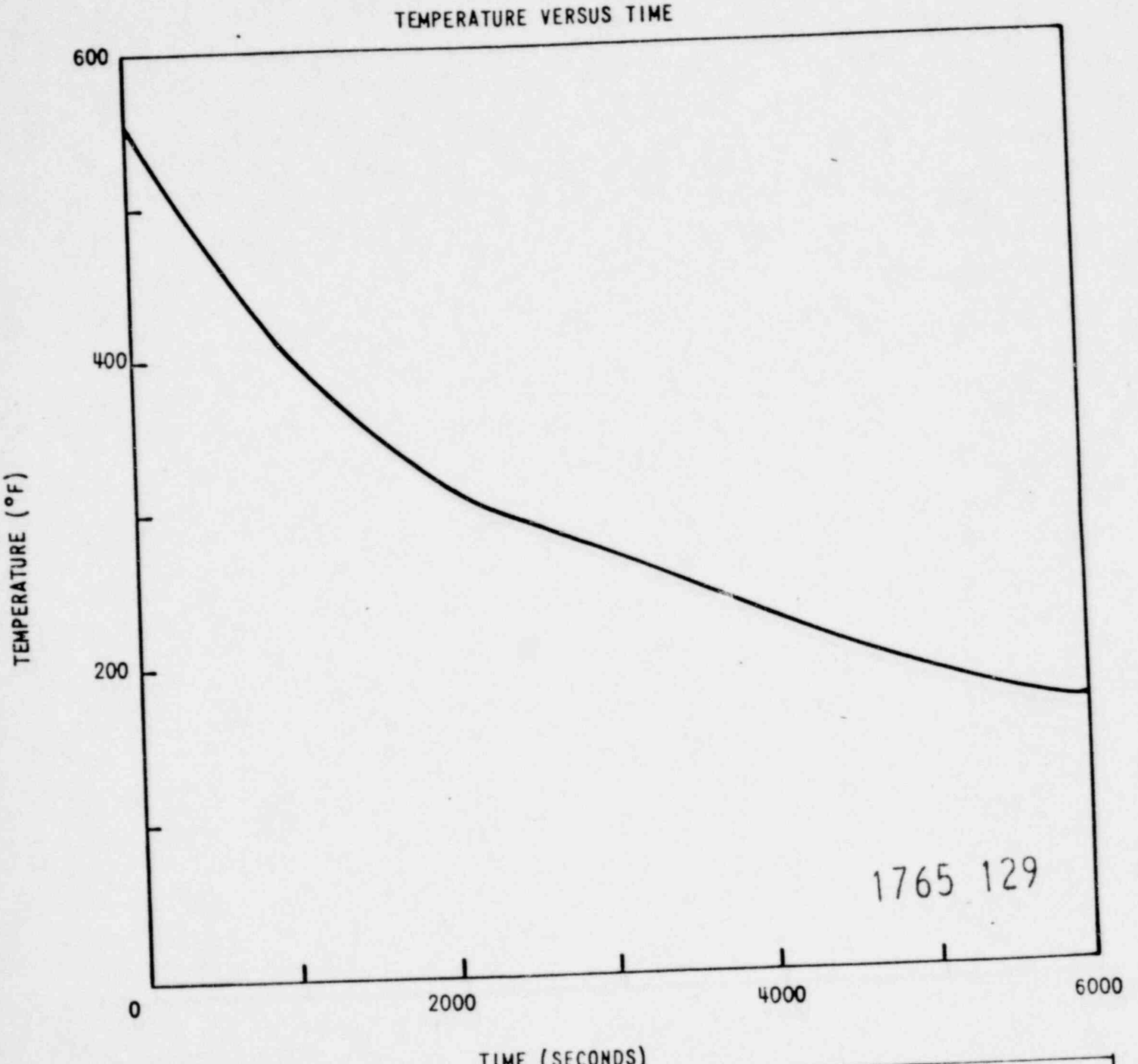
SEQUOYAH NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT
LARGE STEAMLINE BREAK WITH REACTOR COOLANT PUMPS TRIPPED FIGURE Q6.54-5

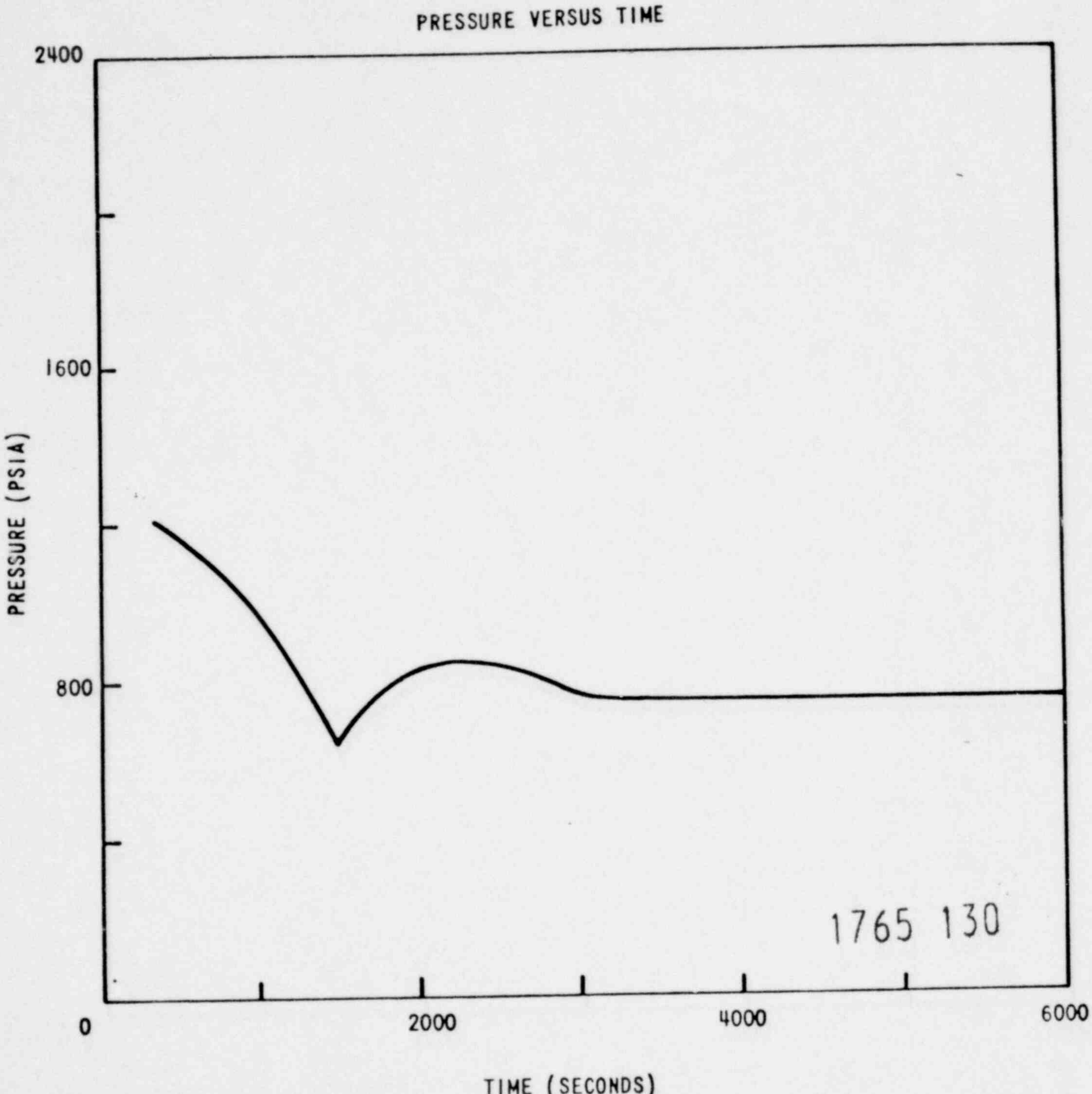
Added by Amendment 58, December 22, 1978



SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

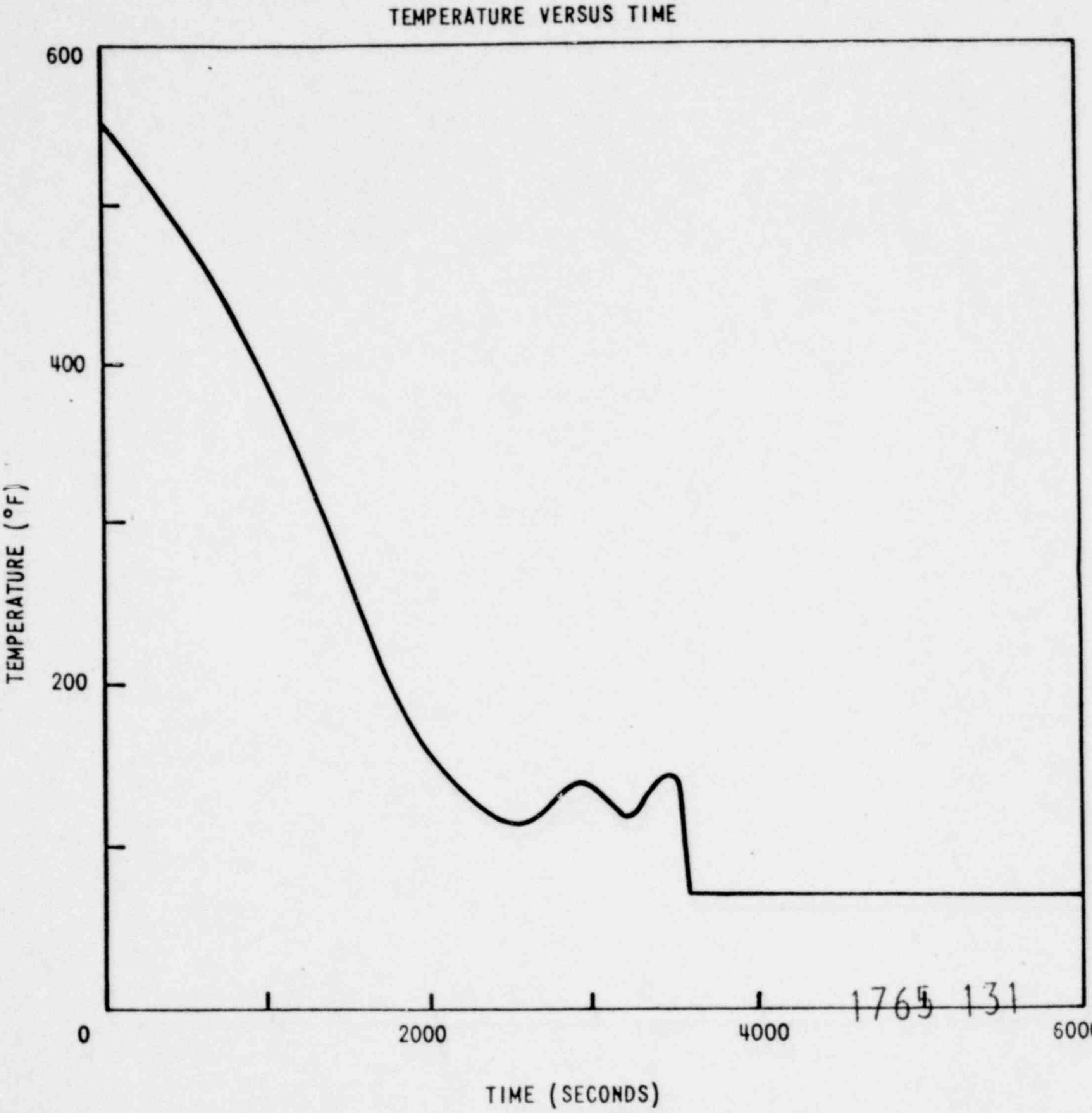
SMALL LOCA
2 INCH DIAMETER BREAK
FIGURE Q6.54-6

Added by Amendment 58, December 22, 1978



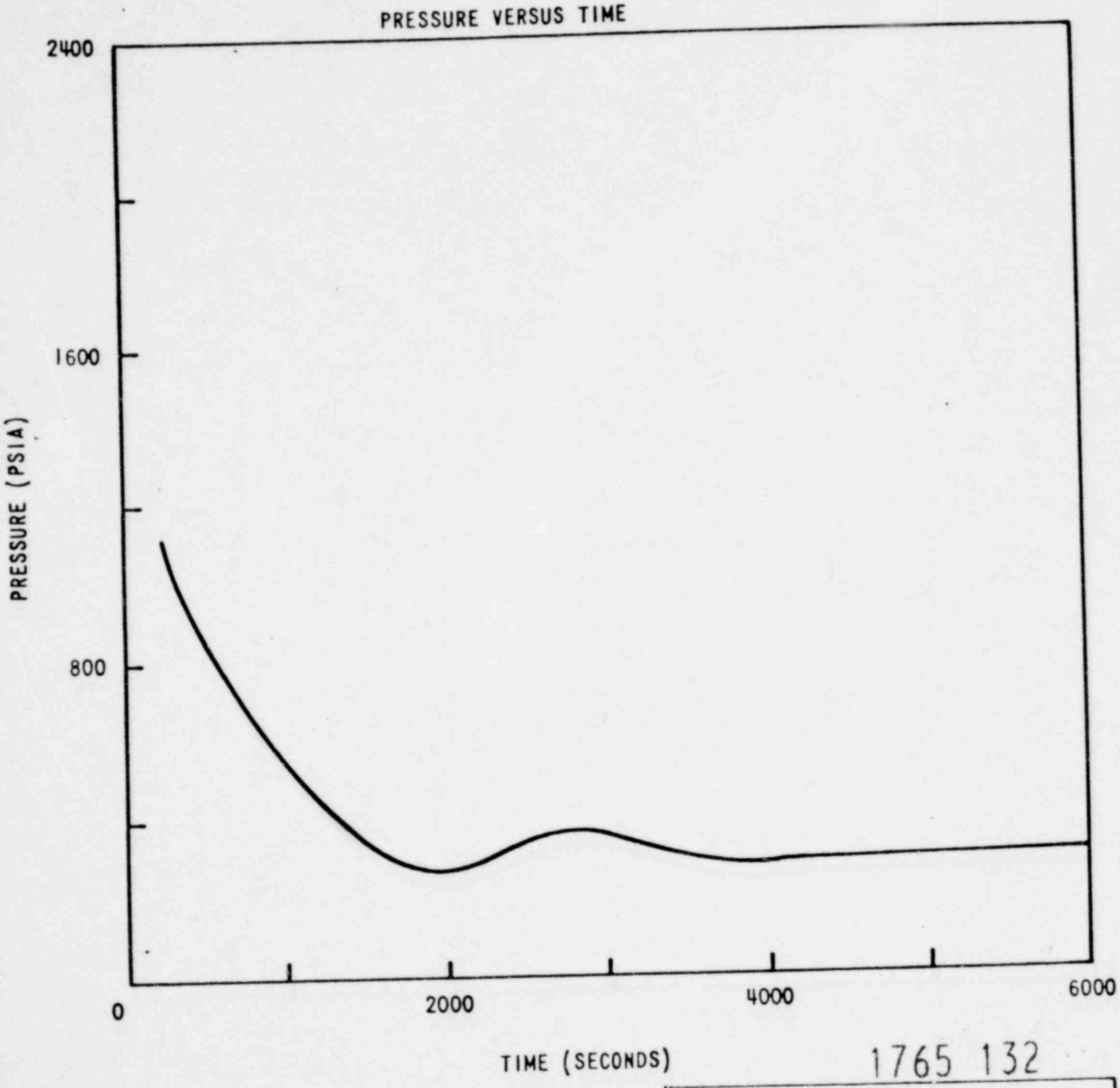
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
2 INCH DIAMETER BREAK
FIGURE Q6.54-7



SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
3 INCH DIAMETER BREAK
FIGURE Q6.54-8



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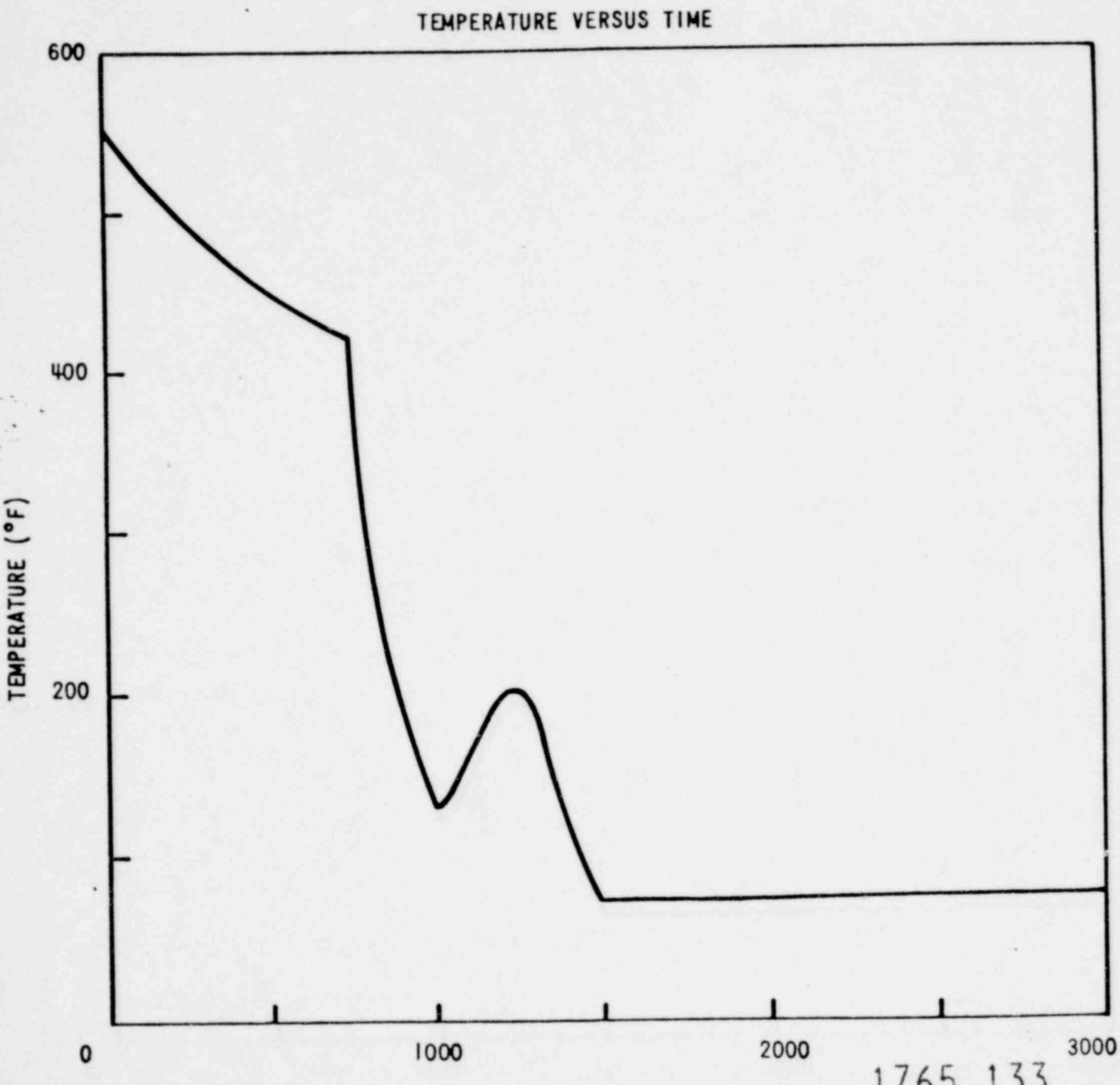
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
3 INCH DIAMETER BREAK
FIGURE Q6.54-9

Added by Amendment 58, December 22, 1978



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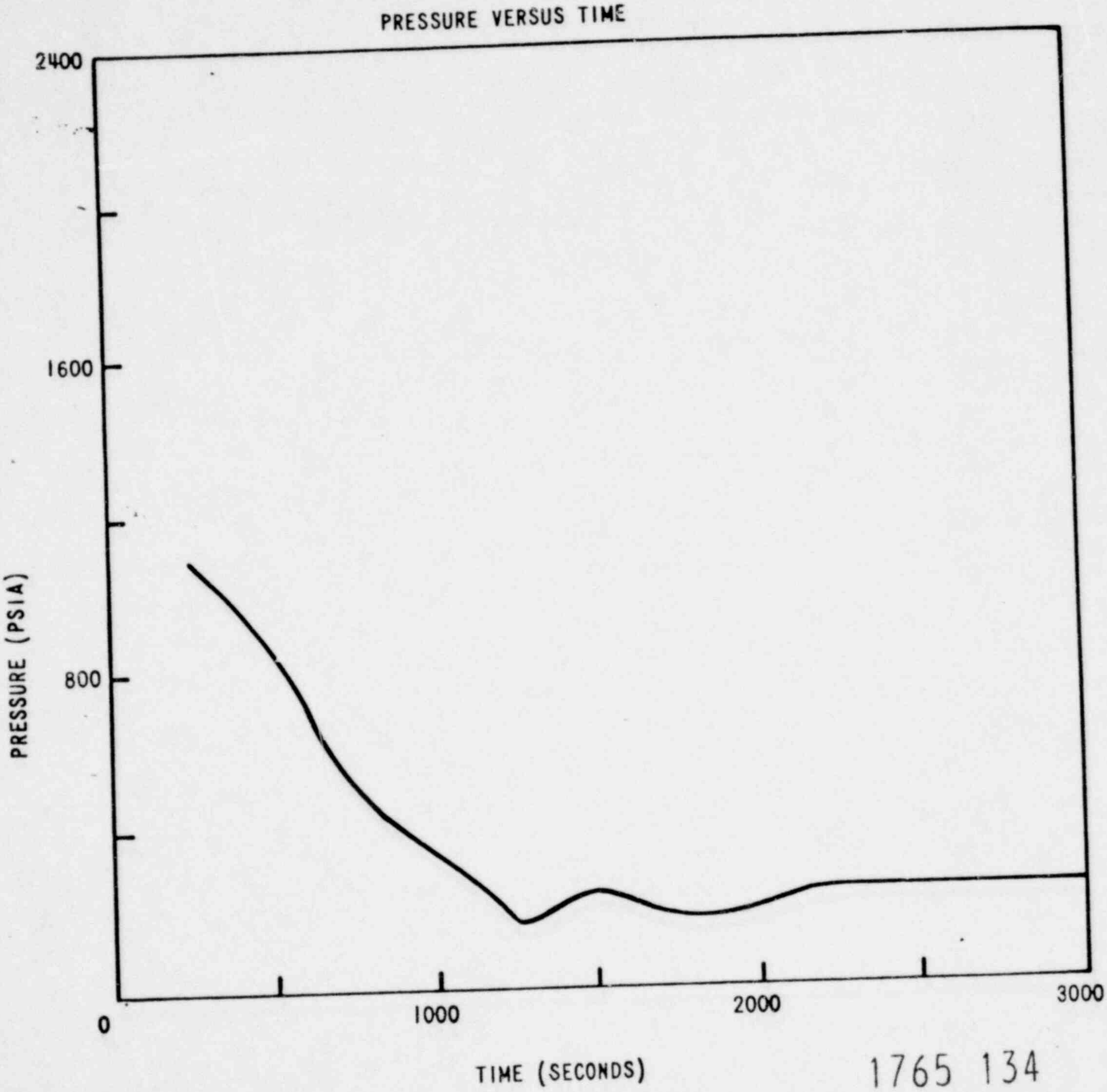
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
4 INCH DIAMETER BREAK
FIGURE 06.54-10

Added by Amendment 58, December 22, 1978



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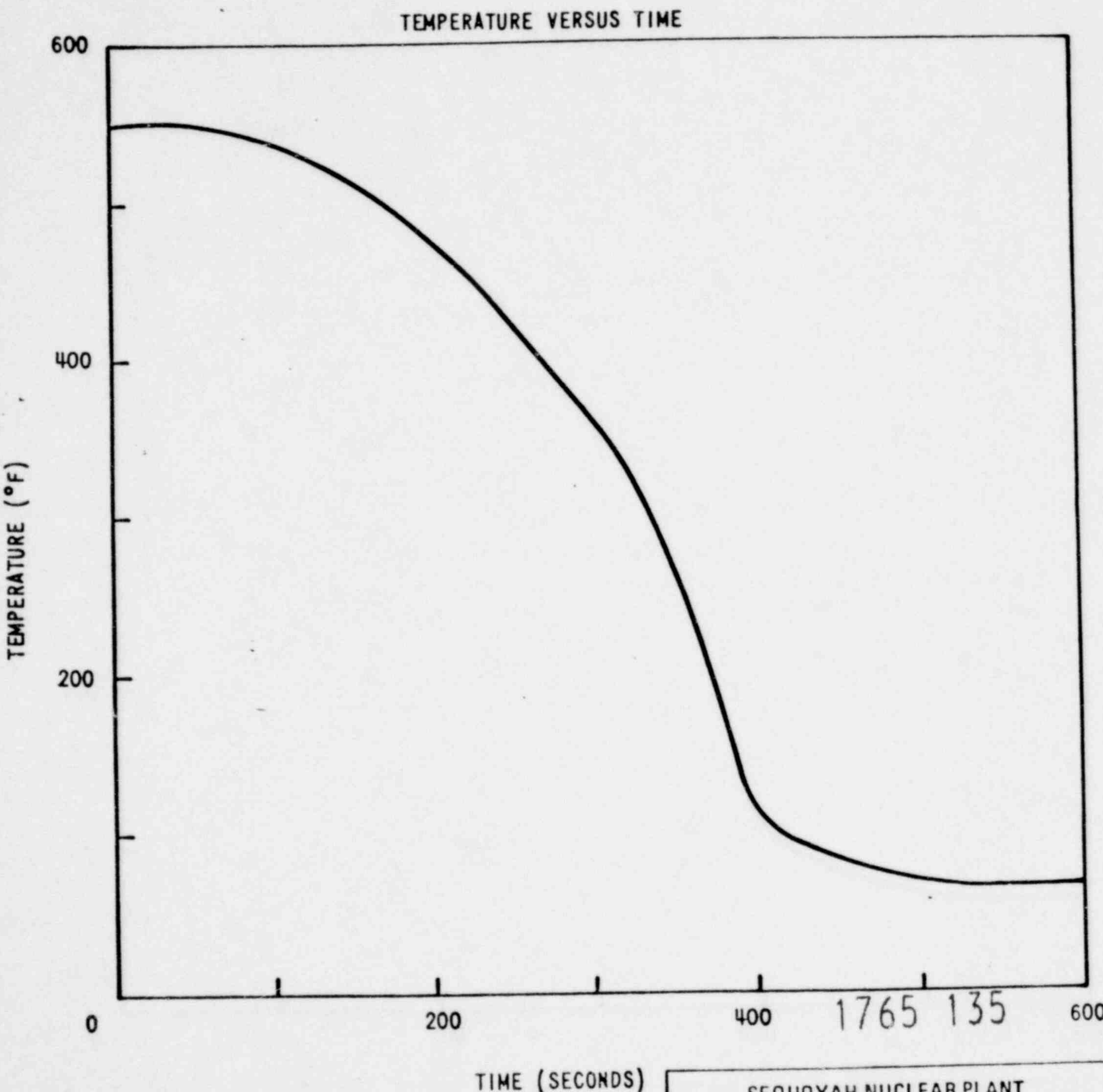
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
4 INCH DIAMETER BREAK
FIGURE Q6.54-11

Added by Amendment 58, December 22, 1978



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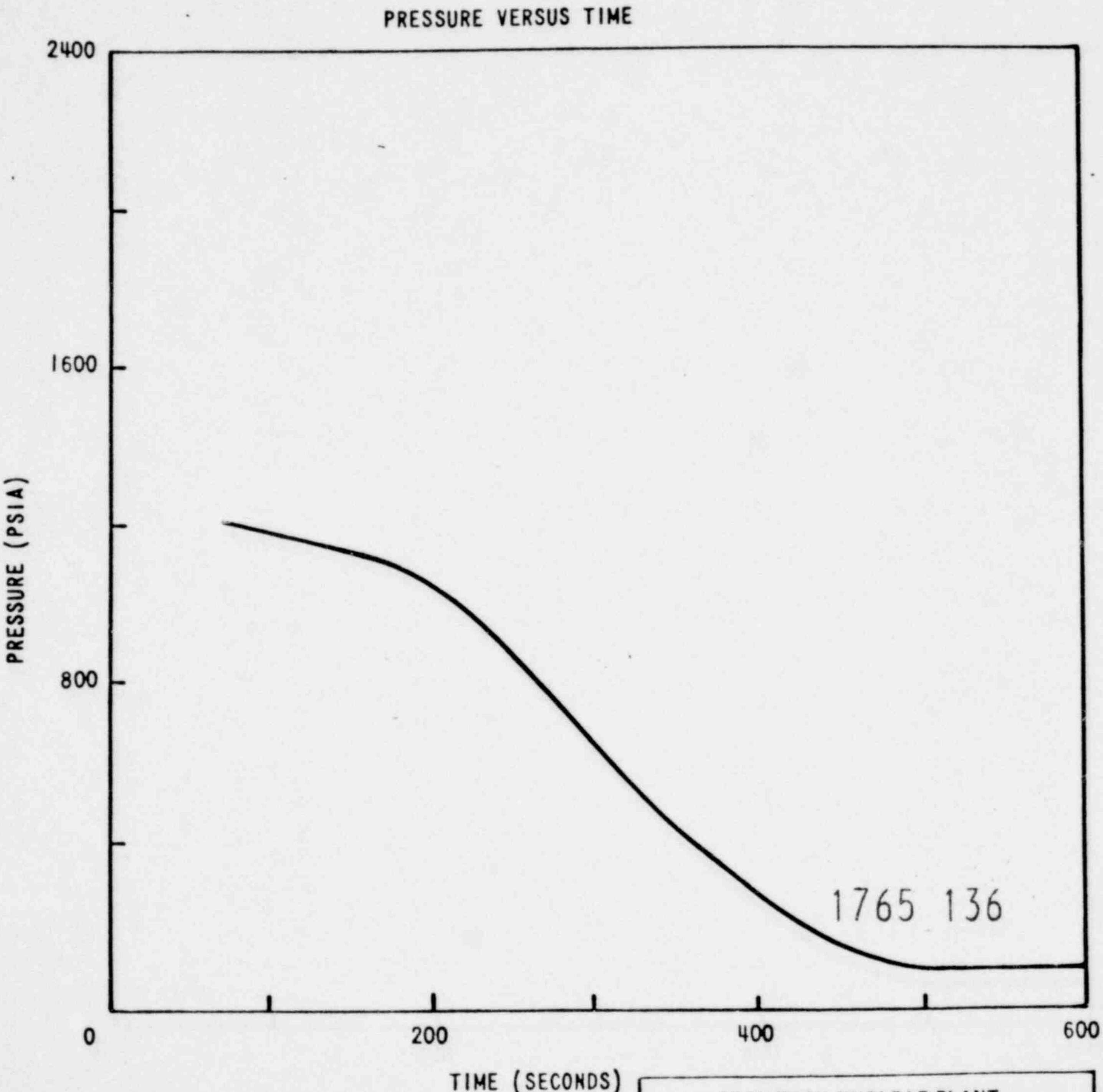
SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
6 INCH DIAMETER BREAK
FIGURE Q6.54-12

Added by Amendment 58, December 22, 1978



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SEQUOYAH NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SMALL LOCA
6 INCH DIAMETER BREAK
FIGURE Q6.54-13

501.

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Objects to continuous NRC demands that util meet addl criteria re overpressurization events, while NRC review process is unjustifiably lengthy & results only in requests for addl info. Believes $4000 fee request is unauthorized.

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NOTES:

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YANKEE ATOMIC ELECTRIC COMPANY

YANKEE

20 Turnpike Road Westborough, Massachusetts 01581

WYR 78-88

October 16, 1978

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United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: William O. Miller, Chief
Licensing Fee Management Branch
Office of Administration

References: (a) License No. DPR-3 (Docket No. 50-29)
(b) YAEC Letter to USNRC dated June 5, 1978 (WYR 78-46)
(c) USNRC Letter, Daniel J. Donoghue, to Licensee, dated February 28, 1978
(d) USNRC Letter to YAEC dated October 2, 1978, certified mail to D. E. Vandenburgh

Dear Sir:

Subject: License Amendment Fee for Proposed Change No. 161 (Reference b)

It has always been Yankee's practice to operate its plants in a safe and responsible manner. Long before the NRC expressed any outward concerns regarding overpressurization events, Yankee, through detailed operating procedures, would not permit any operator action which could develop into a potential Low Temperature Over-Pressurization (LTOP) event. Our operating record is a testimony of our commitment to this philosophy, as Yankee Rowe has never experienced a LTOP transient in its seventeen years of operation.

During the past two years the staff has escalated once common utility/NRC concerns regarding potential overpressurization events to the point where the only acceptable solution is a full commitment to design and install costly systems, in order to meet evolving criteria. The staff has taken the position that regardless of current plant designs and procedural commitments, utilities must redesign to mitigate a LTOP event, rather than prevent its occurrence. Alternate plant specific designs are ignored by the staff as they continue developing additional criteria. Yankee takes exception to this practice, and has offered for review alternative designs which meet the intent of your staff's criteria.

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Following an NRC/utility meeting [...] the staff's growing concerns over LTO[...] procedures were again reviewed and up[...] controls were issued to further precl[...] possibly result in a LTOP; and we sub[...] requirement (then current practice) t[...] tor in the control room, whenever the [...]

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7810260160

POOR ORIGINAL

I think [illegible]
Can [illegible]
into the outline

Johnson's Outline

Characterized By: | **Opinion**

- Considerable Background & Historical Information (mat'ls, design, operation, regs, technology, problems) — OK

- Description of Work Done Under A-11: LEFM; E-PFM; Fluence; Annealing, RPV Data, Accident Analyses — OK – But Not Well Organized

- Description of ASST Results — OK
  - Relevant Results Only

- Do Fracture Analysis and Accident Analysis (In Some Generic Way) to indicate Failure Margin — NO
  - Should indicate how these are to be done and what margins are acceptable.

Entire Paper Got Be Better Organized

- Summary of Conclusions & Recommendations Up Front, e.g.

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POOR ORIGINAL

Can We Get All of This (My Outline + Johnson's) Done By The End of CY 1979? — My guess - no way

Question: What smaller chunk can we take?

Options: ① Do all except accident part (Thermal Schock, Cold Repressurization - How to analyze?. + Acceptance Criteria)

Pros: ① Eliminates the need for systems guy.

② Substantial Reduction in Scope

Cons: ① Leaves undone an important element

Novak - Rancho Seco

Knight, Hazelton, Gamble — Multiply curve by 2 for emergency conditions

Sandy Israel

See A

~~A~~

POOR ORIGINAL

A-11

TAP A-11

Outline of NUREG (goal of program).

I. Introduction

"What are we talking about here?"

A. Steels and welds - used in RPV construction

B. Ductile-brittle fracture transition

C. RPV Design

D. Heat-up and cool-down limitations.

① Accidents & Transients

② Summary of Conclusions and recommendations.

③ Low-toughness problem

II. Historical

Calc techniques in past — A. Linear elastic fracture mechanics.

Limits / B. 10 CFR 50 Criteria

C. ASME Code

ESI → D. RPV Inspection (Appendix H)

Material Degrad. E. Neutron radiation effects

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POOR ORIGINAL

Basis for acceptance

F. 50 ft-lb Criterion and the Charpy Upper Shelf.

III. Problems.

A. Welds - high Cu; specific flux-wire combinations Weld materials problems

B. Surveillance program irregularities. - Surveillance problem

C. The use of LEFM - inapplicable to low-strength metals; questionable re: conservatism. Calc tech. problem

D. Fluence calculation uncertainties. Calc tech. problem

E. Treatment of ~~Treatment of~~ Accidents - what is a credible scenario? ~~What event sequence should~~ be used in defining ~~loads~~

IV. TAP A-11 Elements

Overview

A. Elastic-plastic fracture mechanics analysis.

B. Toughness based on E-P FM (J_{Ic} and $T_{matl.}$; both as functions of temperature and fluence).

?

C. Fluence estimations.

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D. RPV Annealing feasability - including environmental considerations.

E. RPV Data - QA (initial) and surveillance results; flaw state or probability; likelihood of in-service crack initiation and growth; environmental aspects of vessel NDE.

F. Accident analyses. - time-dependent pressure and temperature for worst-case <u>credible</u> scenarios.

V. <u>HSST Results.</u>

A. E_{c-v} and K_{Ic} data

B. J_{Ic} and T results

C. Thermal shock tests

D. ITV Results

VI. <u>RPV Annealing</u>

A. Test results

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B. Re-irradiation effects

C. Efficacy (including environmental aspects)

VII. Fluence

A. Results of NRC Programs (RES, NRL & BNL)

B. Fluence at RPV (utilizing neutron shield tank program results)

C. Potential for neutron flux reduction (at the RPV) through dummy assemblies at core corner locations.

VIII. RPV Data

A. QA and Surveillance

B. Sandia/DOE computer program (storage, retrieval and use in RPV calculations).

C. Radiation Effects (should n w/ E < 1 MeV be counted?)

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IX. Elastic-Plastic Fracture Analysis.

A. Flaw evaluation

B. Toughness – use output from HSST program; surveillance tests and RES E-P FM experimental program at Wash. U. to confirm T-modulus analysis.

C. Accident-related P & T.

D. E-P FM Formulation for RPV beltline region.

E. Results – show failure margin as a function of service life and starting material conditions.

Conclusions

X. Recommendations

outline

Req

Acc Criteria

Normal op & AT

accidents

R.E. Johnson
6-18-79

DRAFT
MAycock
6/18/79

TABLE OF CONTENTS - A-11 NUREG

IV. Acceptance Criteria

A. Acceptance criteria for normal operation (safety margins)

B. Acceptance criteria for postulated accidents and transients (includes a discussion of probabilities of the event sequences developed in III).

V. Calculations of Toughness for Low Toughness Vessels

A. Description of Calculational Technique developed by Paris, et al

B. Discussion of experimental verification of the validity of the technique

C. Conclusions regarding its use in licensing actions

D. Recommendations for Rule Making

VI. Discussion of Possible Methods of Improving Vessel Toughness or Limiting Toughness Degradation

A. Discussion of the Feasibility of Annealing (chances of success, environmental aspects, etc.)

B. Discussion of the possibility of Core Mods - European (German?) Experience

VII. Conclusions and Recommendations

Appendices

A. Summary of Operating Data

B. Proposed Branch Technical Position re: Consideration of postulated accidents and transients

C. Others as needed

Mtg. on A-11 — 6/8/79

Hanauer
Azarch
Johnson
Watt
Randall
Strosnider
Hazelton

<u>Thursday</u> — Going to Annapolis next week.
Break through - NRDC
(Working for research.

• Will the program now underway resolve the problem and if so when?

• Info — Materials fracture resistence
NA.

Stress	Resistance
1. Scenarios	1. Initial Mat'l. prop.
2. Variables fn(t)	2. <u>Properties</u> fn(t, nvt, T, etc
3. Loads	
P(t)	
T(t)	
ΔT(t)	Criteria
Geometry	Actions

PROBLEM — Low Upper Shelf Toughness on Some Vessels

— Did not have the technology to determine the toughness at or below the upper shelf.

— J Integral — Tearing Modulus

The point was to develop this technology to do this

End Point — NUREG Report

① Discussion of the technology

② Two evaluations — one of the material and one of stresses and a comparison.

③ How to test small material and how to scale it up to RV

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NUREG REPORT

① Criteria for selecting event scenarios

② Materials properties info ① survey info
② irradiation effects

③ Calculational methods and experimental verification.

④

INPUT

Technology
Eqn.
CRANIC

Acceptance
Criterion

Normal
Accident
Material
Properties

$\Rightarrow$ $\boxed{\text{Paris's Eqn.}}$ $\Rightarrow$ ANSWER

OK

NOT
OK

$\Rightarrow$ 6 years instead of a year and a half.

①	② (Warren's View)	③	④
INPUT	METHODOLOGY	Acceptance Criteria	Fix, IF NOT OK
Scenarios $\rightarrow$	this is A-11	OK vs. NOT OK How much margin is good enough?	Annealing
Material Properties $\rightarrow$ (I Mad Effects Program) · etc.		e.g. Normal ops. - factor of 2 on Press.	Core Ma[illegible] - [illegible] Fuel Assemblies In Corners

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