FINAL SAFETY ANALYSIS REPORT

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 & 3 · VOLUME 15

Southern California Edison Company

SAN DIEGO GAS & ELECTRIC COMPANY

FOREWORD

The Final Safety Analysis Report (FSAR) for the San Onofre Nuclear Generating Station Units 2 and 3 was prepared based upon Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 2. In addition, appendices have been added to facilitate the organization or presentation of information and to provide additional information.

Standards used for editorial abbreviations and symbols are the latest editions of the following IEEE-approved American National Standards Institute publications: ANSI-Y1.1, Abbreviations; ANSI-Y10.19, Letter Symbols for Units Used in Science and Technology; and ANSI-Y10.5, Letter Symbols for Quantities Used in Electrical Science and Electrical Engineering.

All text pages are numbered by chapter and section. Tables and illustrations are numbered in a similar manner; e.g., table 1.1-1 is the first table in section 1.1. Each table is placed in the text following the page on which it is first referenced; figures are placed at the end of each section.

Appendices are identified by section or chapter number with a suffixed letter and are placed following the applicable section or chapter.

Ammendments to the FSAR are identified by a bold line and the amendment number in the outside margin. The number and date of the most recent amendment affecting a page is placed at the bottom of that page. A list of effective pages is submitted with each amendment to provide a guide for inserting and removing pages.

Questions and Responses initiating amendments to the FSAR appear in separate volumes subdivided by tabs identifying the functional branches originating the questions. References are provided indicating corresponding changes to the text.

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15. ACCIDENT ANALYSES

15.0 TRANSIENT ANALYSES

This chapter presents analytical evaluation of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. These incidents are postulated and their consequences analyzed despite the many precautions which are taken in the design, construction, quality assurance, and plant operation to prevent their occurrence. The potential consequences of such occurrences are then examined to determine their effect on the plant, to determine whether plant design is adequate to minimize consequences of such occurrences, and to assure that the health and safety of the public and plant personnel are protected from the consequences of even the most severe of the hypothetical incidents analyzed.

The structure of this section is based on the eight by three matrix specified in reference 1. Initiating events are placed in one of eight categories of process variable perturbation specified in reference 1 and discussed in subsection 15.0.1. The frequency of each incident[a] was estimated, and each incident was placed in one of three frequency categories specified in reference 1 and discussed in subsection 15.0.1.

15.0.1 IDENTIFICATION OF CAUSES AND FREQUENCY CLASSIFICATION

The analyses of incidents considered in this chapter are presented according to the format explained by table 15.0-1 and illustrated in the Table of Contents for this section. The initiating events are each placed in one of the categories of process variable perturbation listed in table 15.0-1. The initiating events for which analyses are presented are listed in table 15.0-2 along with their respective section designations.

Certain initiating events which are suggested for consideration in reference 1 have not been explicitly analyzed. These initiating events along with the reasons for omission of their analyses are provided in the appropriate paragraphs in this chapter.

The frequency of each incident has been estimated, and each incident has been placed in one of the frequency categories listed in table 15.0-1. These frequency categories are defined as follows:

A. Moderate Frequency Incidents

These are incidents, any one of which may occur during a calendar year for a particular plant.

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a. Incidents are defined in this section as either the initiating event or initiating event in combination with one or more coincident component or system malfunctions and the resulting transient.

Table 15.0-1
CHAPTER 15 SUBSECTION DESIGNATION

Each subsection is identified as 15.W.X.Y.Z with trailing zeros omitted where:

W = 1 Increase in Heat Removal by the Secondary System (Turbine plant)

2 Decrease in Heat Removal by the Secondary System (Turbine plant)

3 Decrease in Reactor Coolant System Flowrate

4 Reactivity and Power Distribution Anomalies

5 Increase in Reactor Coolant Inventory

6 Decrease in Reactor Coolant Inventory

7 Radioactive Release from a Subsystem or Component

8 Anticipated Transients Without Scram

X = 1 Moderate Frequency Incidents

2 Infrequent Incidents

3 Limiting Faults

Y = Initiating Event (see subsection 15.0.1)

Z = 1 Identification of Causes and Frequency Classification

2 Sequence of Events and Systems Operation

3 Core and System Performance

4 Barrier Performance

5 Radiological Consequences

Table 15.0-2
CHAPTER 15 INITIATING EVENTS (Sheet 1 of 2)

Paragraph	Event
	MODERATE FREQUENCY INCIDENTS
15.1.1.1	Decrease in feedwater temperature
15.1.1.2	Increase in feedwater flow
15.1.1.3	Increased main steam flow
15.1.1.4	Inadvertent opening of a steam generator atmospheric dump valve.
15.2.1.1	Loss of external load
15.2.1.2	Turbine trip
15.2.1.3	Loss of condenser vacuum
15.2.1.4	Loss of normal ac power
15.3.1.1	Partial loss of forced reactor coolant flow
15.4.1.1	Uncontrolled CEA withdrawal from a subcritical or low power condition
15.4.1.2	Uncontrolled CEA withdrawal at power
15.4.1.3	CEA misoperation
15.4.1.4	CVCS malfunction (boron dilution)
15.4.1.5	Startup of an inactive reactor coolant system pump
15.5.1.1	CVCS malfunction
15.5.1.2	Inadvertent operation of the ECCS during power operation
	INFREQUENT INCIDENTS
15.1.2.1	Decrease in feedwater temperature[a]
15.1.2.2	Increase in feedwater flow[a]
15.1.2.3	Increased main steam flow[a]
15.1.2.4	Inadvertent opening of a steam generator atmospheric dump valve[a]
15.2.2.1	Loss of external load[a]

a. These incidents involve the same initiating event as the corresponding moderate frequency incidents but include either a concurrent single active component failure or single operator error.

Table 15.0-2
CHAPTER 15 INITIATING EVENTS (Sheet 2 of 2)

Paragraph	Event
15.2.2.2	Turbine trip[a]
15.2.2.3	Loss of condenser vacuum[a]
15.2.2.4	Loss of normal ac power[a]
15.2.2.5	Loss of normal feedwater flow
15.3.2.1	Complete loss of forced reactor coolant flow
15.3.2.2	Partial loss of forced reactor coolant flow[a]
15.3.2.3	Reactor coolant pump shaft seizure
15.5.2.1	CVCS malfunction[a]
	LIMITING FAULTS
15.1.3.1	Steam system piping failures
15.2.3.1	Feedwater system pipe breaks
15.2.3.2	Loss of normal feedwater flow[a]
15.3.3.2	Complete loss of forced reaction coolant flow[a]
15.4.3.1	Inadvertent loading and operation of a fuel assembly in an improper position
15.4.3.2	CEA ejection
15.6.3.1	Primary sample or instrument line break
15.6.3.2	Steam generator tube rupture
15.6.3.3	Loss of coolant accident
15.7.3.1	Waste gas system failure
15.7.3.2	Radioactive liquid waste system leak or failure
15.7.3.3	Postulated radioactive releases due to liquid-containing tank failures
15.7.3.4	Radiological consequences of fuel handling accidents
15.7.3.5	Spent fuel cask drop accidents
15.8	Anticipated transient without scram (ATWS)

B. Infrequent Incidents

These are incidents, any one of which may occur during the lifetime of a particular plant.

C. Limiting Faults

These are incidents that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

Certain malfunctions such as a stuck control element assembly (CEA) and coincident loss of normal ac power and coincident iodine spiking have been analyzed without explicit consideration of their effect on the incident frequency. The extremely low probability of these occurrences combined with the probability of the initiating event would produce an incident probability greatly less than that of the initiating event alone.

15.0.2 SYSTEMS OPERATION

During the course of any incident various systems may be called upon to function. These system are described in chapter 7 and include those systems designed to perform a safety function (see sections 7.2 through 7.6), i.e., the operation of which is necessary to mitigate the consequences of the incident, and those systems not required for safety (see section 7.7).

The reactor protective system (RPS) is described in section 7.2. Table 15.0-3 lists the RPS trips for which credit is taken in the analyses discussed in this section, the setpoints with uncertainties, and the trip delay times associated with each trip utilized in the analyses. The analyses of incidents take into consideration the response times of actuated devices after the trip setting is reached.

The elapsed time between the time when the setpoint condition exists at the sensor and the time when the trip breakers are open is defined as the trip delay time as shown in table 15.0-3. The trip delay times shown in table 15.0-3 are divided, for test purposes, into sensor delay time and plant protection system delay time. Sensor delay time is defined as the elapsed time between the time the condition exists at the sensor until the sensor output signal reaches the trip setpoint. This time is determined by manufacturer's test on typical sensor models. The plant protection system delay time is defined as the elapsed time between the input signal reaching the trip setpoint until the trip circuit breakers open. This time is determined during the preoperational test of the plant protection system. The sum of the sensor delay time and the plant protection system delay time must be less than or equal to the appropriate value listed in table 15.0-3.

The interval between trip breaker opening and the time at which the magnetic flux of the CEA holding coils has decayed enough to allow CEA motion is conservatively assumed to be 0.3 seconds. Finally, a conservative value of 3.0 seconds is assumed for CEA insertion, defined as the

Table 15.0-3
REACTOR PROTECTIVE SYSTEM TRIPS USED IN THE SAFETY ANALYSES

Events	Analysis Setpoint	Nominal Setpoint	Uncertainty	Trip Delay Time (s)
High logarithmic power level	2%	1%	+1% -0.5%	0.4
High linear power level	130%	125%	±5%	0.4
Low DNBR	1.19	1.19	(a)	0.75(b)
High local power density	(c)	(c)	(a)	0.75(b)
High pressurizer pressure	2,422 $lb/in.^2a$	2,400 $lb/in.^2a$	±22 $lb/in.^2$	0.4
Low pressurizer pressure	1,560 $lb/in.^2a$(d)	1,600 $lb/in.^2a$(d)	±40 $lb/in.^2$	0.4
Low steam generator water level	5%(e)(f)	10%(e)	±5%	0.4
Low steam generator pressure	675 $lb/in.^2a$	700 $lb/in.^2a$	±25 $lb/in.^2$	0.4

a. Calculated setpoint for the low DNBR and high local power density trips assure trip before indicated values, taking into account all sensor process delays and uncertainties. Further discussions of these setpoints and uncertainties are given in section 7.2.

b. The low DNBR trip delay time is discussed in section 7.2.

c. Setpoint value is set below the value at which fuel centerline melting would occur, see section 4.4.

d. See section 7.2

e. Percent of distance between the level nozzles above the lower nozzle.

f. The analysis setpoint corresponds to a water level 27.0 ft. above the tube sheet.

elapsed time from the beginning of CEA motion to the time of 90% insertion of the CEAs in the reactor core.

For example, the total time from the occurrence of a high linear power level condition at the sensor until the CEAs reach the 90% insertion position is 3.7 seconds (i.e., 0.4 second for trip delay, plus 0.3 second for CEA holding coil flux decay, plus 3.0 seconds for CEAs to reach 90% insertion position).

The engineered safety feature systems (ESFS) and systems required for safe shutdown are described in sections 7.3 and 7.4, respectively. The manner in which these systems function during incidents are discussed in each incident description.

The instrumentation which is required to be available to the operator in order to assist him in evaluating the nature of the incident and determining required action is described in section 7.5. The use of this instrumentation by the operator during each incident is discussed in each incident description.

Systems which are not required to perform safety functions are described in section 7.7. These include various control systems and the core operating limit supervisory system (COLSS). In general, normal automatic operation of these control systems is assumed unless lack of operation would make the consequences of the incident significantly more adverse. In such cases, the particular control system is assumed to be inoperative until the time of operator action. No credit is taken in the analysis for any operator action prior to initiation of the event which could normally mitigate the consequences of the transient; however, the analyses are performed on the basis that the plant is being operated within all limiting conditions for operation at the initiation of all events.

The effects of malfunctions of single active components or systems and/or operator errors are considered as noted in the discussions of specific incidents.

15.0.3 CORE AND SYSTEM PERFORMANCE

15.0.3.1 Mathematical Model

The nuclear steam supply system (NSSS) response to various incidents was simulated using digital computer programs and analytical methods most of which are documented in reference 2 and have been approved for use by the NRC by reference 3. Most of those programs and methods not documented in reference 2 are documented in topical reports which have been submitted to the NRC for review and are referenced herein.

15.0.3.1.1 Loss of Flow Analysis Method

The method used to analyze incidents which are initiated by a decrease in reactor coolant flowrate (section 15.3) is the static method documented in topical report CENPD-183(4) which was submitted to the NRC for review on August 22, 1975. The only deviation from that method was the use of the TORC computer code (see chapter 4) with the CE-1 CHF correlation (chapter 4) to calculate both the time and value of the minimum DNBR during the transient.

15.0.3.1.2 CEA Ejection Analysis Method

The method used for analysis of the reactivity and power distribution anomalies initiated by a CEA ejection (paragraph 15.4.3.2) is documented in topical report CENPD-190(5) which was approved by NRC for reference in license applications on June 10, 1976.

15.0.3.1.3 Anticipated Transients Without Scram Analysis Method

The method used to analyze the consequences of anticipated transients without reactor scram (section 15.8) are described in topical report CENPD-158, Revision 1(6) which was submitted to NRC for review on June 29, 1976.

15.0.3.1.4 CESEC Computer Program

The CESEC computer program is used to simulate the NSSS. The program is described in reference 7 and was referenced in 2.

CESEC computes key system parameters during a transient including core heat flux, pressures, temperatures, and valve actions. A partial list of the dynamic functions included in this NSSS simulation is: point kinetics neutron behavior, Doppler and moderator reactivity feedback, boron and CEA reactivity effects, multi-node average and hot channel reactor core thermal hydraulics, reactor coolant pressurization and mass transport, reactor coolant system safety valve behavior, steam generation, steam generator water level, main steam bypass, secondary safety and turbine valve behavior, as well as alarm, control, protection, and engineered safety feature systems. The steam turbine and its associated controls are not included in the simulation. Steam generator feedwater enthalpy and flowrate are provided as input to CESEC.

During the course of execution, CESEC obtains steady-state and transient solutions to the set of equations that mathematically describe the physical models of the subsystems mentioned above. Simultaneous numerical integration of a set of nonlinear, first-order differential equations with time-varying coefficients is carried out by means of a predictor corrector Runge-Kutta scheme. As the time variable evolves, edits of the principal system parameters are printed at prespecified intervals. An extensive library of the thermodynamic properties of uranium dioxide, water, and zircaloy is incorporated into this program. Through the use of CESEC, symmetric and asymmetric plant responses over a wide range of operating conditions can be determined.

15.0.3.1.5 CESEC-ATWS Computer Program

The CESEC-ATWS computer program is used to simulate the NSSS. The program is described in references 8 through 12 and was referenced in reference 2.

Several modifications have been made to the CESEC code described in reference 7 in order to extend the range of parameters which may be analyzed. These modifications include:

A. A thermal-hydraulic model of the reactor coolant system (RCS) that provides simultaneous solution of the equations of conservation of energy and mass.

B. A steam generator level model to determine the effective heat transfer area as the steam generator liquid inventory decreases below the top of the steam generator tube bundle.

C. A reactivity feedback model that separately accounts for effects of moderator density, moderator nuclear temperature, and non-uniform moderator distribution (boiling) effects.

D. A pressurizer model that represents filled, empty, and normal operating conditions.

15.0.3.1.6 COAST Computer Program

The COAST computer program is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in any hot or cold legs. The program is described in reference 13 and was referenced in reference 2.

The equation of conservation of momentum is written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. The equation of conservation of mass is written for the appropriate nodal points. Pressure losses due to friction, bends, and shock losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed vs. flow, for a pump at other than full speed.

15.0.3.1.7 STRIKIN-II Computer Program

The STRIKIN-II computer program is used to simulate the heat conduction within a reactor fuel rod and its associated surface heat transfer. The STRIKIN-II program is described in reference 14.

The STRIKIN-II computer program provides a single, or dual, closed channel model of a core flow channel to calculate the clad and fuel temperatures for an average or hot fuel rod, and the extent of the zirconium water reaction, for a cylindrical geometry fuel rod. STRIKIN-II includes:

A. Incorporation of all major reactivity feedback mechanisms

B. A maximum of six delayed neutron groups

C. Both axial (maximum of 20) and radial (maximum of 20) segmentation of the fuel element

D. Control rod scram initiation on high neutron power.

15.0.3.1.8 TORC Computer Program

The TORC computer program is used to simulate the fluid conditions within the reactor core and to predict the existence of DNB on the fuel rods. The TORC program is described in chapter 4 and was referenced in reference 2.

15.0.3.1.9 Reactor Physics Computer Programs

Numerous computer programs are used to produce the input reactor physics parameters required by the NSSS simulation and reactor core programs previously described. These reactor physics computer programs are described in chapter 4.

15.0.3.1.10 Loss of Coolant Accident Analysis Method

The method used to analyze the consequences of the loss of coolant accident (section 15.6) is described in topical reports CENPD-132P[15] and CENPD-137P[16] and was approved for reference in applications in 1974.

15.0.3.2 <u>Initial Conditions</u>

The incidents discussed in this section have been analyzed over a range of values for the principal process variables that affect the margin to fuel thermal design limits. These variables are the core power level, the core power distribution, the core inlet coolant flowrate, the core inlet coolant temperature, and the system pressure.

Analyses over a range of initial conditions is compatible with the monitoring function performed by the COLSS which is described in section 7.7 and the flexibility of plant operation which the COLSS allows. This flexibility is produced by allowing parameter tradeoffs by monitoring the principal process variables, synthesizing the margin to fuel thermal design limits, and displaying to the reactor operator the core power operating limit. The required margin to DNB incorporated in COLSS is established by the total loss of forced reactor coolant flow as described in appendix A to chapter 15. The required margin to DNB is based on the total loss of forced reactor coolant flow since this initiating event produces the most rapid loss of margin to DNB before reactor trip and the maximum loss of margin to DNB after reactor trip. Most often postulated initiating events do not require as much initial margin as evidenced by the fact that the reactor trip may be delayed (i.e., the time of trip is greater than 0.6 seconds) somewhat without causing a violation of the specified acceptable fuel design limit on DNB. The required margin to fuel centerline melting incorporated in COLSS is established by the loss of coolant accident (LOCA) as described in paragraph 15.6.3.3.

The range of values of each of the principal process variables that were considered in analyses of all incidents discussed in this section are listed in table 15.0-4. It is strongly emphasized that no plant operational or safety problems have been identified for operating conditions outside of the range shown in table 15.0-4. This range merely represents a range of expected normal reactor operation.

15.0.3.3 Input Parameters

The parameters used in the analyses are consistent with those listed in preceding sections and are primarily based on first-core values. Based on experience, it is not anticipated that a significant number of these parameters will change for subsequent fuel loadings. Nonetheless, for each licensing submittal for reload core, the calculated parameters for the proposed core will be compared with the values used for the first core. The impact of any parameter changes on the safety analysis will be evaluated. Then if any reanalysis is required, it will be performed and submitted.

15.0.3.3.1 Doppler Coefficient

The effective fuel temperature coefficient of reactivity (Doppler Coefficient) is shown in figure 4.3-38 and is multiplied by a weighting factor to conservatively account for higher feedback effects in the higher power density portions of the core and to account for uncertainties in determining the actual fuel temperature reactivity effects. The Doppler weighting factor, which is specified for each analysis, is 0.85 for cases where a less negative Doppler feedback produces more adverse results and 1.15 for cases where a more negative Doppler feedback produces more adverse results.

Table 15.0-4
SECTION 15 INITIAL CONDITIONS

Parameter	Units	Range
Core power, B	% of 3,410 MWt	$B \leq 102$
Radial 1-pin peaking factor, F_R (with uncertainty)	--	$F_R \leq 1.7$
Axial shape index, ASI[a]	--	$-0.6 \leq ASI \leq +0.6$
Core inlet coolant flowrate, G	%of 143x 10^6 lbm/h	$100 \leq G \leq 120$
Core inlet coolant temperature, T	°F	$520 \leq T \leq 560$ (100% power) $520 \leq T \leq 540$ (0 power)
System pressure, P	$lb/in.^2 a$	$2,000 \leq P \leq 2,300$

a. $$ASI = \frac{\text{area under axial shape in lower half of core} - \text{area under axial shape in upper half of core}}{\text{total area under axial shape}}$$

The effective fuel temperature correlation is discussed in section 4.3. This correlation related the effective fuel temperature, which is used to correlate Doppler reactivity, to the local core power. This correlation is used in both the CESEC (see paragraph 15.0.3.1.4) and CESEC-ATWS (see paragraph 15.0.3.1.5) computer programs to evaluate Doppler reactivity feedback.

15.0.3.3.2 Moderator Temperature Coefficient

The range of moderator temperature coefficient of reactivity at beginning of life (BOL) operating conditions is $+0.5 \times 10^{-4}$ $\Delta\rho/°F$ to -2.1×10^{-4} $\Delta\rho/°F$ and the corresponding range at end of cycle (EOC) conditions is -1.30×10^{-4} $\Delta\rho/°F$ to -3.3×10^{-4} $\Delta\rho/°F$. Allowances are included in the preceding to account for:

A. Changes between first cycle values and later cycle values

B. Changes in coefficient that might occur due to design changes

C. Changes in coefficient that might occur due to difference between design parameters and as built parameters (such as shim loadings, enrichments, etc.)

D. Any changes in parameters that might occur during a cycle

E. Calculational uncertainties or biases.

In addition, the moderator coefficient varies with changes in coolant temperature and the inserted control element assembly (CEA) worth. The most unfavorable value of the moderator coefficient is assumed for a particular analysis.

15.0.3.3.3 Shutdown CEA Reactivity

The shutdown reactivity is dependent on the CEA worth available on reactor trip, the axial power distribution, the position of the regulating CEAs, and the time in cycle life. The minimum total negative reactivity worth of the CEAs available for a reactor trip at full power and zero power is assumed to be -8.85% $\Delta\rho$ and -4.45% $\Delta\rho$ respectively. These values include the most reactive CEA stuck in the fully withdrawn position and the effects of cooldown to hot zero power temperature conditions. The full power value consists of -5.15% $\Delta\rho$ for shutdown and accident analysis allowance of -1.4% $\Delta\rho$ for fuel temperature variation from full power to hot standby.

For all accident analyses, except for the major RCS pipe rupture and major secondary system pipe rupture analyses, moderator void reactivity feedback effects are not taken into account. If no credit is taken for the negative reactivity, which would accompany the possible generation of voids during the course of a transient, then it would be conservative to ignore the positive reactivity feedback associated with the subsequent collapse of the voids. This assumption is justified because the positive feedback associated with collapse of the voids does not exceed the negative feedback (which was neglected) associated with the generation of the voids. Because of the substantial void formation in the core during the major RCS pipe rupture and major secondary pipe rupture transients, the effect of the growth and collapse of these voids on the reactivity feedback is modeled for these analyses.

The shutdown worth vs. position is calculated by assuming that the core is initially unrodded, i.e., all CEAs fully withdrawn. These assumptions are made:

A. Since the unrodded core allows the highest permissible axial peak to be used for the transient calculation

B. Since dropping CEAs into an initially unrodded core is more conservative in terms of the initial negative reactivity insertion during the transient.

The shutdown reactivity worth vs. position curve which was employed in the chapter 15 analyses except where noted in individual discussions of incidents is shown in figure 15.0-1. This shutdown worth vs. position curve was calculated assuming a more conservative rate of negative reactivity insertion than is expected to occur during the majority of operations, including power maneuvering. Accordingly, it is a conservative representation of shutdown reactivity insertion rates for reactor trips which occur as a result of anticipated transients or accidents.

15.0.3.3.4 Effective Delayed Neutron Fraction

The effective neutron lifetime and delayed neutron fraction are functions of fuel burnup. For each analysis, one of the following values of the neutron lifetime and the delayed neutron fraction is selected, depending upon the time in life analyzed.

	Neutron Lifetime (10^{-6} s)	Delayed Neutron Fraction
Beginning of Life (BOL)	30.8	0.007234
End of Cycle (EOC)	31.2	0.005295

15.0.3.3.5 Decay Heat Generation Rate

Analyses based upon full power initial conditions conservatively assume a decay heat generation rate based upon an infinite reactor operating period at full power.

15.0.4 BARRIER PERFORMANCE

15.0.4.1 Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.0.3.1.

15.0.4.2 Initial Conditions

The initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.0.3.2.

15.0.4.3 Input Parameters

The input parameters used for evaluation of barrier performance are identical to those described in paragraph 15.0.3.3.

15.0.5 RADIOLOGICAL CONSEQUENCES

This subsection summarizes the assumptions, parameters, and calculational methods used to determine the doses that result from postulated accidents. The accidents that were quantitatively analyzed are listed below. The radiological consequences of other accidents are referenced to these accidents as appropriate.

Accidents for which radiological consequences are quantitatively analyzed are:

A. Moderate Frequency Incidents

   1. Paragraph 15.1.1.4 - Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

B. Infrequent Incidents

   1. Paragraph 15.1.2.4 - Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

C. Limiting Faults

   1. Paragraph 15.1.3.1 - Steam System Piping Failures

   2. Paragraph 15.4.3.2 - CEA Ejection

   3. Paragraph 15.6.3.1 - Primary Sample or Instrument Line Break

   4. Paragraph 15.6.3.2 - Steam Generator Tube Rupture

   5. Paragraph 15.6.3.3 - Loss of Coolant Accidental

   6. Paragraph 15.7.3.1 - Waste Gas System Failure

   7. Paragraph 15.7.3.2 - Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)

   8. Paragraph 15.7.3.4 - Radiological Consequences of a Fuel Handling Accident

For each limiting fault, two separate analysis were conducted. The first analysis is based on design basis assumptions for purposes of determining adequacy of the plant design to meet 10CFR100 criteria. The second analysis is based on realistic assumptions to help quantify the margins that are inherent in the design basis approach.

Caution should be exercised in interpreting the results of analyses based on realistic assumptions. The definition of a limiting fault, as provided in subsection 15.0.1, is an incident that is not expected to occur but is postulated because its consequences include the potential for the release of significant amounts of radioactive materials. The realistic approach should not be interpreted to imply that the accident (limiting fault) is expected to occur. The parameters that have been modified for the realistic analyses are presented in the description of each limiting fault.

Information used repetitively throughout the section is provided in appendix 15B which contains information on dose models, atmospheric depersion factors, control room parameters, and activity release models.

REFERENCES

1. NCR Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," September 1975.

2. "Combustion Engineering Standard Safety Analysis Report," CESSAR Docket No. STN 50-470, December 1975.

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4. "C-E Methods for Loss of Flow Analysis," C-E, CENPD-183, July 1975.

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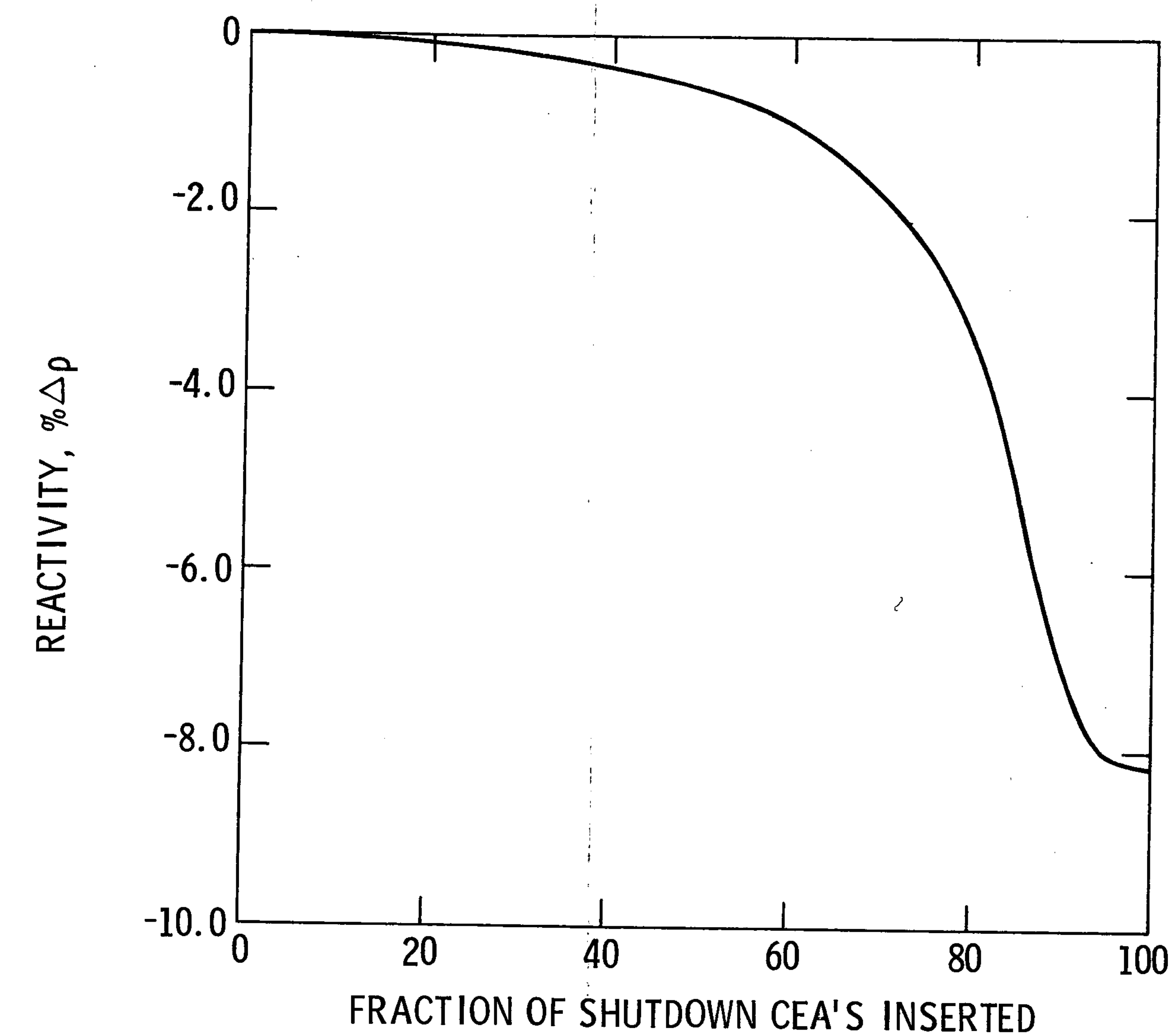

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

REACTIVITY vs. FRACTION OF SHUTDOWN (CEAs INSERTED)

Figure 15.0-1

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18OC6

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (TURBINE PLANT)

15.1.1 MODERATE FREQUENCY INCIDENTS

15.1.1.1 Decrease in Feedwater Temperature

15.1.1.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a decrease in feedwater temperature classifies it as a moderate frequency incident as defined in reference 1 of section 15.0. A decrease in feedwater temperature is caused by loss of one of several feedwater heaters. The loss could be due to interruption of steam extraction flow or to an opening of a feedwater heater bypass line. The high pressure heaters increase the feedwater enthalpy by 76 Btu/lb. The loss of any of the low pressure heaters before the feedwater pumps will produce a smaller effect (i.e., no more than 70 Btu/lb) due to the compensating effect of the high-pressure heater in that train.

15.1.1.1.2 Sequence of Events and Systems Operation

A decrease in feedwater temperature causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient and a decrease in the reactor coolant system (RCS) and steam generator pressures. Detection of these conditions is accomplished by the RCS and the steam generator pressure alarms and the high reactor power alarm. If the transient were to result in an approach to specified acceptable fuel design limits, trip signals generated from information provided by the core protection calculators would assure that low departure from nucleate boiling ratio (DNBR) or high local power density limits are not exceeded.

Because of a smaller cooldown rate, the systems' operations described above and the resulting sequence of events would produce consequences no more adverse than those following an increased main steam flow which is described in paragraph 15.1.1.3. The consequences of a single malfunction of a component or system following a decrease in feedwater temperature is discussed in paragraph 15.1.2.1.

15.1.1.1.3 Core and System Performance

The core and system performance parameters following a decrease in feedwater temperature would be no more adverse than those following an increased main steam flow which is described in paragraph 15.1.1.3.

15.1.1.1.4 Barrier Performance

The barrier performance parameters following a decrease of feedwater temperature would be less adverse than those following increased main steam flow (see paragraph 15.1.1.3).

15.1.1.1.5 Radiological Consequences

The radiological consequences of this event are less severe than results of the inadvertent opening of a steam generator atmospheric dump valve discussed in paragraph 15.1.1.4.5.

15.1.1.2 Increase in Feedwater Flow

15.1.1.2.1 Identification of Causes and Frequency Classification

The estimated frequency of an increase in feedwater flow classifies it as a moderate frequency incident as defined in reference 1 of section 15.0. An increase in feedwater flow is caused by:

A. Further opening of a feedwater control valve or an increase in feedwater pump speed. The maximum flow increase at full power is approximately 10% above nominal.

B. Startup of auxiliary feedwater with normal feedwater in the manual mode: The auxiliary feedwater system supplies relatively cold water from the condensate storage tank to the steam generators; the starting of this system would simultaneously increase feedwater flow and decrease feedwater temperature. If normal feedwater were in the automatic mode, the feedwater control valves would compensate for the increase in feedwater flow, and startup of the auxiliary feedwater would only result in a reduction in the feedwater enthalpy of no more than 20 Btu/lb.

15.1.1.2.2 Sequence of Events and System Operation

An increase in feedwater flow causes a decrease in the temperature of the reactor coolant, an increase in reactor power due to the negative moderator temperature coefficient, a decrease in the RCS and steam generator pressures and an increase in steam generator water level. Detection of these conditions is accomplished by the RCS and steam generator low-pressure alarms, high reactor power alarm, and high steam generator water level alarm. Protection against the violation of specified acceptable fuel design limits, as a consequence of an increase in feedwater flow, is provided by the low DNBR and high local power density trips. Protection against steam generator high water level is provided by the high steam generator water level trip.

Because of a smaller cooldown rate, the systems operations described above and the resulting sequence of events would produce consequences no more adverse than those following an increased main steam flow which is described in paragraph 15.1.1.3. The consequences of a single malfunction of a component or system following an increase in feedwater flow are discussed in paragraph 15.1.2.2.

15.1.1.2.3 Core and System Performance

The core and system performance parameters following an increase in feedwater flow would be no more adverse than those following an increased main steam flow which is described in paragraph 15.1.1.3.

15.1.1.2.4 Barrier Performance

The barrier performance parameters following an increase in feedwater flow will be no more adverse than those following an increased main steam flow (see paragraph 15.1.1.3).

15.1.1.2.5 Radiological Consequences

The radiological consequences of this event are less severe than results of the inadvertent opening of a steam generator atmospheric dump valve discussed in paragraph 15.1.1.4.5.

15.1.1.3 Increased Main Steam Flow

15.1.1.3.1 Identification of Causes and Frequency Classification

The estimated frequency of an increased steam flow incident classifies it as a moderate frequency incident as defined in reference 1 of section 15.0. The increased main steam flow incident results in the most adverse consequencies as a result of the closest approach to the specified acceptable fuel design limits (SAFDL).

The increase in heat removal by the steam generators as a result of increased main steam flow is defined as any rapid increase in steam generator steam flow, other than a steam line rupture, without the accompaniment of a turbine trip. Protection against violation of SAFDL as a consequence of the excessive heat removal is provided by the low DNBR and high local power density trips. The low steam generator water level trip, high reactor power trip, and low steam generator pressure trip will also serve to protect the plant from exceeding barrier design conditions.

An increase in main steam flow may be caused by any one of the following incidents of moderate frequency:

A. An inadvertent increased opening of the turbine admission valves caused by operator error or turbine load limit malfunction. This can result in an additional 10% flow.

B. Failure in the turbine bypass control system which would result in an opening of one or more of the turbine bypass valves. The flowrate of each valve is approximately 11% of the full power turbine flowrate. There are four turbine bypass valves for a total of 45% flow.

C. An inadvertent opening of an atmospheric dump valve or steam generator safety valve (for a discussion of this occurrence and presentation of results see paragraph 15.1.1.4) caused by operator error or failure within the valve itself. Each atmospheric dump and safety valve can release approximately 5% of the full power turbine flowrate.

As indicated by the possible increases in steam flow, the most severe of these incidents is case "B", the inadvertent opening of all of the turbine bypass valves at full power. This case results in the closest approach to the SAFDL since this case will release initially, approximately 145% of full main steam flow resulting in the most rapid cooldown and consequently largest power increase.

15.1.1.3.2 Sequence of Events and Systems Operations

Upon turbine trip, the steam bypass control system (SBCS) generates a quick opening signal to all of the turbine bypass valves resulting in the quick opening of these valves. The most severe excess heat removal is caused by a spurious generation of a quick open signal with no turbine trip. It is assumed that the failure in the SBCS results in these valves remaining open, even in the presence of closure signals generated by the SBCS due to adverse steam generator (e.g., low pressure, and low level) or condenser conditions until the operator takes action to close these valves or until the main steam isolation valves close. The increased main steam flow will result in an increase in core power and heat flux, and decrease in RCS temperature and pressure. The low DNBR trip will prevent the violation of fuel thermal limits. The initiation of the auxiliary feedwater system in conjunction with the low steam generator water level trip signal will act to maintain adequate inventory in the steam generators. The closure of the main steam isolation valves, following the low steam generator pressure signal, will stop the steam flow from the TBS valves. The increased main steam flow incident results in the most adverse consequences of any of the moderate frequency incidents, considered in the increase in heat removal, as a result of the closest approach to the SAFDL's.

Table 15.1-1 presents a step-by-step sequence of events from the generation of a "quick open" signal to the final stabilized condition.

15.1.1.3.3 Core and System Performance

15.1.1.3.3.1 <u>Mathematical Model</u>. The NSSS response to an increased main steam flow was simulated using the CESEC computer program described in section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with the CE-1 CHF correlation described in chapter 4.

15.1.1.3.3.2 <u>Input Parameters and Initial Conditions</u>. The input parameters and initial conditions used to analyze the NSSS response to an

Table 15.1-1
SEQUENCE OF EVENTS OF THE INCREASED MAIN STEAM
FLOW INCIDENT (Sheet 1 of 2)

Time (s)	Event	Setpoint or Value
0.0	A postulated spurious quick open signal generated by the steam bypass control system, all of the turbine bypass valves begin to open	----
1.0	All of the turbine bypass valves fully open	----
13.75	Low DNBR trip signal generated	1.19 projected
13.9	Trip breakers open	
14.2	Shutdown CEAs begin to drop into core	
14.6	Maximum core power, % of rated power	112
15.0	Maximum heat core average flux occurs, % of full power average channel heat flux	109
15.05	Minimum hot channel DNBR	1.19
17.1	Turbine admission and stop valves closed	
33.2	Feedwater control valves fully closed, main feedwater reaches 5% of full flow	----
45.6	Low pressurizer pressure alarm, $lb/in.^2a$	1,648
51.6	Pressurizer empties	
51.6	Low pressurizer pressure trip signal, $lb/in.^2a$	1,560
83.2	Low steam generator level alarm, feet above tubesheet	28.5
90.0	Low steam generator pressure trip signal, $lb/in.^2a$	675
90.0	Main steam isolation valves begin to close, feedwater isolation begin to close	

Table 15.1-1
SEQUENCE OF EVENTS OF THE INCREASED MAIN STEAM
FLOW INCIDENT (Sheet 2 of 2)

Time (s)	Event	Setpoint or Value
92.0	Low steam generator level trip signal, feet above tubesheet	27.0
92.4	Minimum steam generator pressure, $lb/in.^2a$	666
95.0	Main steam and feedwater isolation valves closed	
113.5	Minimum pressurizer pressure, $lb/in.^2a$	751
134.0	Auxiliary feedwater enters steam generator	
1,800.0	Operator initiates cooldown procedures if the malfunction has not already been corrected.	

increased main steam flow are discussed in section 15.0. In particular, those parameters, which were unique to the analysis discussed below, are listed in table 15.1-2.

The initial conditions for the principal process variables monitored by the core operating limit supervisory system (COLSS) were varied within the reactor operating space given in table 15.0-4 to determine the set of conditions that would produce the most adverse consequences following an increased main steam flow. Various combinations of initial core inlet temperature, core inlet flowrate, and pressurizer pressure were considered. Varying the RCS pressure or core flowrate had very little effect on the transient. Increasing the core inlet temperature resulted in a more rapid approach to the SAFDL, and also maximized the steam generator pressure; thereby resulting in greater steam releases. Various combinations of power level, moderate temperature coefficient, and peaking factors, each set of which represents a COLSS limit, were also considered. Sinee the results of all of these cases are essentially the same (i.e., a low DNBR trip with the minimum transient DNBR no less than 1.19), except for the time at which various events occur (e.g., the low DNBR trip could occur up to 3 seconds earlier as the core power is decreased and the peaking factors are increased) only the full power case is shown. This particular case is the most adverse case, when combined with a single failure (subsection 15.1.2), since the flattest axial, associated with the full power initial condition, produces a great number of pins having the highest peaking factors which results in the greatest potential for fuel damage. The moderator coefficient of reactivity was chosen to be the least negative at end of cycle (EOC) conditions, since this resulted in the most rapid approach to the SAFDL.

15.1.1.3.3.3 <u>Results</u>. The dynamic behavior of important NSSS parameters following an increased main steam flow are presented in figures 15.1-1 through 15.1-11.

The excess heat removal that occurs as a result of the opening of all of the turbine bypass valves results in the decrease in steam generator pressure and temperature. This decrease causes an increase in the RCS steam generator temperature difference, which results in more heat being transferred to the steam generator than is produced in the RCS, thus causing a decrease in the RCS temperature and pressure. The core power and consequently the heat flux increases due to the negative moderator coefficient of reactivity. The decreasing RCS pressure along with the increasing core heat flux results in a decreasing DNBR such that at 13.2 seconds the core protection calculators (CPC) DNBR projection generates a trip signal which acts to prevent violation of the SAFDL. At this point, the turbine stop valves begin to close and are fully closed in 3 seconds. The feedwater control valves also begin to close and are fully closed in 20 seconds. At 14.2 seconds the CEAs begin to enter the core and the peak core power of 112% is reached. The resulting decrease in heat flux arrests the decrease in hot channel DNBR at 15.0 seconds at a value greater than 1.19 (CE-1). The peak average core heat flux of 109% of full power heat flux also occurs here.

Table 15.1-2
ASSUMPTIONS FOR THE INCREASED MAIN STEAM FLOW ANALYSIS

Parameter	Assumption
Initial core power level, MWt	3,478
Core inlet coolant temperature, ^{o}F	560
Core mass flowrate, 10^6 lb_m/h	141.5
Reactor coolant system pressure, $lb/in.^2a$	2,200
Steam generator pressure, $lb/in.^2a$	945
Total nuclear heat flux factor, with uncertainty	2.37
Moderator temperature coefficient, 10^{-4} $\Delta\rho/^{o}F$	-1.3
Doppler coefficient multiplier	0.85
CEA worth for trip, 10^{-2} $\Delta\rho$	8.15
Turbine bypass system	Fails
Reactor regulating system	Manual
Feedwater regulating system	Automatic

The cooldown continues as a result of more energy being released by the turbine bypass valves than is produced by the core until at 90 seconds the low steam generator pressure trip signal is reached. This initiates a main steam isolation signal, which closes the main steam isolation valves 5 seconds later, closing the main steam line up stream of the turbine bypass valves, thus terminating flow. RCS cooldown continues, however, since more energy is required to heat the auxiliary feedwater to saturation conditions than is produced by the core. At 1800 seconds the oeprator initiates normal cooldown procedures, if the malfunction has not been corrected. The analysis presented conservatively assumes operator action is delayed until 30 minutes after first indication of the event.

The maximum RCS and secondary pressure do not exceed 110% of design pressure following an increased main steam flow, thus assuring the integrity of the RCS and main steam system is maintained. The minimum DNBR of greater than 1.19 indicates no violation of the fuel thermal limits.

15.1.1.3.4 Barrier Performance

15.1.1.3.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.1.1.3.3.

15.1.1.3.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of Barrier Performance are identical to those described in paragraph 15.1.1.3.3.

15.1.1.3.4.3 Results. After 30 minutes, the steam generator safety valves and pressurizer safety valves have not discharged any mass. The operator will then open the main steam isolation valve bypass lines and cool the plant via the condenser.

The radiological releases for this case will therefore be less severe than the radioactivity releases from the inadvertent opening of a steam generator atmospheric dump valve (see paragraph 15.1.1.4).

15.1.1.3.5 Radiological Consequences

The radiological consequences of this event are less severe than results of the inadvertent opening of a steam generator atmospheric dump valve discussed in paragraph 15.1.1.4.5.

15.1.1.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

15.1.1.4.1 Identification of Causes and Frequency Classification

The estimated frequency of an inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) classifies it as a moderate frequency incident as defined in reference 1 of section 15.0. This incident will result in the greatest radioactivity release.

An atmospheric dump valve may be inadvertently opened by the operator or may open due to failure in the control system that opens the valve. A steam generator safety valve may be opened only as a result of a valve failure. The inadvertent opening of either valve will result in the same consequences because they relieve steam at the same flowrate (5% of full power turbine flowrate).

15.1.1.4.2 Sequence of Events and Systems Operation

The inadvertent opening of a steam generator atmospheric dump valve is analyzed at a power level of 1 MWt. The initial conditions of the most adverse radiological consequences occur at this condition for the following reasons.

A. Turbine load controller is in manual, this results in the maximum steam generator pressure throughout the transient and subsequently greatest amount of steam released.

B. Feedwater controller in manual.

C. Greatest steam generator water mass.

D. Least rapid Doppler feedback effects.

The turbine controller is positioned in manual mode and remains closed throughout the incident. The main feedwater valves remain closed throughout the incident. No credit is taken for the action of the auxiliary feedwater system, which normally would have been activated as a result of the low steam generator trip signal generation that occurs during the transient.

In addition, the existing differential pressure between the affected and unaffected steam generator will isolate auxiliary feedwater to the affected steam generator throughout the EFAS control logic circuitry. This incident will result in the greatest radioactivity release of the moderate frequency incidents which result in an increase in heat removal by the secondary side.

Table 15.1-3 gives a step by step sequence of events from the opening of a steam generator atmospheric dump valve to the time when the operator takes control of the plant. The analysis presented conservatively assumes operator action is delayed until 30 minutes after first indication of the event.

Table 15.1-3
SEQUENCE OF EVENTS FOR THE INADVERTENT OPENING OF
A STEAM GENERATOR ATMOSPHERIC DUMP VALVE

Time (s)	Event	Setpoint or Value
0.0	One atmospheric dump valve opens fully	----
71.0	Local minimum pressurizer pressure, $lb/in.^2a$	2,202
84.0	Minimum steam generator pressure	903.5 affected steam generator 979.0 intact steam generator
102.5	Peak core power, % of rated power	11.2
104.0	Peak average core heat flux. % of full power heat flux	11
126.0	Maximum pressurizer pressure, $lb/in.^2a$	2,289
201.5	Maximum pressure in affected steam generator, $lb/in.^2a$	959.5
225.8	Maximum pressure in the intact steam generator, $lb/in.^2a$	1,033
812.0	Low steam generator level alarm, feet above tubesheet	28.5
886.5	Low steam generator level trip signal, feet above tubesheet	27
886.9	Trip breakers open	
887.2	Shutdown CEAs begin to enter core	
1,800.0	Operator takes control of plant	

15.1.1.4.3 Core and System Performance

15.1.1.4.3.1 Mathematical Model. The NSSS response to an inadvertent opening of a steam generator atmospheric dump valve was simulated with the CESEC computer program described in section 15.0.

15.1.1.4.3.2 Input Parameters and Initial Conditions. The assumptions and initial conditions given in table 15.1-4, in addition to the parameters described in section 15.0 are used for this analysis. COLSS does not monitor process variables at 1 MW so initial conditions were chosen to maximize steam release. The response of the RCS during an IOSGADV is insensitive to RCS initial conditions; therefore, these conditions were chosen at design conditions at zero power. The secondary conditions chosen to maximize steam release were the following:

A. 1 MW, the steam generator pressure is the highest at this power level.

B. Steam generator water level is just below the high steam generator water level trip setpoint; this will maximize the time until the low water level trip setpoint is reached.

C. 1 gal/min primary-to-secondary leak, this will maximize the radioactivity in the steam generator.

15.1.1.4.3.3 Results. The dynamic behavior of important NSSS parameters are presented in figures 15.1-12 through 15.1-19. The inadvertent opening of the steam generator atmospheric dump valve results in an excessive heat removal from the steam generator. The mass released from the valve is not made up by the feedwater, which is in the closed manual mode, so that the steam generator water level begins to decrease. The affected steam generator pressure begins to decrease, due to the excessive heat removal. The decreasing pressure and hence temperature in the affected steam generator results in a greater temperature difference between RCS and steam generator and hence more heat being transferred from the RCS to the steam generator. This action lowers the RCS temperatures and results in an increase in reactor power due to the negative moderator coefficient of reactivity. At about 104 seconds the core power and heat flux reaches their maximum value of 11% of rated power. This increase in power results in the heatup of the RCS, since the heat entering the RCS is greater than that extracted by the steam generator. The pressurizer pressure and RCS temperatures begin to increase, such that at 126 seconds the peak pressurizer pressure reaches its maximum value of 2289 $lb/in.^2a$. The increase in RCS temperatures results in a greater RCS-to-steam generator temperature difference, resulting in more heat being transferred to the steam generator and causing the steam generator temperatures and pressure to increase. As the power increases, the fuel temperatures increase, and as a result, the Doppler contribution increases. This decreases the positive reactivity and results in a decrease in core power and heat flux. At 200 seconds, a quasi-steady

Table 15.1-4
ASSUMPTIONS FOR THE INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE

Parameter	Assumption
Initial core power level, MWt	1
Core inlet coolant temperature, ^{o}F	544
Core mass flowrate, 10^{6} lb_{m}/h	141.5
Reactor coolant system pressure, $lb/in.^{2}a$	2,250
Steam generator pressure, $lb/in.^{2}a$	995
Total nuclear heat flux factor	2.37
Moderator temperature coefficient with uncertainties, 10^{-4} $\Delta\rho$	-3.75
Doppler coefficient multiplier	0.85
CEA worth on trip, 10^{-2} $\Delta\rho$	- 4.45
Reactor regulating system	Manual
Steam bypass system	Fails
Feedwater regulating system	Manual

state is reached at about 4% of rated core power. At 886.5 the affected steam generator water level has reached the low water level trip setpoint and initiates a reactor trip. The RCS and steam generators cool at a faster rate as a result of the lessened core power. At 1169.0, the low steam generator pressure alarm is triggered when the affected steam generator pressure reached 800 lb/in.^{2}. At 1800 seconds the operator takes control of the plant and begins an orderly cooldown using the condenser.

The maximum RCS and secondary pressure occur initially and therefore do not exceed 110% of design pressure.

15.1.1.4.4 Barrier Performance

15.1.1.4.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.1.1.4.3.

15.1.1.4.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.1.1.4.3.

15.1.1.4.4.3 Results. Figure 15.1-18 gives the steam generator atmospheric dump valve flowrate versus time for the IOSGAV. At 30 minutes after the atmospheric steam dump valves are opened, no more than 289,300 pounds of steam will have been discharged. The operator will then cool the plant via the condenser, resulting in very little additional radioactivity release to the environment.

15.1.1.4.5 Radiological Consequences

15.1.1.4.5.1 Physical Model. To evaluate the radiological consequences of the inadvertent opening of a steam generator atmospheric dump valve, it is assumed that the atmospheric dump valve remains open for 30 minutes (1800 seconds) until the operator takes control of the plant. The sequence of events and system operations is presented in paragraph 15.1.1.4.2. The secondary mass flowrate and integrated mass release from the affected steam generator is presented in table 15.1-5.

15.1.1.4.5.2 Assumptions, Parameters, and Calculational Methods. The major assumptions, parameters, and calculational methods used to evaluate the radiological consequences of the IOSGADV are presented in table 15.1-6. Additional clarification is provided as follows:

A. The reactor coolant system (RCS) equilibrium activity is based on long-term operation at 105% of the ultimate core power level of 3390 MWt (3390 MWt x 1.05 = 3560 MWt) with 1% failed fuel. Refer to table 11.1-2 for the isotopic distribution of RCS activity.

Table 15.1-5
MASS RELEASE - INADVERTENT OPENING OF STEAM GENERATOR
ATMOSPHERIC DUMP VALVE (IOSGADV)

Time (Seconds)	Rate Mass Flowrate Out of Steam Generators (lb_m/s)		Integrated Mass Flow Out (10^4 lb_m)		Water Mass Remaining in Steam Generators (10^5 lb_m)	
	Affected	Unaffected	Affected	Unaffected	Affected	Unaffected
0.0	0.0	0.0	0.00	0.0	2.92	2.92
50.0	199.5	0.0	1.02	0.0	2.82	2.92
100.0	198.2	0.0	2.00	0.0	2.71	2.92
200.0	208.7	0.0	4.06	0.0	2.50	2.92
741.2	205.5	0.0	15.21	0.0	1.33	2.92
1,000.0	174.9	0.0	20.14	0.0	0.85	2.92
1,500.0	126.2	0.0	27.60	0.0	0.13	2.92
1,608.5	0.0	0.0	28.93	0.0	0.00	2.92
1,800.0	0.0	0.0	28.93	0.0	0.00	2.92

Table 15.1-6
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A POSTULATED INADVERTENT OPENING OF A STEAM
GENERATOR ATMOSPHERIC DUMP VALVE (Sheet 1 of 3)

Parameter	Assumption
Data and assumptions used to estimate radioactive source	
General	
Power level, MWt	1
Burnup	EOC (Equilibrium)
Fuel perforated, %	0
Reactor coolant system activity	Table 11.1-2
Steam generator activity before accident (μCi/g dose equivalent I-131)	
Affected steam generator	0.1
Unaffected steam generator	0
Activity release from steam generators	
Unaffected steam generator, Ci	0
Affected steam generator, Ci	
Isotope (duration - 0 to 30 minutes)	
I-131	10.6
I-132	1.8
I-133	10.1
I-134	0.2
I-135	2.9
Kr-85M	0.2
Kr-85	0.5

Table 15.1-6
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A POSTULATED INADVERTENT OPENING OF A STEAM
GENERATOR ATMOSPHERIC DUMP VALVE (Sheet 2 of 3)

Parameter	Assumption
Isotope (duration - 0 to 30 minutes)	
Kr-87	0.1
Kr-88	0.4
Xe-131M	0.2
Xe-133	29.8
Xe-135M	0.1
Xe-135	0.8
Xe-138	0.1
Data and assumptions used to estimate activity released	
General	
Loss of offsite power	No
Credit for radioactive decay in transit to dose point after release	No
Auxiliary feedwater flow	No
Affected steam generator	
Primary-to-secondary leakage rate, gal/min	1
Mass of primary-to-secondary leakage (integrated for 1800 seconds), lb_m	180
Secondary mass release to atmosphere, lb_m	2.89 (s) (refer to table 15.1-7)
Steam generator decontamination factor between steam and water phase	1

Table 15.1-6
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A POSTULATED INADVERTENT OPENING OF A STEAM
GENERATOR ATMOSPHERIC DUMP VALVE (Sheet 3 of 3)

Parameter	Assumption
Unaffected steam generator	
Primary to secondary leakage rate, gal/min	0
Secondary mass release to atmosphere, lb_m	0
Dispersion data	
Distance to EAB, meters	576
Distance to LP3 outer boundary, meters	3,140
Atmospheric dispersion data	5% level X/Qs (refer to table 15B-4)
Dose data	
Method of dose calculation	Refer to appendix 15B
Dose conversion assumptions	Refer to appendix 15B

B. The steam generator equilibrium activity for the affected steam generator is assumed to be 0.1 μCi/g dose equivalent Iodine-131 (I-131) prior to the accident. This is the technical specification limit for steam generator activity.

C. Offsite power is available. At 1800 seconds the operator(s) takes control of the plant and conducts an orderly cooldown using the main condenser. Consequently there are no steam releases after 1800 seconds.

D. Only one steam generator is affected.

E. The primary-to-secondary leakage of 1 gal/min (technical specification limit) is assumed to continue to the affected steam generator for 1800 seconds. At 1800 seconds, the operator(s) is assumed to shut the affected steam generator atmospheric dump valve.

F. No credit is assumed for auxiliary feedwater flow. This allows the affected steam generator to blow down (i.e., dry) prior to 1800 seconds. A post accident DF of 1 was used for steam releases between the steam and water phase.

G. Calculated secondary mass releases are presented in table 15.1-7.

H. The activity released from the affected steam generator is immediately vented to the atmosphere. No credit for radioactive decay in transit to dose point is assumed.

I. The mathematical model used to analyze the activity released during the course of the accident is described in appendix 15B.

J. The atmospheric dispersion factors used in this analysis, which are based on meteorological conditions assumed present during the course of the accident, are calculated according to the model described in subsection 2.3.4. The 5% level 70/Qs presented in table 15B-4 were used.

K. The potential thyroid inhalation doses and beta-skin and whole body gamma immersion doses to an individual exposed at the exclusion area boundary (EAB) or outer boundary of the low population zone (LPZ) are analyzed using the models described in appendix 15B.

15.1.1.4.5.3 <u>Identification of Uncertainties and Conservatisms in the Evaluation of the Results</u>. The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of IOSGADV are as follows:

A. The RCS equilibrium activity is based on 1% failed fuel, which is greater by a factor of two to eight than that normally observed in past PWR operation.

Table 15.1-7
SECONDARY SYSTEM RELEASE (Sheet 1 of 2)

Time (seconds)	Unaffected Steam Generator				Affected Steam Generator	
	Steam Dump Valve Release		Release Through Break		Release Through Break	
	Flow (lb_m/s)	Integrated (lb_m)	Flow (lb_m/s)	Integrated (lb_m)	Flow (lb_m/s)	Integrated (lb_m)
0	70	0	7,008.0	0	8,026.0	0.0
2	70	0	5,758.0	12,813.8	6,417.7	14,495.3
4	70	0	4,862.1	23,426.0	5,305.2	26,202.2
6	70	0	3,346.3	32,305.6	4,516.8	36,106.2
8	70	0	1,247.3	37,209.2	3,941.1	44,640.2
10	70	0	0	38,284.1	3,509.9	52,144.5
20	70	0	0	38,284.1	2,372.8	80,853.8
30	70	0	0	38,284.1	1,789.4	10,156.9
40	70	0	0	38,284.1	1,409.5	117,500.7
50	70	0	0	38,284.1	1,138.3	130,232.1
60	70	0	0	38,284.1	925.9	140,554.6
70	70	0	0	38,284.1	748.5	148,941.2
80	70	0	0	38,284.1	588.5	155,647.4
90	70	0	0	38,284.1	469.2	160,904.5
100	70	0	0	38,284.1	397.7	165,224.7
200	70	0	0	38,284.1	241.8	193,883.0
300	70	0	0	38,284.1	219.4	216,752.6
400	70	0	0	38,284.1	197.5	237,621.7
500	70	0	0	38,284.1	177.9	256,323.1

Table 15.1-7
SECONDARY SYSTEM RELEASE (Sheet 2 of 2)

Time (seconds)	Unaffected Steam Generator				Affected Steam Generator	
	Steam Dump Valve Release		Release Through Break		Release Through Break	
	Flow (lb_m/s)	Integrated (lb_m)	Flow (lb_m/s)	Integrated (lb_m)	Flow (lb_m/s)	Integrated (lb_m)
540	70	0.0	0	38,284.1	0	270,287.0
1,800	74.1	0.0	0	38,284.1	0	270,287.0
2,000	74.1	14,820.0	0	38,284.1	0	270,287.0
5,000	74.1	237,120.0	0	38,284.1	0	270,287.0
10,000	74.1	607,620.0	0	38,284.1	0	270,287.0
15,000	74.1	978,120.0	0	38,284.1	0	270,287.0
16,920[a]	74.1	1,120,392.0	0	38,284.1	0	270,287.0

a. Time at which reactor coolant system temperature reaches 350F and shutdown cooling initiates.

B. The steam generator equilibrium activity for the affected steam generator is assumed to be equal to the technical specification limit (0.1 μCi/g dose equivalent I-131). This specific activity is greater by a factor of approximately 1300 than the usual expected steam generator activity (refer to table 11.1-21).

C. The assumption that the primary to secondary leakage of 1 gal/min (technical specification limit) in the affected steam generator is conservative because:

1. The 1 gal/min limit is applicable to both steam generators.

2. Operation with a 1 gal/min primary-to-secondary leak is not expected.

D. The assumption of no auxiliary feedwater flow is conservative as it allows the affected steam generator to blowdown (i.e., dry). Consequently, all of the activity present in the affected steam generator is assumed to be released (DF of 1 between steam and water phases).

E. The atmospheric dump valve is assumed to be inadvertently opened to the full open position. Inadvertent opening of this valve to any other position results in less severe offsite doses. Additionally, this valve is administratively controlled from the control room to prevent inadvertent operation.

F. The meteorological conditions assumed to be present at the site during the course of the accident are based on 5% level X/Qs. Meteorological conditions will be less severe 95% of the time. This results in the poorest values of atmospheric dispersion calculated for the EAB or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the EAB or LPZ outer boundary.

G. The assumption of no operator action for 1800 seconds (30 minutes) is a conservative assumption.

15.1.1.4.5.4 Conclusions.

15.1.1.4.5.4.1 Filter Loading. The only ESF filtration system considered in the analysis which limit the consequences of the inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) is the control room filtration system. Activity loading on the control room filter was based on the more serious loss-of-coolant accident. Since the control room filters are capable of accommodating the potential design-basis LOCA fission produce iodine loadings, more than adequate design margin is available with respect to the postulated IOSGADV accident releases.

15.1.1.4.5.4.2 Dose to an individual at the EAB and the Outer Boundary of the LPZ. The potential radiological consequences resulting from the occurrence of a postulated IOSGADV were conservatively analyzed, using assumptions and models described in previous sections.

The thyroid inhalation dose and the beta skin and whole body gamma doses due to immersion were analyzed for the 0 to 2-hour period at the EAB and for the duration of the accident at the outer boundary of the LPZ. These results are listed in table 15.1-8.

15.1.2 INFREQUENT INCIDENTS

15.1.2.1 Decrease in Feedwater Temperature with a Concurrent Single Failure of an Active Component

15.1.2.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a decrease in feedwater temperature with a concurrent single failure of an active component classifies it as an infrequent incident as defined in reference 1 of section 15.0. A decrease in feedwater temperature is caused by the possibilities described in paragraph 15.1.1.1.1.

Table 15.1-8
RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE

Result	Offsite Dose
I. Exclusion Area Boundary Dose (0 to 2 hours), rem	
Thyroid	1.9
Whole-body gamma	1.5×10^{-3}
Beta skin	8.3×10^{-4}
II. LPZ Outer Boundary Dose (duration), rem	
Thyroid	5.4×10^{-2}
Whole-body gamma	4.2×10^{-5}
Beta skin	2.4×10^{-5}

15.1.2.1.2 Sequence of Events and Systems Operation

The systems operations following a decrease in feedwater temperature with a concurrent single failure of an active component are the same as those described in paragraph 15.1.1.1.2. The single malfunction of a component or system is discussed in paragraph 15.1.2.3.1 for the increased main steam flow with a concurrent single failure of an active component. The smaller cooldown rate and, therefore the resultant sequence of events would produce consequences no more adverse than those following an increased main steam flow with a concurrent single failure of an active component which is described in paragraph 15.1.2.3.

15.1.2.1.3 Core and System Performance

The core and system performance parameters following a decrease in feedwater temperature with a concurrent single failure of an active component would be no more adverse than those following an increased main steam flow with a concurrent single failure of an active component which is described in paragraph 15.1.2.3.

15.1.2.1.4 Barrier Performance

The barrier performance parameters following a decrease in feedwater temperature with a concurrent single failure of an active component would be less adverse than those following an increased main steam flow with a concurrent single failure of an active component (see paragraph 15.1.2.3).

15.1.2.1.5 Radiological Consequences

The radiological consequences of this event are less severe than the results of the inadvertent opening of a steam generator atmospheric dump valve with a concurrent loss of offsite power discussed in paragraph 15.1.2.4.5.

15.1.2.2 <u>Increase in Feedwater Flow with a Concurrent Single Failure of an Active Component</u>

15.1.2.2.1 Identification of Causes and Frequency Classification

The estimated frequency of an increase in feedwater flow with a concurrent single failure of an active component classifies it as an infrequent incident as defined in reference 1 of section 15.0. An increase in feedwater flow is caused by the possibilities described in paragraph 15.1.1.2.1.

15.1.2.2.2 Sequence of Events and Systems Operation

The systems operations following an increase in feedwater flow with a concurrent single failure of an active component are the same as those described in paragraph 15.1.1.2.2. The single malfunction of a component or system is discussed in paragraph 15.1.2.3.1 for the increased main steam flow with a concurrent single failure of an active component. Because of the smaller cooldown rate, the resultant sequence of events would produce consequences no more adverse than those following an increased main steam flow with a concurrent single failure of an active component which is described in paragraph 15.1.2.3.

15.1.2.2.3 Core and System Performance

The core and system performance parameters following an increase in feed-water flow with a concurrent single failure of an active component would be no more adverse than those following an increased main feedwater flow with a concurrent single failure of an active component which is described in paragraph 15.1.2.3.

15.1.2.2.4 Barrier Performance

The barrier performance parameters following an increase in feedwater flow with a concurrent single failure of an active component would be less adverse than those following an increased main steam flow with a concurrent single failure of an active component (see paragraph 15.1.2.3).

15.1.2.2.5 Radiological Consequences

The radiological consequences of this event are less severe than results of the inadvertent opening of a steam generator atmospheric dump valve with a concurrent loss of offsite power discussed in paragraph 15.1.2.4.5.

15.1.2.3 <u>Increased Main Steam Flow with a Concurrent Single Failure of an Active Component</u>

15.1.2.3.1 Identification of Causes and Frequency Classification

The estimated frequency of an increased main steam flow with a concurrent single failure of an active component classifies this incident as an infrequent incident as defined in reference 1 of section 15.0. The cause of the increased main steam flow is discussed in paragraph 15.1.1.3.1.

A review of potential active component single failures to determine which failure would have the most adverse effect following an increased main steam flow indicates that the following single failures are most limiting: (1) loss of all ac power at any time during the tansient and (2) failure or unavailability of all of the condenser circulating water pumps that could result in overpressurization of the condenser. Parametric analysis has determined that the loss of all ac power, when a reactor trip condition exists, produces the most adverse consequences following an increased main steam flow. This failure is an independent loss of all normal onsite and offsite power.

15.1.2.3.2 Sequence of Events and System Operation

The systems and reactor trip that operate following an increased main steam flow with loss of all ac power, when a reactor trip condition exists, are the same as those described in paragraph 15.1.1.3.2 following an increased main steam flow with the following exceptions. The loss of all ac power, when a reactor trip condition exists, will result in the closure of the turbine bypass valves, since power is removed to the solenoids that act to keep the turbine bypass valve lines open. Also, no credit is taken for auxiliary feedwater flow in order to maximize steam release from the steam generators. The auxiliary feedwater system will be activated on the low steam generator water level trip, which occurs at 417.2 seconds, but credit is not taken for the auxiliary feedwater flow until the operator initiates cooldown.

Table 15.1-9 gives a sequence of events that occur following an increased main steam flow with concurrent loss of all ac power, when a reactor trip condition exists.

15.1.2.3.3 Core and System Performance

15.1.2.3.3.1 Mathematical Model. The mathematical model used for evaluation of core and system performance is identical to that described in paragraph 15.1.1.3.3.

15.1.2.3.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of core and systems performance are identical to those described in paragraph 15.1.1.3.3.

15.1.2.3.3.3 Results. The dynamic behavior of important NSSS parameters following an increased main steam flow with concurrent loss of all ac power, when a reactor trip condition exists, are presented in figures 15.1-20 through 15.1-32.

The dynamic behavior of the NSSS following an increased main steam flow with loss of all offsite power is identical to the increased main steam flow presented in paragraph 15.1.1.3 up until the time of trip. At 13.2 seconds, the NSSS also experiences a loss of forced reactor coolant flow and loss of main feedwater flow due to the loss of all ac power. At 13.6 seconds, the core power reaches its maximum value of 111% of rated power, and at 14.65 seconds, the core heat flux reaches its maximum value

Table 15.1-9
SEQUENCE OF EVENTS OF THE INCREASED MAIN STEAM FLOW
INCIDENT WITH CONCURRENT SINGLE FAILURE

Time (s)	Event	Setpoint or Value
0.0	Quick open signal generated by the steam bypass control signal, all of the turbine bypass valves begin to open	
1.0	All of the turbine bypass valves open	
13.2	Loss of all onsite and offsite electrical power	
13.75	Low DNBR trip signal generated	1.19 projected
13.9	Trip breakers open	
14.0	Minimum steam generator pressure, lb/in.^{2}a	828
14.2	Shutdown CEAs begin to drop into core	
14.2	Maximum core power, % of rated core power	111
14.2	Turbine bypass valves closed	
14.6	Maximum average core heat flux, % of full power heat flux	107
15.3	Minimum hot channel DNBR as calculated by TORC CE-1 correlation	1.06
16.4	Turbine stop and admission valves closed	
22.8	Steam generator safety valves open, lb/in.^{2}a	1,100
26.4	Maximum steam generator pressure, lb/in.^{2}a	1,139
73.6	Minimum pressurizer pressure, lb/in.^{2}a	1,900
206.0	Low steam generator level alarm, feet from tubesheet	28.5
417.2	Low steam generator level trip signal, feet from tubesheet	27.0
1,800.0	Operator takes control of plant	

of 107% of full-power heat flux. The steam generator pressure begins to increase due to the closure of the turbine, the turbine bypass valves, and main feedwater valves. The decreasing forced reactor coolant flow results in minimum DNBR of 1.06 at 15.3 seconds. At 16.4 seconds, the turbine stop valves and turbine bypass valves have fully closed, resulting in a cessation of steam flow. The steam generator pressure increases much more rapidly, until at 22.8 seconds, the steam generator safety valves open when the steam generator pressure reaches 1100 $lb/in.^2a$. At 26 seconds, the steam generator has reached its maximum 1139 $lb/in.^2a$ and begins to decrease. The cooldown continues as a result of more energy being released by the steam generator safety valves than is produced by the core. At 1800 seconds the operator takes control of the plant and begins an orderly cooldown. The analysis presented conservatively assumes operator action is delayed until 30 minutes after first indication of the event.

The peak RCS and main steam system pressures are 2200 and 1139 $lb/in.^2a$, respectively. These pressures are within 110% of design assuring the integrity of the RCS and NSSS is maintained following an increased main steam flow with loss of all ac power when a reactor trip condition exists. The minimum DNBR of 1.06 indicates that approximately 0.1% of the fuel pins will have experienced DNB using the method presented in reference 4 of section 15.0.

15.1.2.3.4 Barrier Performance

15.1.2.3.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.1.1.3.3.

15.1.2.3.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.1.1.3.3.

15.1.2.3.4.3 Results. Figure 15.1-31 gives the steam generator safety valve flowrate versus time following an increased main steam flow with loss of all ac power, when a reactor trip condition exists. Until the operator takes action at 30 minutes, the total steam release to the atmosphere through the steam generator safety valves is 226,400 pounds. The operator would then begin a controlled NSSS cooldown at 75F/h by opening the atmospheric dump valves. After 3 hours, the RCS will have reached a temperature of 350F, at which point, the shutdown cooling system may be placed in operation. About 668,000 pounds of steam are released during the cooldown. The total steam release to the atmosphere during the course of this transient is approximately 894,000 pounds.

15.1.2.3.5 Radiological Consequences

The radiological consequences of this event are less severe than results of the inadvertent opening of a steam generator atmospheric dump valve with a concurrent loss of offsite power discussed in paragraph 15.1.2.4.5.

15.1.2.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Single Failure of an Active Component

15.1.2.4.1 Identification of Causes and Frequency Classification

The estimated frequency of a IOSGAV with a concurrent single failure of an active component classifies it as an infrequent frequency incident, as defined in reference 1 of section 15.0.

15.1.2.4.2 Sequence of Events and Systems Operation

The systems operations following a IOSGAV with a concurrent single failure of an active component are the same as those described in paragraph 15.1.1.4.2. The single malfunction of a component or system is discussed in paragraph 15.1.2.3.1 for the increased main steam flow with a concurrent single failure of an active component. The resultant sequence of events would produce consequences no more adverse than those following an increased main steam flow with a concurrent single failure of an active component, which is described in paragraph 15.1.2.3.

15.1.2.4.3 Core and System Performance

The core and system performance parameters following an IOSGAV with a concurrent single failure of an active component would be no more adverse than those following an increased main steam flow with a concurrent single failure of an active component, which is described in paragraph 15.1.2.3.

15.1.2.4.4 Barrier Performance

The barrier performance parameters following an IOSGAV with a concurrent single failure of an active component would be less adverse than those following an increased main steam flow with a concurrent single failure of an active component (see paragraph 15.1.2.3), because of lower initial power level.

15.1.2.4.5 Radiological Consequences

15.1.2.4.5.1 Physical Model. To evaluate the radiological consequences of the inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) with loss of offsite power, it is assumed that the atmospheric dump valve on the affected steam generator remains open for the duration of the accident. Table 15.1-3 presents the sequence of events for the IOSGADV. The accident is considered to be terminated when shutdown cooling is initiated.

The affected steam generator boils dry in approximately 1600 seconds. Subsequent to boiling dry, steam releases from the affected steam generator are from a primary-to-secondary leak of 1 gal/min which is assumed to be present in the affected steam generator for the duration of the accident.

At 1800 seconds, cooldown of the plant is initiated by releasing steam from the unaffected steam generator. This cooldown continues until shutdown cooling is initiated at approximately 18,750 seconds.

Integrated mass releases from the affected and unaffected steam generators are presented in table 15.1-10.

15.1.2.4.5.2 Assumptions, Parameters, and Calculational Methods. The major assumptions, parameters, and calculational methods used to evaluate the radiological consequences of the IOSGADV are presented in table 15.1-11. Additional clarification is provided as follows:

A. The reactor coolant system (RCS) equilibrium activity is based on long term operation at 105% of the ultimate core power level of 3390 MWt (3390 MWt x 1.05 = 3560 MWt) with 1% failed fuel. Refer to table 11.1-2 for the isotopic distribution of RCS activity.

B. The steam generator equilibrium activity for the affected and unaffected steam generators is assumed to be 0.1 uCi/g dose equivalent Iodine-131 (I-131) prior to the accident. That is the technical specification limit for steam generator activity.

Table 15.1-10
MASS RELEASE - INADVERTENT OPENING OF STEAM GENERATOR ATMOSPHERIC DUMP VALVE (IOSGADV) WITH CONCURRENT LOSS OF OFFSITE POWER

Time (seconds)	Affected Steam Generator (lb_m)	Unaffected Steam Generator (lb_m)	Primary-to-Secondary Leakage (lb_m)
0	0	0	0
7,200 (2 hours)	2.893×10^5	4.1×10^5	893
18,750	2.893×10^5	8.66×10^5	2,325

Table 15.1-11
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE (Sheet 1 of 3)

Parameter	Assumptions	
Data and assumptions used to estimate radioactive source		
General		
Power level, MWt	1	
Percent of fuel perforated	0	
Reactor coolant system activity	Table 11.1-2	
Steam generator activity before accident (μCi/g dose equivalent I-131)		
Affected steam generator	0.1	
Unaffected steam generator	0.1	
Activity release from steam generators (duration of accident), curries		
Isotope	Affected Steam Generator[a]	Unaffected Steam Generator
I-131	15.1	3.1
I-132	3.1	0.5
I-133	15.9	2.9
I-134	0.7	0.
I-135	5.4	0.8
Kr-85M	2.5	0.1
Kr-85	5.4	0.1

a. Released activity from the affected steam generator includes contribution due to primary-to-secondary leakage of 1 gal/min for the duration of the accident.

Table 15.1-11
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A POSTULATED INADVERTENT OPENING OF A STEAM
GENERATOR ATMOSPHERIC DUMP VALVE (Sheet 2 of 3)

Parameter	Assumptions	
Isotope	Affected Steam Generator[a]	Unaffected Steam Generator
Kr-87	1.3	0.
Kr-88	4.3	0.1
Xe-131M	2.5	0.1
Xe-133	354.5	7.4
Xe-135M	1.2	0.
Xe-135	9.8	0.2
Xe-138	0.5	0
Data and assumptions used to estimate activity released.		
General		
Loss of offsite power	Yes	
Credit for radioactive decay in transit to dose point after release	No	
Auxiliary feedwater flow	Not to affected steam generator	
Affected steam generator		
Primary-to-secondary leakage rate, gal/min	1	
Mass of primary-to-secondary leakage (integrated for accident duration, lb_m)	Refer to table 15.1-7	
Secondary mass release to atmosphere, lb_m	Refer to table 15.1-7	

Table 15.1-11
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE (Sheet 3 of 3)

Parameter	Assumptions
Steam generator decontamination factor between steam and water phase	1 (iodines) 1 (noble gases)
Unaffected steam generator	
Primary-to-secondary leakage rate, gal/min	0
Secondary mass release to atmosphere, lb_m	Refer to table 15.1-7
Steam generator decontamination factor between steam and water phases	10 (iodines) 1 (noble gases)
Dispersion data	
Distance to EAB, meters	516
Distance to LPZ outer boundary, meters	3,140
Atmospheric dispersion data	5% level χ/Qs (refer to table 15B-4)
Dose data	
Method of dose calculation	Refer to appendix 15B
Dose conversion assumptions	Refer to appendix 15B

C. Loss of offsite power occurs concurrent with the opening of the atmospheric dump valve. At 1800 seconds the operator(s) takes control of the plant and conducts a cooldown using the atmospheric dump valve on the unaffected steam generator.

D. The atmospheric dump valve on one steam generator is assumed to be inadvertently opened.

E. The primary-to-secondary leakage of 1 gal/min (technical specification limit) is assumed to continue to the affected steam generator for the duration of the accident.

F. 1. No credit is assumed for auxiliary feedwater flow to the affected steam generator. This allows the affected steam generator to blow down (i.e. dry) prior to 1800 seconds. A post accident DF of 1 (iodines) was used for steam releases between the steam and water phase in the affected steam generator.

   2. A post-accident DF of 10 (iodines) was used for steam releases between the steam and water phase in the unaffected steam generator.

G. Calculated secondary mass releases are presented in table 15.1-7.

H. The activity released from the affected and unaffected steam generators is immediately vented to the atmosphere. No credit for radioactive decay in transit to dose point is assumed.

I. The mathematical model used to analyze the activity released during the course of the accident is described in appendix 15B.

J. The atmospheric dispersion factors used in this analysis, which are based on meteorological conditions assumed present during the course of the accident, are calculated according to the model described in section 2.3.4. The 5% level X/Qs presented in table 15B-4 were used.

K. The potential thyroid inhalation doses and beta-skin and whole body gamma immersion doses to an individual exposed at the exclusion area boundary (EAB) or outer boundary of the low population zone (LPZ) are analyzed using the models described in appendix 15B.

15.1.2.4.5.3 Identification of Uncertainties and Conservatisms in the Evaluation of the Results. The uncertainties and conservatisms in the assumptions used to evaluate the radiological consequences of an IOSGADV are as follows:

A. The RCS equilibrium activity is based on 1% failed fuel, which is greater by a factor of two to eight than that normally observed in past PWR operation.

B. The steam generator equilibrium activity for the affected steam generator is assumed to be equal to the technical specification limit (0.1 μCi/g dose equivalent I-131). This specific activity is greater by a factor of approximately 1300 than the normal expected steam generator activity (refer to table 11.1-21).

C. The assumption that the primary-to-secondary leakage of 1 gal/min (technical specification limit) is in the affected steam generator is conservative because:

   1. The 1 gal/min limit is applicable to both steam generators.

   2. Operation with a 1 gal/min primary-to-secondary leak is not expected.

D. The assumption of no auxiliary feedwater flow is conservative as it allows the affected steam generator to blowdown (i.e. dry). Consequently, all of the activity present in the affected steam generator is assumed to be released (DF of 1 between steam and water phases for iodines).

E. The atmospheric dump valve is assumed to be inadvertently opened to the full open position. Inadvertent opening of this valve to any other position results in less severe offsite doses. Additionally, this valve is administratively controlled from the control room to prevent inadvertent operation.

F. The meteorological conditions assumed to be present at the site during the course of the accident are based on 5% level X/Qs. Meteorological conditions will be less severe 95% of the time. This results in the poorest values of atmospheric dispersion calculated for the EAB or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the EAB or LPZ outer boundary.

G. The assumption of no operator action for 1800 seconds (30 minutes) is a conservative assumption.

H. The assumption that the atmospheric dump valve on the affected steam generator remains open for the duration of the accident is a conservative assumption. The atmospheric dump valve is provided with a manual operator and can therefore be shut independently from a electro-pneumatic malfunction.

15.1.2.4.5.4 <u>Conclusions</u>.

15.1.2.4.5.4.1 Filter Loadings. The only ESF filtration system considered in the analysis which limit the consequences of the inadvertent opening of a steam generator atmospheric dump valve (IOSGADV) is the control room filtration system. Activity loading on the control room filter has been based on the more serious loss-of-coolant accident. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to the postulated IOSGADV accident releases.

15.1.2.4.5.4.2 Dose to an Individual at the EAB and the Outer Boundary of the LPZ. The potential radiological consequences resulting from the occurrence of a postulated IOSGADV with a concurrent loss of offsite power have been conservatively analyzed, using assumptions and models described in previous sections.

The thyroid inhalation dose and the beta skin and whole body gamma doses due to immersion have been analyzed for the 0 to 2-hour period at the EAG and for the duration of the accident at the outer boundary of the LPZ. These results are listed in table 15.1-12.

Table 15.1-12
RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE WITH CONCURRENT LOSS OF OFFSITE POWER

Result	Offsite Dose
Exclusion Area Boundary Dose (0 to 2 hrs), rem	
Thyroid	2.4
Whole-body gamma	2.6×10^{-3}
Beta skin	2.1×10^{-3}
LPZ Outer Boundary Dose (duration), rem	
Thyroid	9.5×10^{-2}
Whole-body gamma	1.3×10^{-4}
Beta skin	1.3×10^{-4}

15.1.3 LIMITING FAULTS

15.1.3.1 Steam System Piping Failures

15.1.3.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a steam line break classifies it as a limiting fault as defined in reference 1 of section 15.0. A steam line break is defined as a pipe break in the main steam system.

15.1.3.1.2 Sequence of Events and Systems Operation

The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator, and subsequently the reactor coolant system (RCS), which results in a reduction of the reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The reactor trips, which may occur due to a steam line break, assuming no loss of offsite ac power, are low steam generator pressure, low steam generator water level, and high linear power level. For cases that assume a concurrent loss of offsite ac power, a reactor trip may also be caused by a low DNBR trip initiated by the core protection calculators. For any reactor trip, the control rod assembly of maximum worth is conservatively assumed to be held in the fully withdrawn position. In all cases, a low steam generator pressure signal would also initiate a main steam isolation signal (MSIS) which begins closure of the main steam isolation valves (MSIV) and main feedwater isolation valves (MFIV). The reduction of the RCS pressure empties the pressurizer and initiates a safety injection actuation signal (SIAS). The emptying of the steam generator associated with the ruptured steam line and the initiation of safety injection boron causes the core reactivity to decrease. A parametric review of the single failures that could occur during the SLB transient has determined that the failure of one of the high-pressure safety injection (HPSI) pumps to start subsequent to the SIAS has the most adverse effect. Consequently, one HPSI pump is conservatively assumed to fail. The operator, via the appropriate emergency procedure, may initiate plant cooldown by manual control of the atmospheric steam dump valves, or the MSIV bypass valves associated with the intact steam generator, anytime after reactor trip occurs. This analysis presented herein conservatively assumes operator action is delayed until 30 minutes after first indication of the event. The plant is then cooled to 350F at which point shutdown cooling is initiated.

The sequence of events, following a steam line break until stabilization of the plant for three cases representing the most adverse potential for core damage before and after trip and the most adverse radiological consequences, are presented in tables 15.1-13, 15.1-14, and 15.1-15. They are respectively: (1) a full power, inside containment, double-ended steam line break with concurrent loss of offsite ac power; (2) a full power, inside containment, double-ended steam line break with no loss of offsite ac power; and (3) a hot zero power, outside containment, double-ended steam line break

Table 15.1-13
SEQUENCE OF EVENTS FOR A STEAM LINE BREAK AT FULL POWER INSIDE CONTAINMENT WITH DOUBLE ENDED RUPTURE OF THE STEAM LINE AND CONCURRENT LOSS OF OFFSITE AC POWER

Time (Seconds)	Event
0.0	Steam line break upstream of the main steam isolation valve initiated; loss of offsite ac power occurs
0.6	Low DNBR reactor trip signal generated by core protection calculators
0.75	Trip breakers open
1.05	Shutdown CEAs begin dropping into the core
2.2	Low steam generator pressure trip signal and MSIS initiated; main steam isolation valves begin to close; feedwater isolation valves begin to close
7.2	MSIVs closed
7.6	Pressurizer empties
16.6	Low RCS pressure initiates SIAS
22.2	MFIVs closed
27.6	High-pressure safety injection pump reaches full speed
60.0	Safety injection boron begins to reach core
107.5	Affected steam generator empties
770.0	Pressurizer liquid level re-established
1,800.0	Plant cooldown initiated by manual control of the atmospheric steam dump valves for the intact steam generator
14,630.0	Reactor coolant system temperature has dropped to point of initiation of shutdown cooling system.

Table 15.1-14
SEQUENCE OF EVENTS FOR A STEAM LINE BREAK AT FULL POWER INSIDE CONTAINMENT WITH DOUBLE ENDED RUPTURE OF THE STEAM LINE AND NO LOSS OF OFFSITE AC POWER

Time (Seconds)	Event
0.0	Steam line break upstream of the main steam isolation valve initiated
2.2	Low steam generator pressure trip signal and MSIS initiated; main steam isolation valves begin to close; feedwater isolation valves begin to close
2.6	Trip breakers open
2.9	Shutdown CEAs begin dropping into the core
7.2	MSIVs closed
7.4	Pressurizer empties
12.7	Low RCS pressure initiates SIAS
22.2	MFIVs closed
23.7	High pressure safety injection pump reaches full speed
50.0	Safety injection boron begins to reach core
52.4	Affected steam generator empties
1,800.0	Plant cooldown initiated by manual control of the atmospheric dump valves for the intact steam generator
13,240.0	Reactor coolant system temperature has dropped to point of initiation of shutdown cooling system

Table 15.1-15
SEQUENCE OF EVENTS FOR A STEAM LINE BREAK AT HOT ZERO POWER OUTSIDE CONTAINMENT WITH CONCURRENT LOSS OF OFFSITE AC POWER AND BLOWDOWN RESTRICTED BY FLOW VENTURI IN THE MAIN STEAM LINE

Time (Seconds)	Event
0.0	Steam line break upstream of main steam isolation valve initiated; loss of offsite ac power occurs
4.3	Low steam generator pressure trip signal and MSIS initiated; main steam isolation valves begin to close; feedwater isolation valves begin to close
4.7	Trip breakers open
5.0	Shutdown CEAs begin dropping into the core
9.3	MSIVs closed
15.4	Pressurizer empties
17.8	Low RCS pressure initiates SIAS
24.3	MFIV's closed
28.8	High pressure safety injection pump reaches full speed
50.0	Safety injection boron begins to reach the core
539.4	Affected steam generator empties
1,800.0	Plant cooldown initiated by manual control of the atmospheric steam dump valves associated with the intact steam generator
16,920.0	Reactor coolant system temperature has dropped to the point of initiation of shutdown cooling system

with concurrent loss of offsite ac power. For the third case, the blowdown rate of the steam generator is restricted by the flow venturis located in the steam lines and the transient is conservatively assumed to be initiated shortly after shutdown from full power.

15.1.3.1.3 Core and System Performance

15.1.3.1.3.1 Mathematical Model. The NSSS response to a steam line break was simulated using the CESEC computer program described in section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with the CE-1 CHF correlation described in chapter 4.

15.1.3.1.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used to analyze the NSSS response to a steam line break representing the most adverse potential for core damage before and after trip are listed in table 15.1-16. The initial conditions for the principal process variables monitored by the COLSS were varied within the reactor operating space given in table 15.0-4 to determine the set of conditions that produce the most adverse consequences following a steam line break. Various combinations of initial core inlet temperature, core inlet flowrate, pressurizer pressure, and axial power distribution were considered. Variation of the initial RCS pressure and axial power distribution had only minor effects upon the transient. Decreasing the core inlet flow initiates the transient at a higher average coolant temperature and produces a larger cooldown of the reactor coolant; consequently, causing a larger reactivity increase due to the moderator reactivity function. Increasing the core inlet temperature produces a moderator cooldown over a more adverse portion of the moderator reactivity function, resulting in a larger reactivity increase during cooldown. Previous parametric analyses performed for the Arkansas Nuclear One Unit 2 FSAR steam line break accident indicate that variations in the decay heat rate assumed as large as 20% have an insignificant effect on the initial consequences. Variation of the core inlet temperature produced the most significant effect on the transient. The set of initial operating conditions that yields the most adverse consequences following a steam line break are the minimum pressurizer pressure, the minimum core inlet flowrate, the maximum core inlet temperature, and the most top peaked axial power shape allowed by the operating space given in table 15.0-4.

In addition, various assumptions as to the time of loss of offsite ac power and the location of the steam line break inside and outside of containment were analyzed. A range of break sizes were considered to determine the size which resulted in the most severe potential fuel damage. A flow venturi in each main steam line restricts the blowdown rate of the appropriate steam generator for outside containment breaks. A loss of offsite ac power coincident with the steam line break for a guillotine break of a main steam line inside containment represents the most adverse potential for fuel damage.

Table 15.1-16
ASSUMPTIONS FOR A STEAM LINE BREAK AT FULL POWER INSIDE
CONTAINMENT WITH DOUBLE ENDED RUPTURE OF THE
STEAM LINE

Parameter	Assumption
Initial core power level, Mwt	3478
Core inlet coolant temperature, ^{o}F	560
Core mass flowrate, 10^{6} lbm/h	132.2
Reactor coolant system pressure, lb/in.^{2}a	2,000
One pin radial peaking factor, with uncertainty	1.3
Initial core minimum DNBR	1.29
Steam generator pressure, lb/in.^{2}a	949
Doppler coefficient multiplier	1.15
Moderator coefficient multiplier	1.10
CEA worth for trip, 10^{-2} $\Delta\rho$	-8.55
Steam bypass control system	Inoperative
Pressurizer pressure control system	Inoperative
High pressure safety injection pumps	One pump inoperative
Core burnup	End of first cycle
Blowdown fluid	100% steam
Break area, ft^{2}	7.41

Conservative assumptions regarding initial plant conditions and postulated system failures include:

A. End-of-cycle core conditions to yield the most negative moderator temperature coefficient, void coefficient, and Doppler coefficient.

B. Loss of offsite ac power to the plant at the most adverse time. The most adverse time for the loss of offsite ac power to occur was found to be coincident with the steam line break.

C. The CEA of maximum worth stuck in the fully withdrawn position after reactor trip.

D. A failure of one HPSI pump as the worst single active component failure.

E. Feedwater flow at the start of the transient coresponds to initial steady-state operation.

Feedwater flow is automatically reduced from 100% to zero% in 20 seconds following low steam generator pressure trip by closure of the feedwater isolation valves.

Conservative assumptions regarding parameters used in the analysis include:

A. 100% quality steam with no moisture carryover during the steam generator blowdown to yield the maximum energy removal.

B. A 15% increase for the slope of the Doppler reactivity versus fuel temperature function to assure that the calculation of the reactivity increase due to cooldown of the fuel from its nominal temperature is conservative.

C. A 10% increase for the slope of the moderator reactivity versus coolant temperature function to assure that the calculation of the reactivity increase due to cooldown of the moderator is conservative.

D. Moderator reactivity as a function of the lowest cold leg temperature to account conservatively for the effect of uneven temperature distribution on the moderator reactivity.

E. No allowance for the void reactivity feedback associated with local boiling in the hot channel.

F. Zero mixing of the reactor coolant in the lower plenum of the reactor core.

Assumptions considered as to the worst single active component failure included:

A. Failure of one HPSI pump to start after SIAS.

B. Failure of one main feedwater isolation valve to close after MSIS.

C. Failure of one main steam isolation valve to close after MSIS.

D. Failure of the turbine stop valves to close after reactor trip.

E. Failure of one diesel generator to start after loss of offsite ac power.

The worst single active component failure was the failure of one HPSI pump to start, delaying the time for safety injection boron to reach the reactor core.

15.1.3.1.3.3 Results

A. The dynamic behavior of the salient NSSS parameters, following a double ended steam line break inside containment at full power with a concurrent loss of offsite ac power, is presented in figures 15.1-33 through 15.1-45. This case represents the most adverse potential for fuel damage before reactor trip.

Concurrent with a steam line break, a loss of offsite ac power occurs. At this time, an actuation signal for the emergency diesel generators is initiated and, since the NSSS is conservatively assumed to initially be at a COLSS limit, conditions exist for a low DNBR trip. At 0.6 seconds a low DNBR trip signal is initiated by the core protection calculators. At 0.75 seconds the reactor trip breakers open. After a 0.3 second coil decay delay, the CEAs begin dropping into the core at 1.05 seconds. At 2.2 seconds, the steam generator pressure drops below the low steam generator pressure trip setpoint of 675 $lb/in.^2a$ and initiates an MSIS. The MSIS begins closure of the main steam isolation valves and the feedwater isolation valves. The MSIVs close at 7.2 seconds. At 7.6 seconds the pressurizer empties. At 15.0 seconds the diesel generators reach full speed and voltage. At 16.6 seconds the RCS pressure drops below the setpoint of 1560 $lb/in.^2a$ and initiates an SIAS. At 22.2 seconds the MFIVs close. The HPSI pump reaches full speed at 27.6 seconds, and safety injection boron begins to reach the core at 60 seconds. At 107.5 seconds the steam generator associated with the ruptured steam line empties. At 186.3 seconds, the core reactivity begins to decrease. At 770 seconds, the pressurizer liquid level is re-established. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the atmospheric steam dump valves, assuming that offsite ac power has not been restored. At approximately 4 hours, the RCS reaches 350F at which time shutdown cooling is initiated.

The maximum RCS pressure does not exceed 110% of design pressure following a steam line break, thus assuring the integrity of the RCS. The minimum DNBR does not violate the SAFDL DNBR of 1.19.

B. The dynamic behavior of the salient NSSS parameters following a double-ended steam line break inside containment at full power with no loss of offsite ac power is presented in figures 15.1-46 through 15.1-58. This case represents the most adverse potential for fuel damage due to a possible return-to-power after reactor trip.

At 2.2 seconds after initiation of the steam line break, the affected steam generator pressure drops below the low steam generator pressure trip setpoint of 675 $lb/in.^2a$ and initiates a MSIS. The MSIVs close at 7.2 seconds. At 7.4 seconds, the pressurizer empties. At 12.7 seconds, the RCS pressure drops below the low pressurizer pressure trip setpoint of 1560 $lb/in.^2a$ and initiates a SIAS. At 22.2 seconds the MFIVs close. The HPSI pump reaches full speed at 23.7 seconds, and safety injection boron begins to reach the core at 50.0 seconds. At 55 seconds, a peak return-to-power of 9.9%, which is only 5% above the power level that would have existed if the total reactivity had not increased during the transient. At 52.4 seconds, the steam generator associated with the ruptured steam line empties. At 55.8 seconds, the core activity begins to decrease. At a maximum of 30 minutes, the operator, via the appropriate emergency procedures, initiates plant cooldown by manual control of the MSIV bypass valves associated with the intact steam generator. At approximately 4 hours, the RCS reaches 350F at which time shutdown cooling is initiated.

The maximum RCS pressure does not exceed 110% of design pressure following a steam line break, thus assuring the integrity of the RCS. The minimum DNBR does not violate the SAFDL DNBR of 1.19.

15.1.3.1.4 Barrier Performance

15.1.3.1.4.1 <u>Mathematical Model</u>. The mathematical model used for evaluation of Barrier Performance is identical to that described in paragraph 15.1.3.1.3.

15.1.3.1.4.2 <u>Input Parameters and Initial Conditions</u>. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.1.3.1.3 with the exception of any parameters listed in table 15.1-17 and any assumptions listed below.

The most adverse mass release and radiological consequences following a steam line break occur for a double-ended steam line break outside containment at hot zero power conditions with a concurrent loss of offsite ac power where the transient is initiated shortly after a shutdown from full power. The hot zero power conditions assure the maximum water inventory in the steam generators, and the shutdown from full power assures the maximum decay heat which must be removed by manual control of the atmospheric steam dump valves associated with the intact steam generator,

Table 15.1-17
ASSUMPTIONS FOR A STEAM LINE BREAK AT HOT ZERO POWER OUTSIDE CONTAINMENT WITH CONCURRENT LOSS OF OFFSITE AC POWER AND BLOWDOWN RESTRICTED BY FLOW VENTURI IN THE MAIN STEAM LINE

Parameter	Assumption
Initial core power level, Mwt	1.0
Core inlet coolant temperature, ^{o}F	542
Core mass flowrate, 10^6 lbm/h	132.2
Reactor coolant system pressure, $lb/in.^2a$	2,000
Initial core minimum DNBR	1.29
Steam generator pressure, $lb/in.^2a$	1003
Doppler coefficient multiplier	0.85
Moderator coefficient multiplier	1.10
CEA worth for trip, 10^{-2} $\Delta\rho$	-4.45
Pressurizer pressure control system	Inoperative
High-pressure safety injection pumps	One pump inoperative
Core burnup	End of first cycle
Blowdown fluid	100% steam
Blowdown area, ft^2	4.13
Decay heat	Full power

assuming that offsite ac power cannot be restored before or during the cooldown period. For the outside containment steam line break case, the blowdown rate of the steam generators is restricted by the presence of flow venturis located in the main steam lines.

Assumptions regarding initial plant conditions different from those of paragraph 15.1.3.1.3 include: (1) no load on the steam turbine and consequently a larger initial mass inventory in the steam generators; (2) a hot zero power core inlet temperature of 545F; and (3) feedwater flow is assumed to match energy input by the reactor coolant pumps and the 1 MWt core power.

Conservative assumptions regarding the hot zero power steam line break analysis different from those of paragraph 15.1.3.1.3 include the use of full power decay heat versus time curve for calculation of the energy to be removed from the RCS during plant cooldown.

15.1.3.1.4.3 Results. The dynamic behavior of the salient NSSS mass release parameters following a double-ended steam line break outside of containment at hot zero power, shortly after shutdown from full power, with a concurrent loss of offsite ac power and blowdown of the steam generators restricted by a flow venturi in each main steam line, is presented in figures 15.1-59 through 15.1-61. This case maximizes the mass releases and radiological consequences to the environment.

At a maximum 30 minutes after initiation of the steam line break, the operator, via the appropriate emergency procedures, begins plant cooldown by manual control of the atmospheric steam dump valves, assuming that offsite ac power has not been restored. At this time, no more than 308,600 pounds of steam with a decontamination factor (DF) of 1.0 will have been discharged through the steam line break. Approximately 1,120,000 pounds of steam with a DF of 10 will have been discharged through the atmospheric steam dump valves associated with the intact steam generator during the 4.2-hour cooldown of the plant to a reactor coolant temperature of 350F. The primary-to-secondary leakage to the steam generator associated with the ruptured steam line is conservatively assumed to be the entire 1 gal/min (i.e., the technical specification value). The total steam released to the environment will have been approximately 1,430,000 pounds.

15.1.3.1.5 Radiological Consequences

15.1.3.1.5.1 Design Basis - Method of Analysis - No Iodine Spike.

15.1.3.1.5.1.1 Design Basis - Physical Model (No Iodine Spike). To evaluate the radiological consequences due to a postulated main steam line break (outside containment), it is assumed that there is a complete severance of a main steam line outside the containment with the plant in a hot zero power condition where the transient is initiated shortly after full-power operation. It is also assumed that there is a simultaneous loss of offsite power. The hot zero power condition assures the maximum water inventory in the steam generators and the shutdown from full power (in conjunction with the loss of offsite power) assures the maximum decay heat which must be removed by manual control of the atmospheric dump valve associated with the intact steam generator.

The main steam isolation valves are installed in the main steam lines from each steam generator, downstream from the safety relief valves and atmospheric dump valves outside containment. The severance of the main steam line is assumed to be upstream of the main steam isolation valve. A reactor trip is actuated by a low steam generator pressure signal. A main steam isolation signal (MSIS) is actuated to shut the main steam isolation valves from both steam generators. The affected steam generator (steam generator connected to the severed steam line) blows down completely. The steam is vented directly to the atmosphere. The atmospheric dump valve of the unaffected steam generator is used to initiate a 75F/hr cooldown of the reactor coolant system 1800 seconds after initiation of the accident. The steam is vented directly to the atmosphere. Mass release from the unaffected steam generator is terminated when the shutdown cooling system is initiated at a reactor coolant system temperature of 350F.

The sequence of events for this accident is presented in table 15.1-15.

15.1.3.1.5.1.2 Design Basis (No Iodine Spike) - Assumptions, Parameters, and Calculational Methods. The major assumptions, parameters, and calculational methods used in the design basis analysis are presented in table 15.1-18. Additional clarification is provided as follows:

A. Reactor coolant activity

The reactor coolant equilibrium activity is based on long term operation at 105% of the ultimate core power level of 3390 MWt (3390 MWt x 1.05 = 3560 MWt) and 1% failed fuel. Source terms are listed in table 11.1-2. Reactor coolant activity does not increase after the accident.

B. Secondary system activity

The activity in both steam generators is conservatively assumed to be equal to 0.1 μCi/g dose equivalent Iodine-131 (I-131). This activity is the technical specification limit presented in chapter 16.

C. Primary-to-secondary leakage

The primary to secondary leakage of 1 gal/min (technical specification limit) was assumed to continue through the affected steam generator at a constant rate until the reactor coolant system temperature reaches 212F. The calculated time until this temperature is reached is 23,635 seconds.

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 1 of 7)

Parameter	Design Basis Assumptions	Realistic Assumptions
Data and assumptions used to estimate radioactive source		
General		
Power level, MWt	1	1
Burnup	End of cycle	End of cycle
Percent of fuel perforated	0	0
Reactor coolant activity before accident		
No iodine spike	Table 11.1-2	Table 11.1-3
Coincident (existing) iodine spike	60 μCi/g dose equivalent I-131	No spike
Iodine spike caused by accident	Table 11.1-2	No spike
Reactor coolant activity after accident		
No iodine spike	Table 11.1-2	Table 11.1-3
Coincident (existing) iodine spike	60 μCi/g dose equivalent I-131	No spike
Iodine spike caused by accident	Figure 15.1-62	No spike
Steam generator activity before accident	0.1 μCi/g dose equivalent I-131 (technical specification limit)	Table 11.1-21 (normal case)

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 2 of 7)

Parameter	Design Basis Assumptions		Realistic Assumptions	
Secondary mass inventory, lb_m				
Liquid	260,380		260,380	
Steam	9,814		9,814	
Activity release down steam generators				
No iodine spike, Ci				
Isotope	0-2 hours	2-hour duration	0-2 hours	2-hour duration
I-131	24.79	30.82	1.66×10^{-1}	5.07×10^{-1}
I-132	4.31	5.72	3.74×10^{-2}	1.12×10^{-1}
I-133	23.7	30.48	1.88×10^{-1}	5.82×10^{-1}
I-134	0.48	.933	1.61×10^{-2}	5.26×10^{-2}
I-135	6.51	9.05	7.79×10^{-2}	2.46×10^{-1}
Xe-131m	0.84	2.66	3.60×10^{-2}	1.18×10^{-1}
Xe-133	118	373.73	5.89	19.3
Xe-135m	0.39	1.23	4.25×10^{-3}	1.40×10^{-2}
Xe-135	3.26	10.32	1.15×10^{-1}	3.76×10^{-1}
Xe-138	0.20	0.63	1.44×10^{-2}	4.72×10^{-2}
Kr-85m	0.83	2.62	7.20×10^{-2}	2.36×10^{-1}
Kr-85	1.81	5.74	4.91×10^{-2}	1.61×10^{-1}
Kr-87	0.44	1.4	1.96×10^{-2}	6.44×10^{-2}
Kr-88	1.44	4.55	6.54×10^{-2}	2.15×10^{-1}

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 3 of 7)

Parameter	Design Basis Assumptions	Realistic Assumptions
Coincident (existing) iodine spike, Ci		
Isotope	0-2 hour	
I-131	37.29	No iodine spike
I-132	7.84	No iodine spike
I-133	39.43	No iodine spike
I-134	2.01	No iodine spike
I-135	13.42	No iodine spike
Xe-131m	0.84	No iodine spike
Xe-133	118	No iodine spike
Xe-135m	0.39	No iodine spike
Xe-135	3.26	No iodine spike
Xe-138	0.20	No iodine spike
Kr-85m	0.83	No iodine spike
Kr-85	1.81	No iodine spike
Kr-87	0.44	No iodine spike
Kr-88	1.44	No iodine spike

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 4 of 7)

Parameter	Design Basis Assumptions	Realistic Assumptions
Iodine spike caused by accident, Ci		
Isotope	0-2 hour	
I-131	53.5	No iodine spike
I-132	12.43	No iodine spike
I-133	59.08	No iodine spike
I-134	4.0	No iodine spike
I-135	22.07	No iodine spike
Xe-131m	0.84	No iodine spike
Xe-133	118	No iodine spike
Xe-135m	0.39	No iodine spike
Xe-135	3.26	No iodine spike
Xe-138	0.20	No iodine spike
Kr-85m	0.83	No iodine spike
Kr-85	1.81	No iodine spike
Kr-87	0.44	No iodine spike
Kr-88	1.44	No iodine spike

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 5 of 7)

Parameter	Design Basis Assumptions	Realistic Assumptions
Data and assumptions used to estimate activity released.		
General		
Loss of offsite power	Yes	Yes
Credit for radioactive decay in transit to dose point	No	No
Affected steam generator		
Primary-to-secondary leakage rate, lb/d	8,640 (1 gal/min)	100
Secondary mass release to atmosphere (through severed line), lb_m	270,387	270,287
Mass of primary-to-secondary leakage (integrated for 23,635 seconds when RCS temperature reaches 212F), lb_m	2,363.5	2,363.5
Steam generator decontamination factor between steam and water phase	1	1
Unaffected steam generator		
Primary-to-secondary leakage rate, lb/d	0	0

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 6 of 7)

Parameter	Design Basis Assumptions	Realistic Assumptions
Secondary mass release to atmosphere, lb_m		
Through severed line before main steam isolation valve is shut	38,284	38,284
Through steam dump (integrated for 16,920 seconds when shutdown cooling initiated)	1,120,392	1,120,392
Steam generator decontamination factor between steam and water phases		
Through severed line		
Noble gases	1	1
Iodines	1	1
Through steam dump		
Noble gases	1	1
Iodines	10	10
Dispersion data		
Distance to EAB, meters	576	576
Distance to LPZ outer boundary, meters	3,140	3,140
Atmospheric dispersion factors	5% level X/Qs (table 15B-4)	50% level X/Qs (table 15B-4)

Table 15.1-18
PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES
OF A MAIN STEAM LINE BREAK ACCIDENT (MSLBA) (Sheet 7 of 7)

Parameter	Design Basis Assumptions	Realistic Assumptions
Dose data		
Method of dose calculation	Refer to appendix 15B	
Dose conversion assumptions	Refer to appendix 15B	

D. Secondary releases to atmosphere

The calculated steam releases from the affected and unaffected steam generators are presented in table 15.1-7.

15.1.3.1.5.1.3 Design Basis - Identification of Uncertainties and Conservatisms in the Evaluation of the Results (No Iodine Spike)

A. Reactor coolant equilibrium activities are based on 1% failed fuel, which is greater by a factor of two to eight than that normally observed in past PWR operation.

B. An 8640 lb_m/d (1 gal/min) steam generator primary-to-secondary leakage is assumed, which is greater by a factor of 50 to 200 than that normally observed in past PWR operation.

C. The steam generator equilibrium activity for both steam generators is assumed to be equal to the technical specification limit (0.1 μCi/g dose equivalent I-131) for the duration of the accident. This specific activity is greater than the normal steam generator equilibrium activity (refer to table 11.1-21) by a factor of approximately 1300.

D. The meteorological conditions assumed to be present at the site during the course of the accident are based on X/Q values which are expected to be conservative 95% of the time. This condition results in the poorest values of atmospheric dispersion calculated for the exclusion area boundary or LPZ outer boundary. Furthermore, no credit has been taken for the transit time required for activity to travel from the point of release to the exclusion area boundary or LPZ outer boundary. Hence, the radiological consequences evaluated under these conditions will be conservative.

E. A conservative steam generator decontamination factor (DF) of 10 is used in the cooldown phase (release to atmospheric dump valve).

15.1.3.1.5.1.4 Design Basis - Conclusions (No Iodine Spike)

A. Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the main steam line break is the control room filtration system. Activity loadings on the control room charcoal filter are based on the flowrate through the filter, the concentration of activity at the filter inlet, and the filter efficiency.

Activity loading on the control room filter has been designed for the more serious LOCA. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated main steam line break accident releases.

B. Dose to an Individual at the Exclusion Area Boundary and the Outer Boundary of the Low Population Zone.

The potential radiological consequences resulting from the occurrence of a postulated main steam line break have been conservatively analyzed, using assumptions and models described in previous sections.

The beta-skin and the total body gamma dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0 to 2 hour dose at the exclusion area boundary and for the duration of the accident at the outer boundary of the low-population zone. The results are listed in table 15.1-19. The resultant doses are small fractions of the guideline values of 10CFR100.

15.1.3.1.5.2 <u>Design Basis</u> - Coincident (Existing) Iodine Spike and Main Steamline Break. In this evaluation, a case with a coincident iodine spike which already exists due to a previous power transient was considered. The mathematical models, assumptions, and parameters used in this analysis were identical with the design basic main steamline break accident without an iodine spike as described in paragraph 15.1.3.1.5.1 with the following exception:

The reactor coolant system inventory was assumed to be 60 μCi/g dose equivalent Iodine 131 vice the reactor coolant inventory shown in table 11.1-2 which is based on 105% of design core power and 1% failed fuel. This 60 μCi/g is the technical specification limit (section 16.3/4.4-18) for full power operation following an iodine spike for periods of up to 48 hours. Radiological consequences are presented in table 15.1-19.

Table 15.1-19
RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED MAIN STEAM LINE BREAK (Sheet 1 of 2)

Result	Design Basis Value			Realistic Value
	No Iodine Spike	Coincident (Existing) Iodine Spike	Iodine Spike Caused by Accident	No Iodine Spike
Exclusion Area Boundary Dose (0 to 2 hours) rem:				
Thyroid	4.46	6.9	10.1	4.16×10^{-4}
Beta-skin	2.44×10^{-3}	3.36×10^{-3}	4.57×10^{-3}	9.11×10^{-7}
Total-body gamma	3.78×10^{-3}	6.35×10^{-2}	9.71×10^{-3}	8.13×10^{-7}
LPZ Outer Boundary Dose (duration), rem:				
Thyroid	1.59×10^{-1}			3.27×10^{-7}
Beta-skin	1.53×10^{-4}			7.6×10^{-7}
Total-body gamma	1.78×10^{-4}			6.7×10^{-7}

Table 15.1-19
RADIOLOGICAL CONSEQUENCES DUE TO A POSTULATED MAIN
STEAM LINE BREAK (Sheet 2 of 2)

Result	Design Basis Value	Realistic Value
Dispersion data		
Distance to EAB, meters	576	576
Distance to LPZ outer boundary, meters	3,140	3,140
Atmosphere dispersion factors	5% level X/Qs (table 15B-4)	50% level X/Qs (table 15B-4)
Dose data		
Method of dose calculation	Refer to appendix 15B	
Dose conversion assumptions	Refer to appendix 15B	
Control room design parameters	Refer to table 15B-5	

15.1.3.1.5.3 <u>Design Basis</u> - Iodine Spike Caused by the Main Steam Line Break. In this evaluation, a case with an iodine spike caused by the main steam line break accident was evaluated for radiological consequences. The mathematical models, assumptions, and parameters used in this analysis were identical with the design basis main steam line break accident without an iodine spike as described in paragraph 15.1.3.1.5.1 with the following exception:

Prior to the main steam line break accident the reactor coolant system activity is based on 105% of design power and 1% failed fuel. This reactor coolant inventory is the same as used in paragraph 15.1.3.1.5.1.2, listing A. However, at the initiation of the SGTR accident, the I-131 equivalent source term (released from fuel) is assumed to increase as shown in figure 15.1-62. This figure is based on the methods described in reference 1. The iodine release rate is assumed to increase by a factor of 500.

Radiological consequences are presented in table 15.1-19.

15.1.3.1.5.4 Realistic Analysis. A realistic analysis of the radiological consequences of a postulated main steam line break accident was performed. This analysis is identical with the evaluation presented in paragraph 15.1.3.1.5.1 with the following exceptions:

A. Reactor coolant system inventory is based on 0.12% failed fuel vice 1% failed fuel. Isotopic inventory is presented in table 11.1-3.

B. An iodine spike, pre-existing or caused by the accident, does not occur.

C. Steam generator equilibrium activity prior to the accident is based on a 100 lb/d and 0.12% failued fuel versus the technical specification limit. Steam generator activity is presented in table 11.1-21 (normal case).

D. 50% level X/Qs are used instead of 5% level X/Qs.

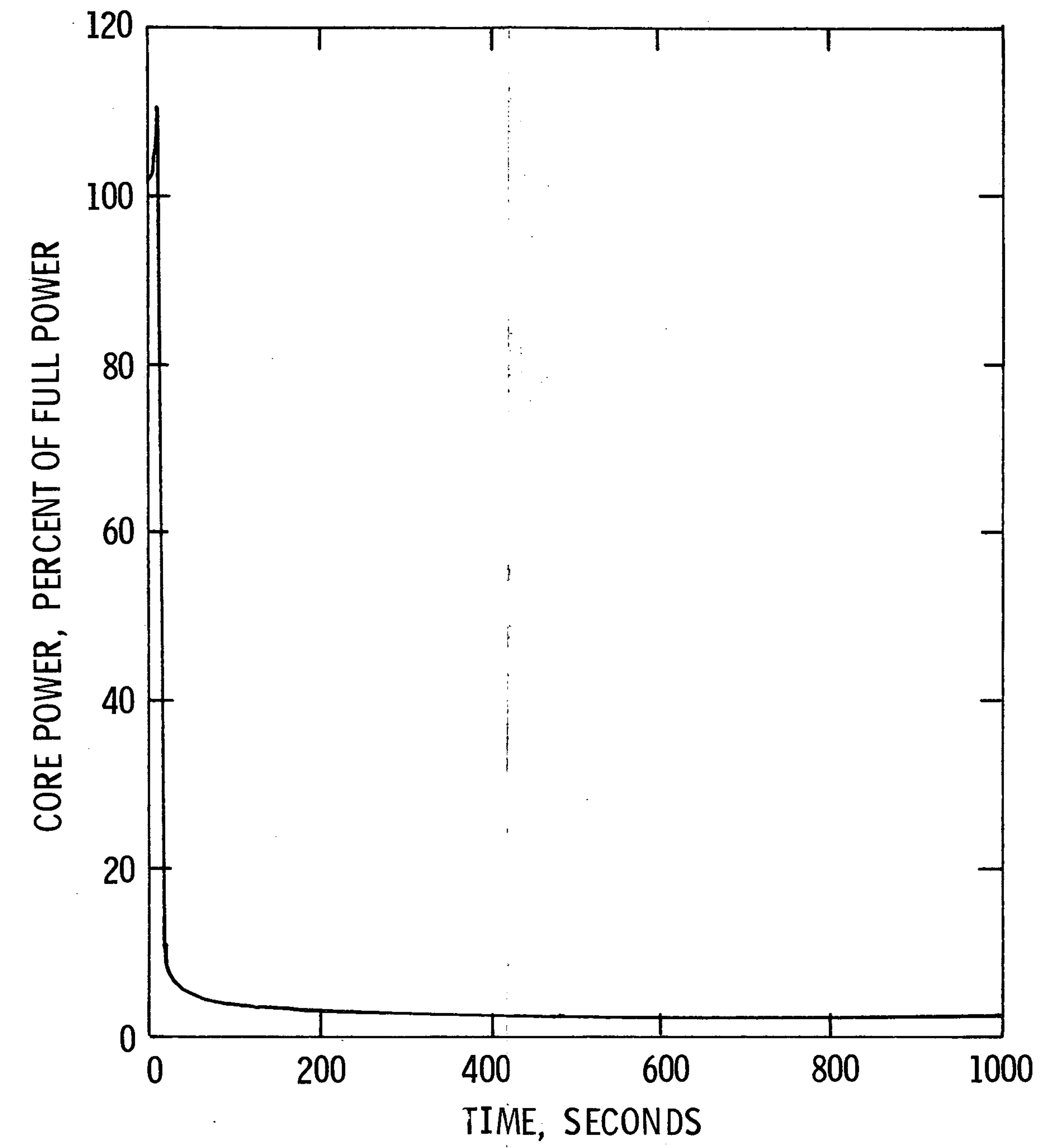
E. A post-accident DF of 100 was used between the water and steam phases versus 10 for the design basis case for the unaffected steam generator.

Major assumptions and parameters used in the realistic analysis are presented in table 15.1-18. The radiological consequences are presented in table 15.1-19.

A main steam line break accident is classified as a limiting fault. This accident is not expected to occur during the life of the plant but is postulated because the consequences of a main steam line break accident include the potential for the release of significant amounts of radioactive materials. The term "realistic analysis" as used in this section does not imply that the accident is expected to occur during the life of the plant. The term "realistic analysis" signifies that more realistic assumptions and parameters have been used to evaluate the radiological consequences of a limiting fault as defined by Revision 2 of Regulatory Guide 1.70.

REFERENCES

1. "Iodine Spiking - Radioiodine Behavior in the Reactor Coolant System During Reactor Operation," Combustion Engineering Inc., CENPD-180, April 1976.

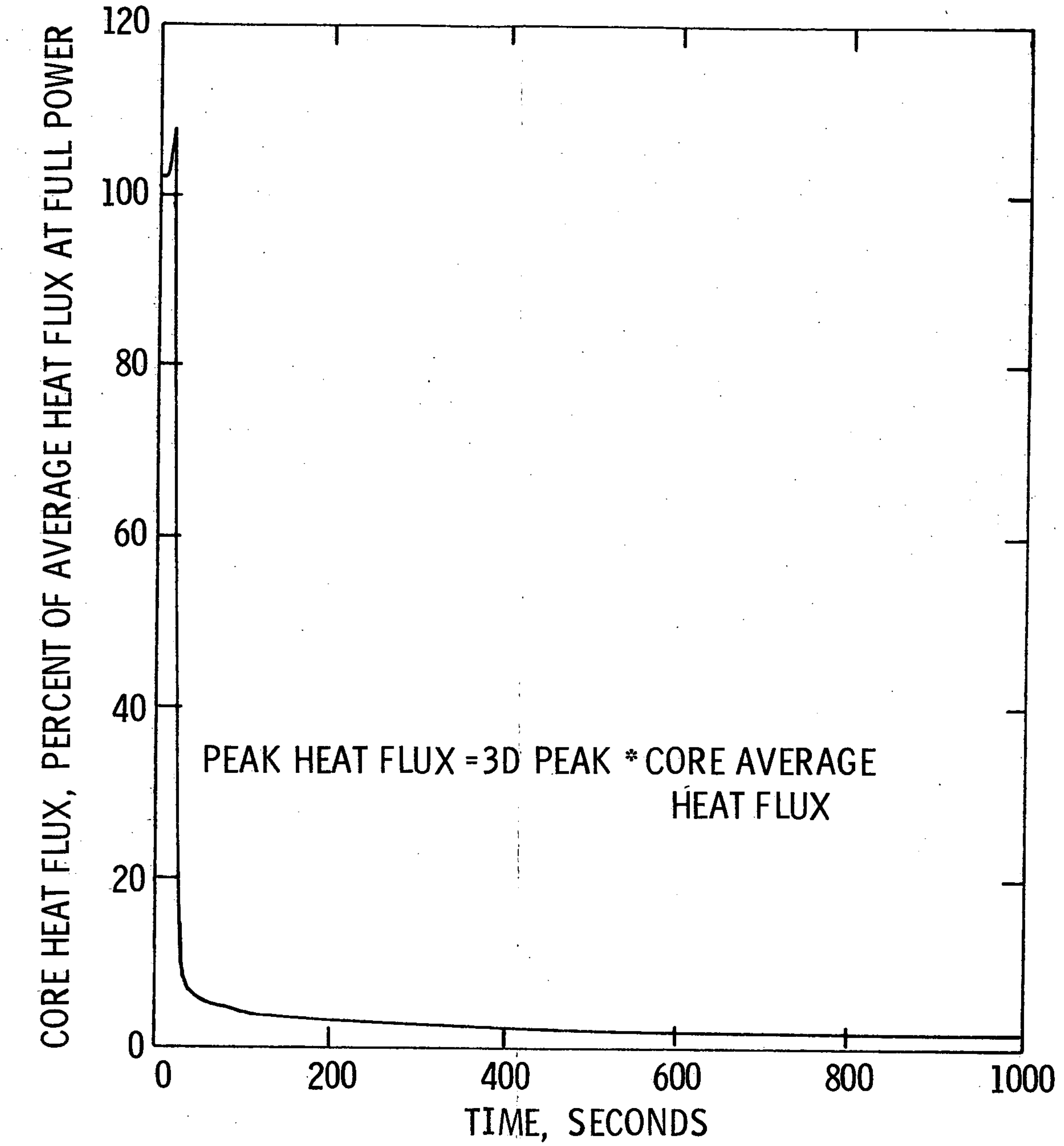


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
CORE POWER vs. TIME

Figure 15.1-1

10079-11D-921
200C6

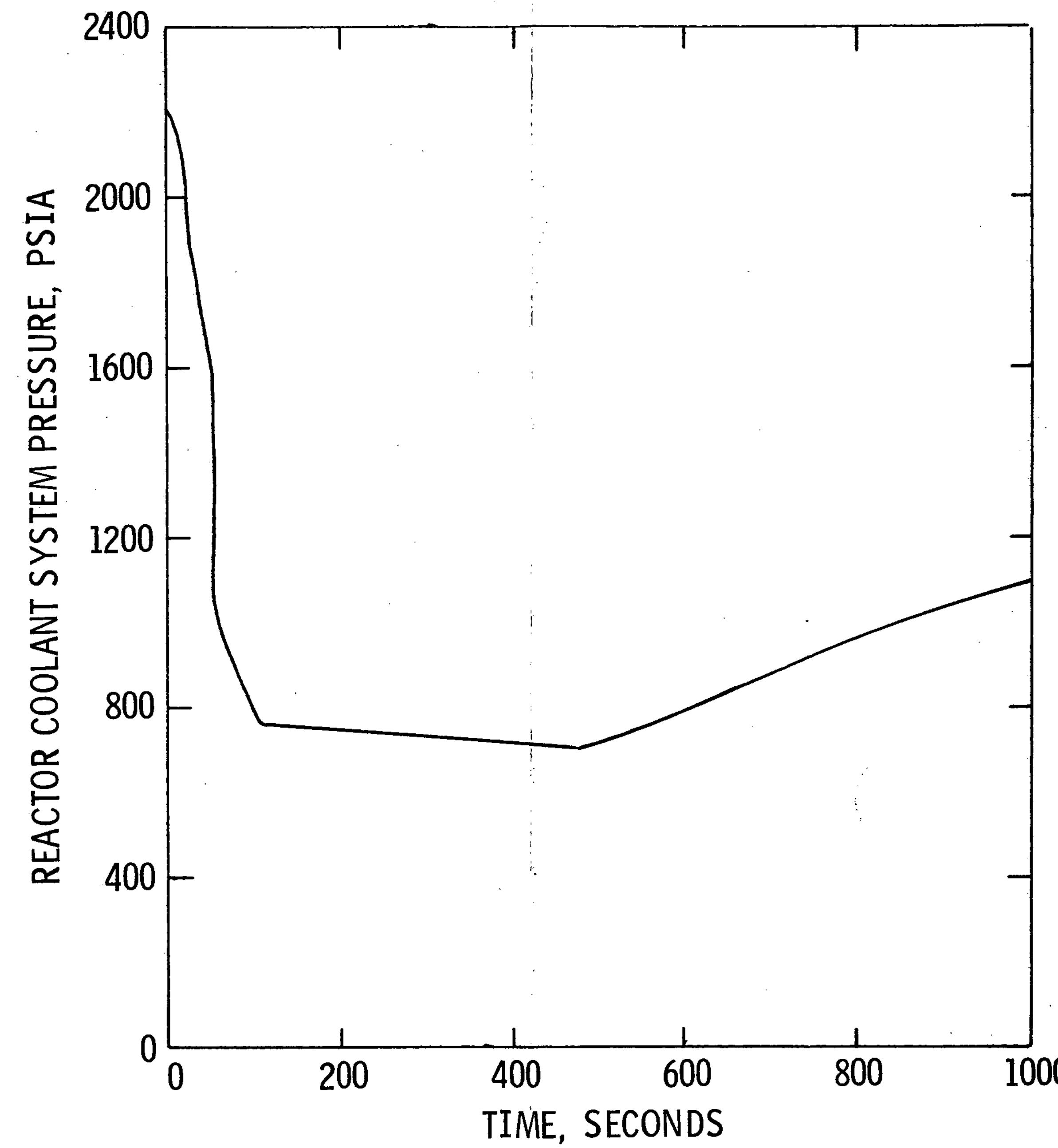


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
CORE HEAT FLUX vs. TIME

Figure 15.1-2

10079-11D-922
20OC6

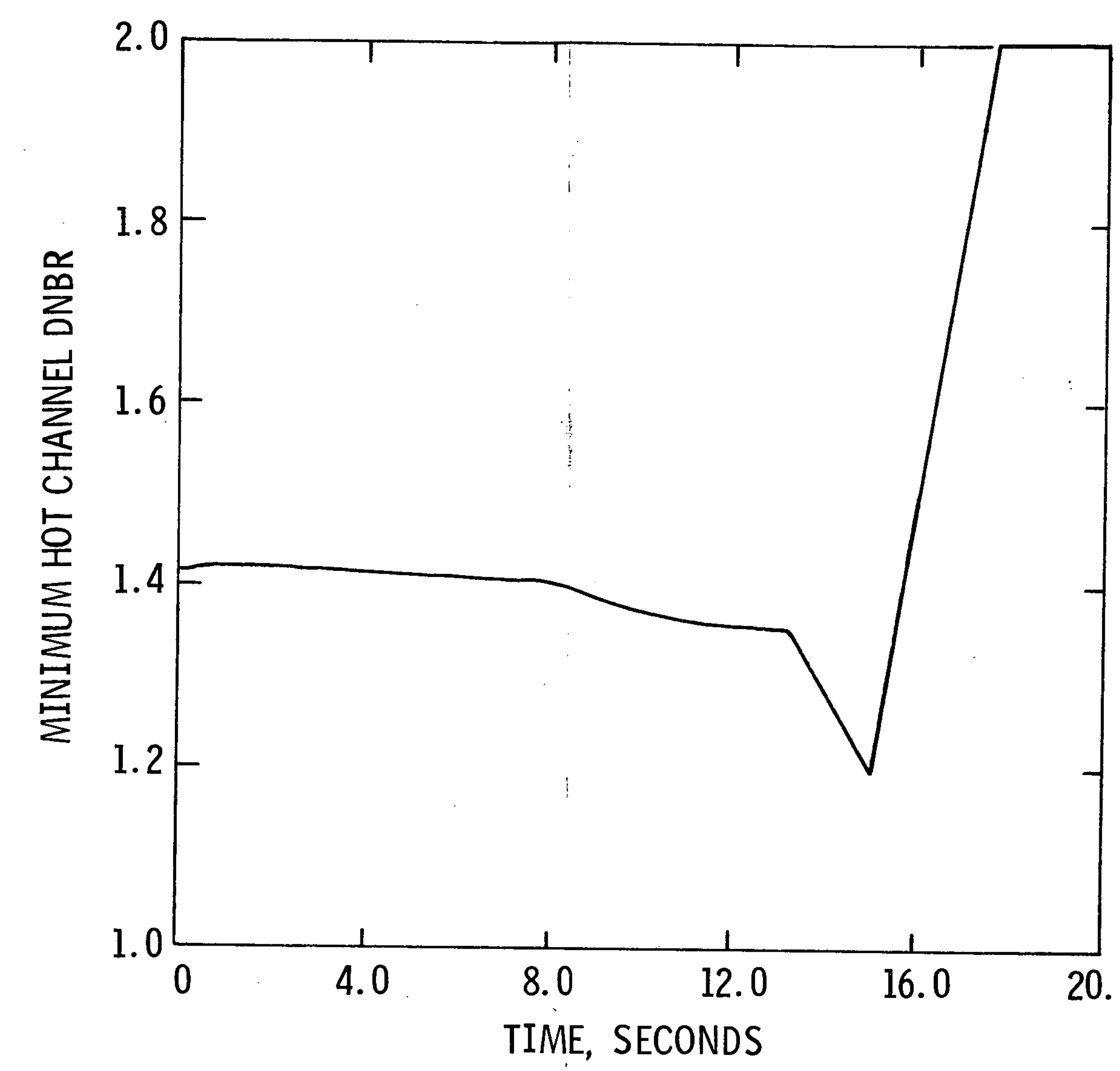


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW REACTOR COOLANT SYSTEM PRESSURE vs. TIME

Figure 15.1-3

10079-11D-923
20OC6

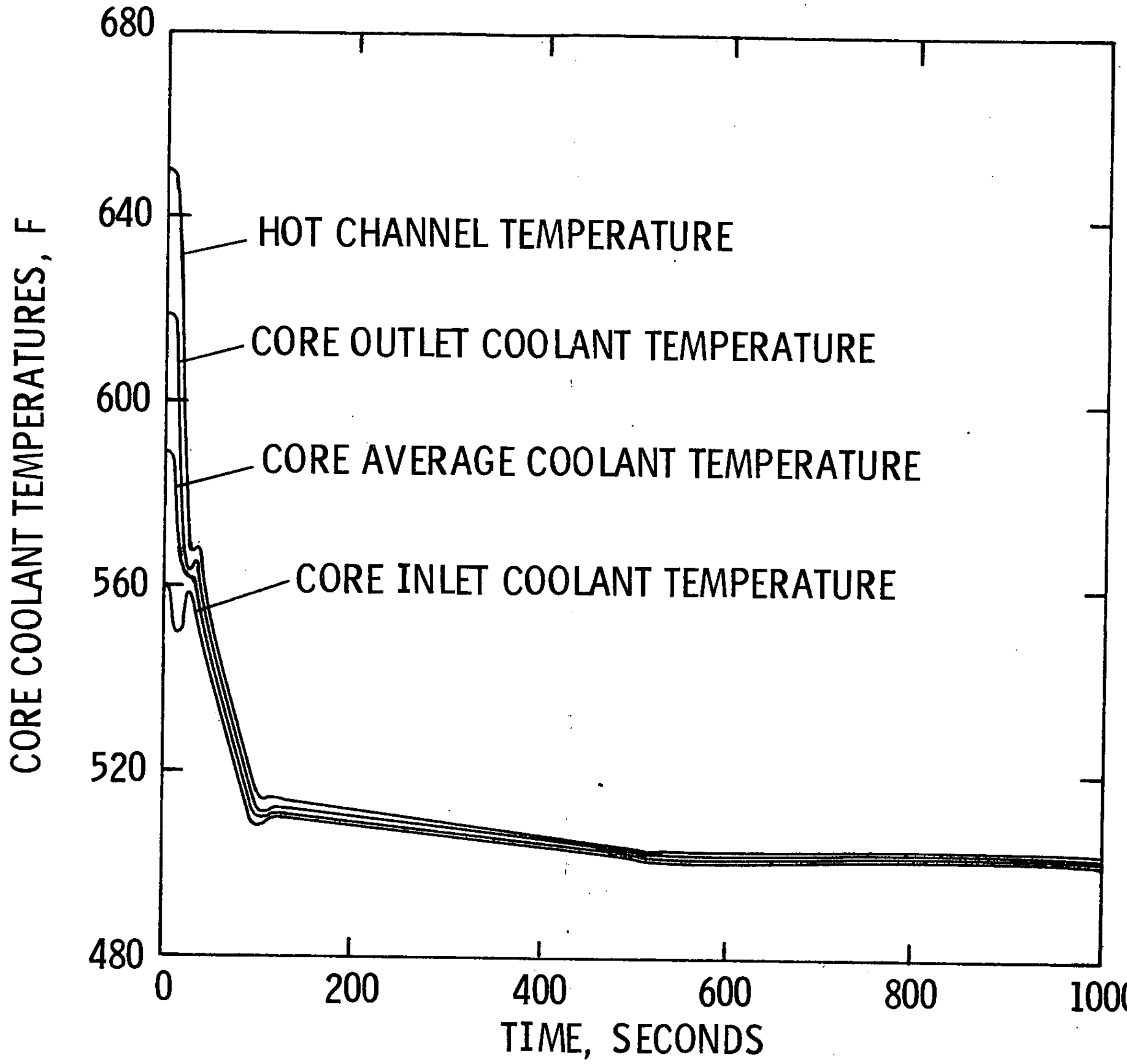


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW MINIMUM
HOT CHANNEL DNBR vs. TIME

Figure 15.1-4

10079-11D-924

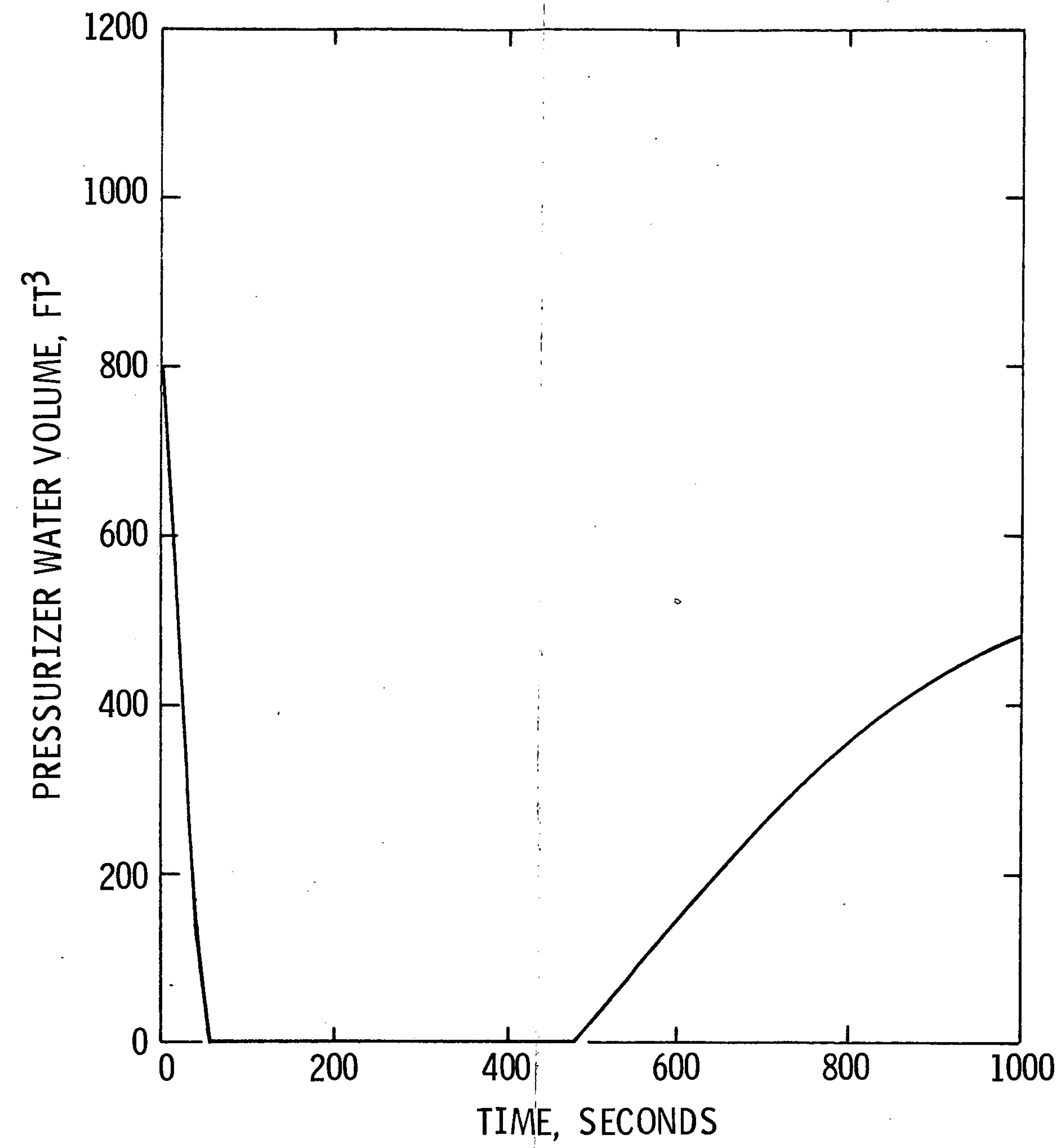


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW REACTOR COOLANT TEMPERATURE vs. TIME

Figure 15.1-5

10079-11D-925
20OC6

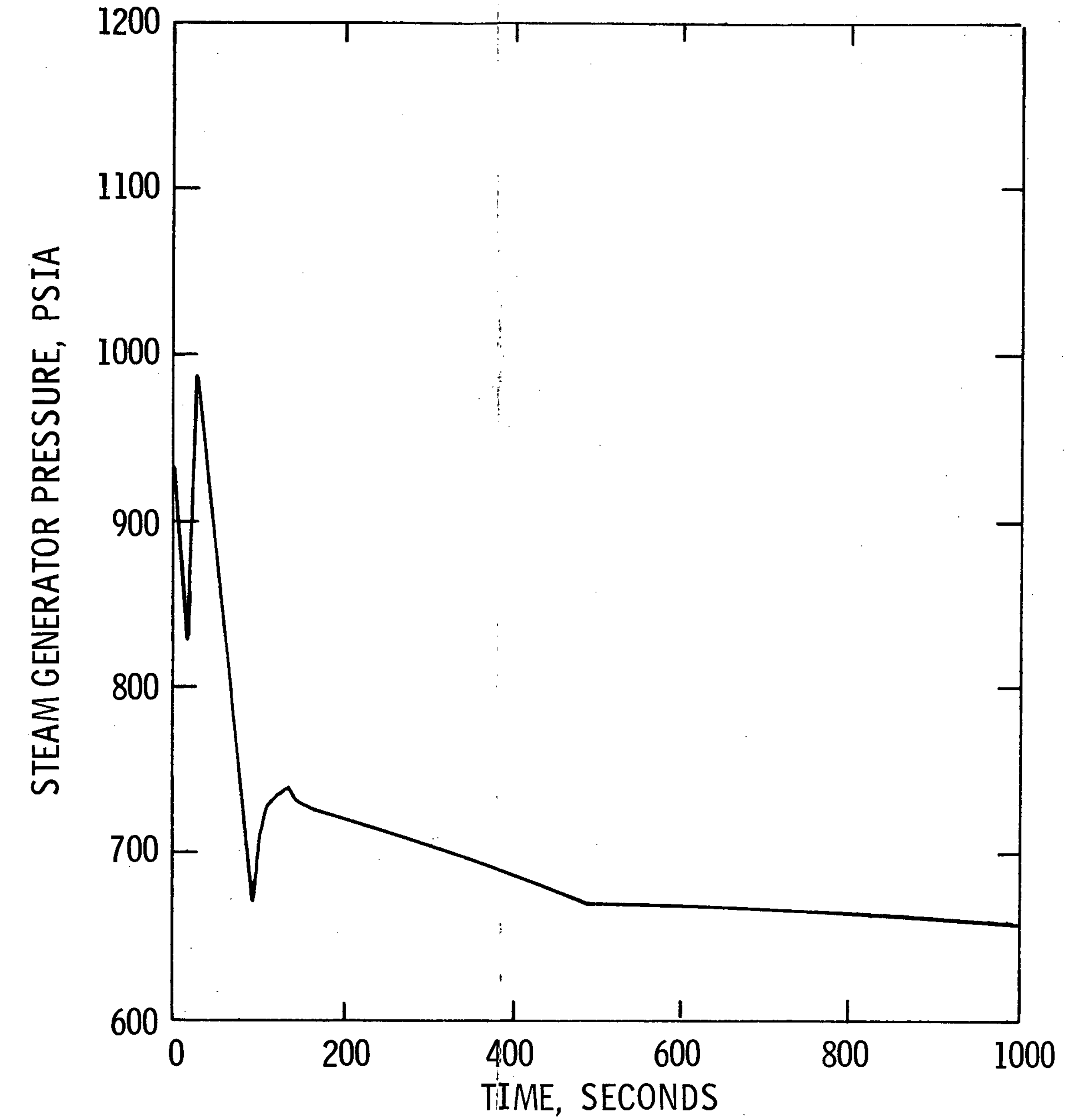


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW PRESSURIZER WATER VOLUME vs. TIME

Figure 15.1-6

10079-11D-926
20OC6

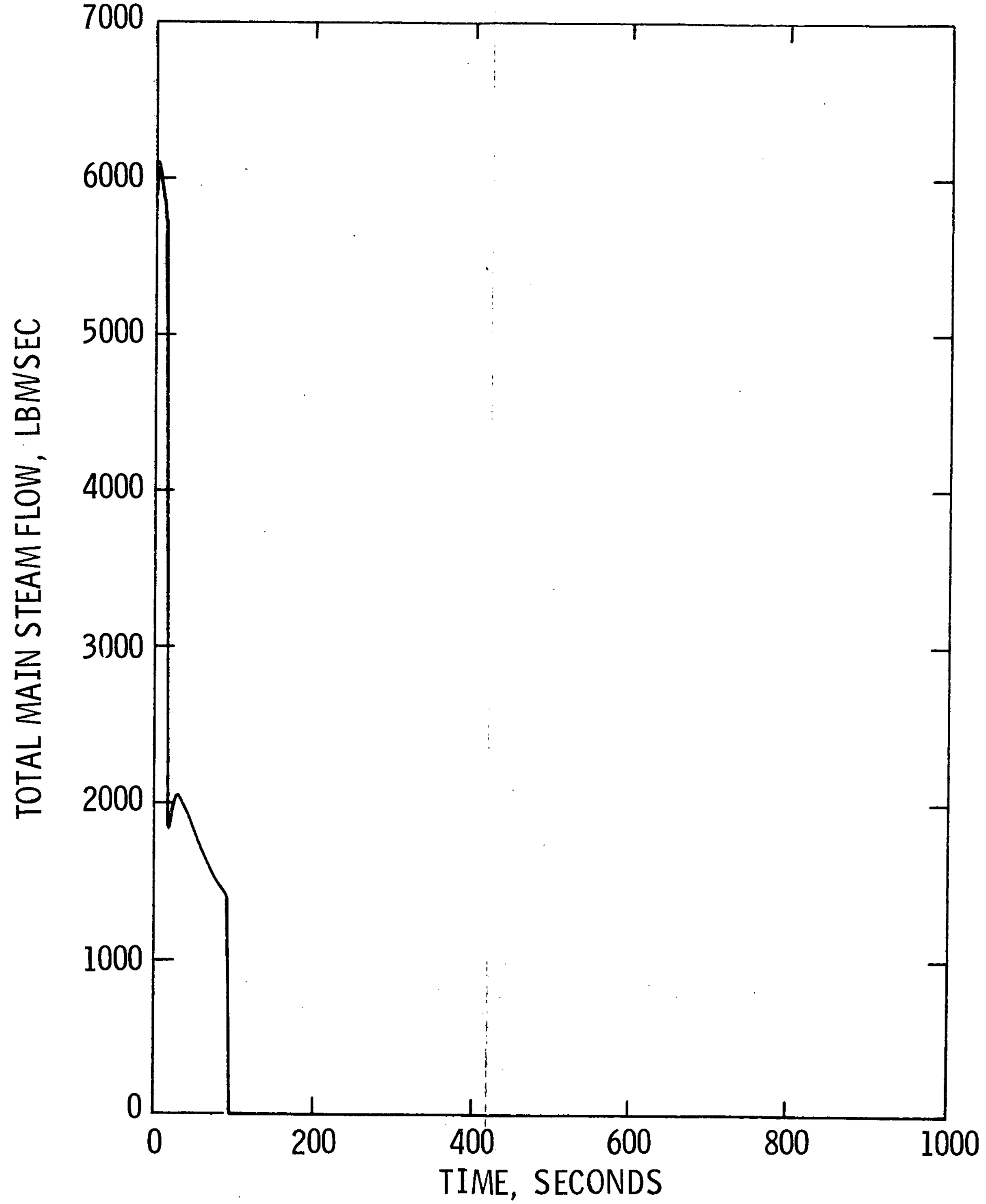


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW STEAM GENERATOR PRESSURE vs. TIME

Figure 15.1-7

10079-11D-927
20OC6

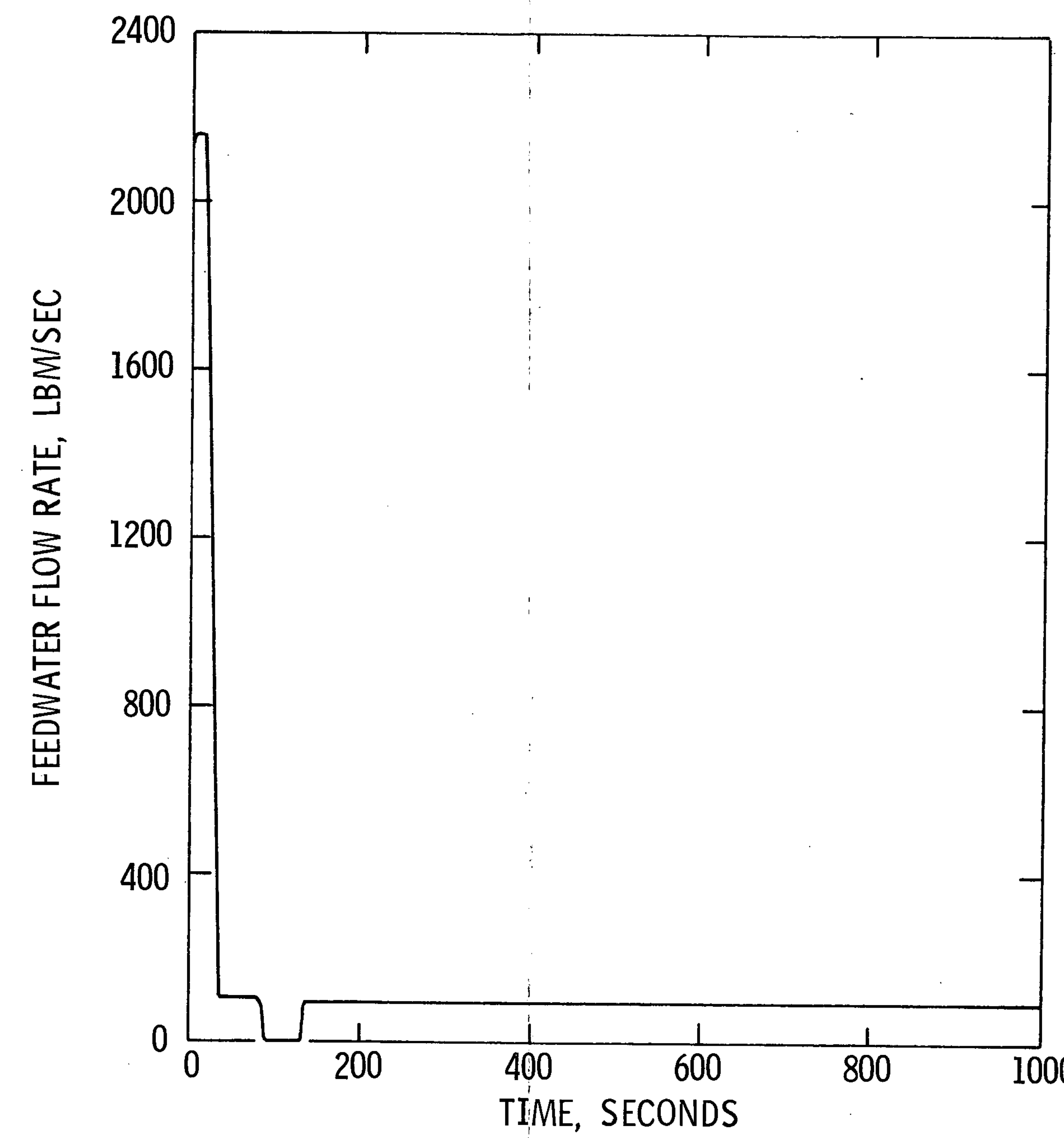


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW TOTAL
MAIN STEAM FLOW vs. TIME

Figure 15.1-8

10079-11D-928
20OC6

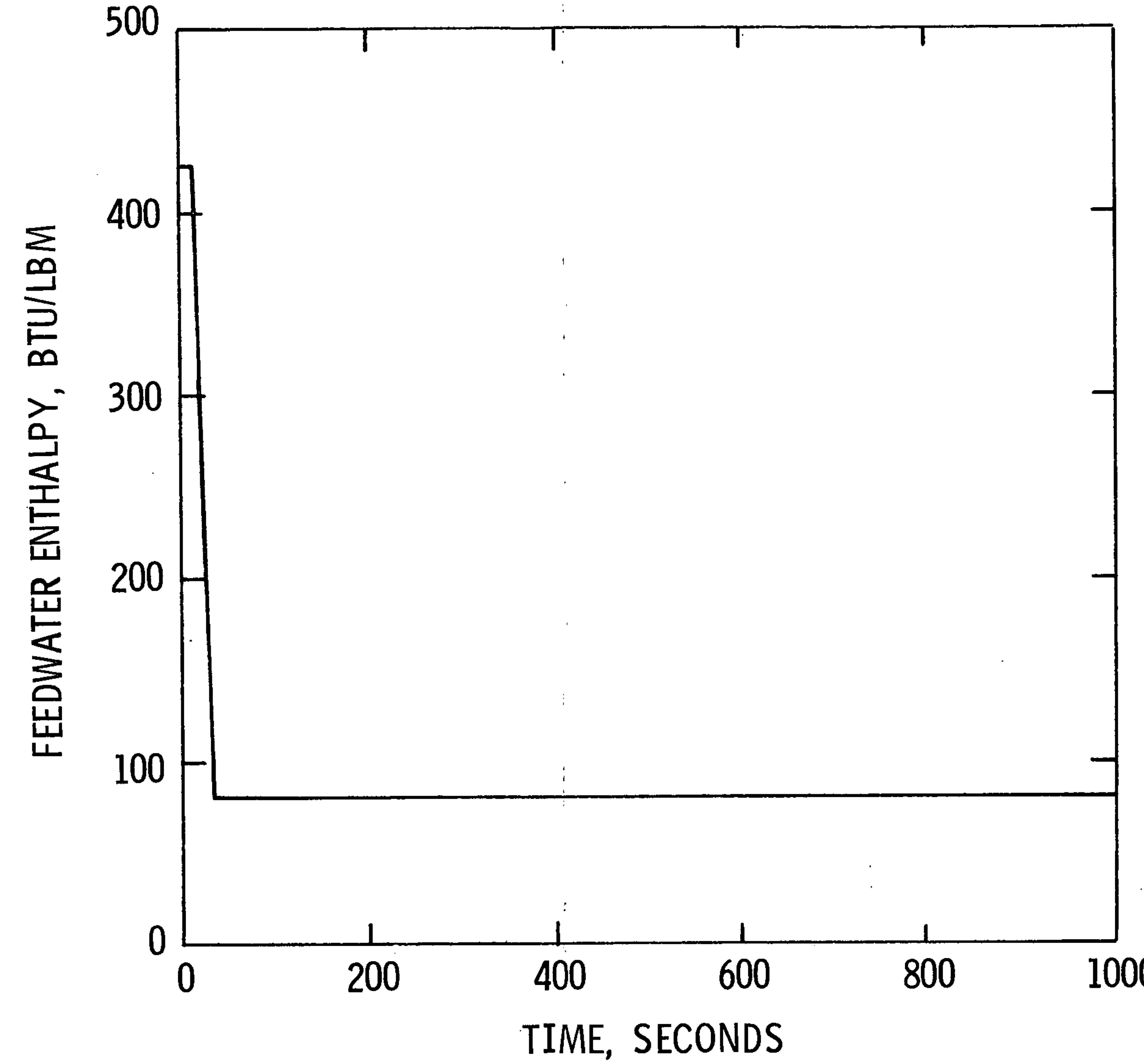


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW
FEEDWATER FLOW RATE vs. TIME

Figure 15.1-9

10079-11D-929
20OC6

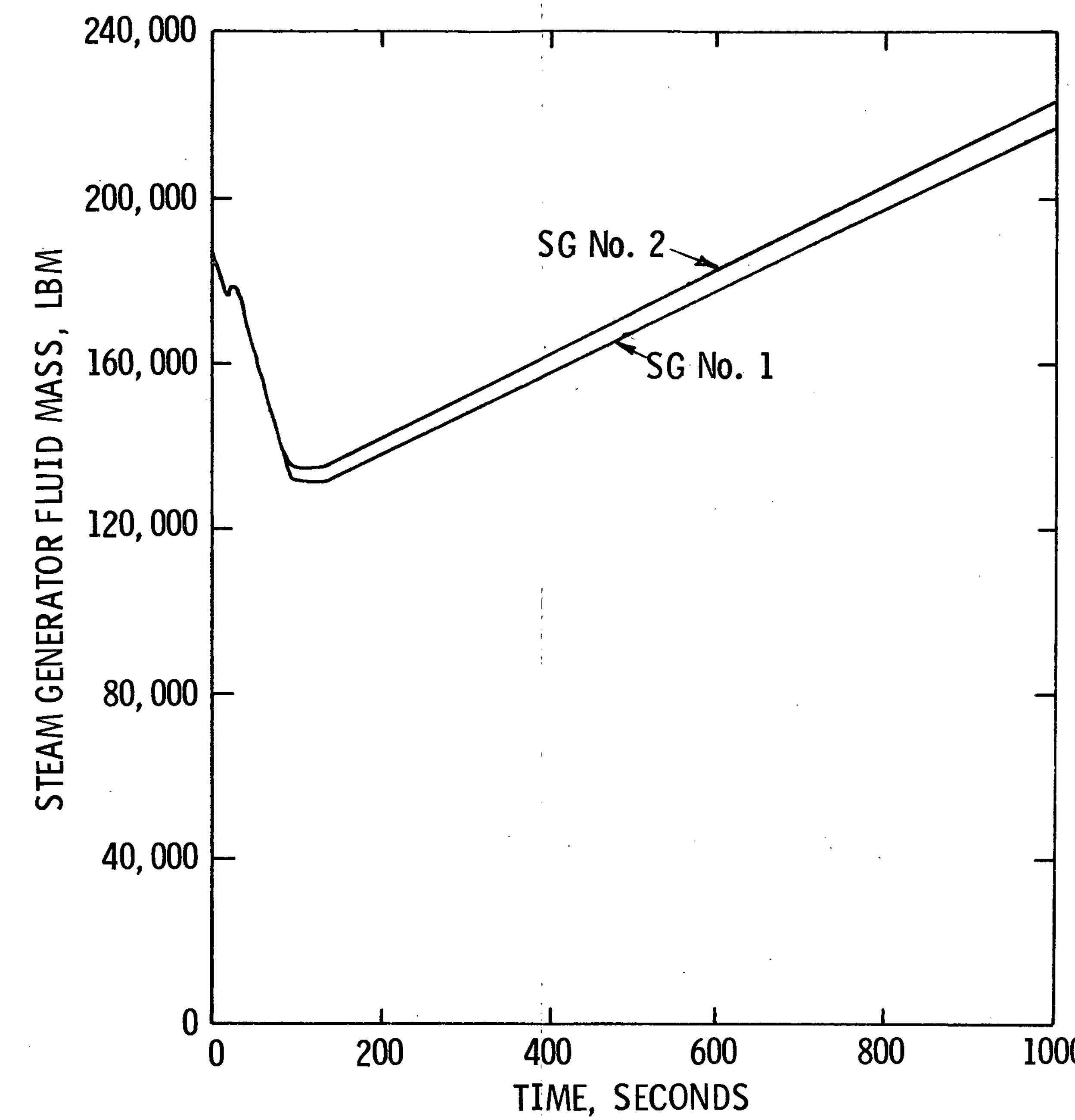


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW FEEDWATER ENTHALPY vs. TIME

Figure 15.1-10

10079-11D-930
20OC6



SAN ONOFRE
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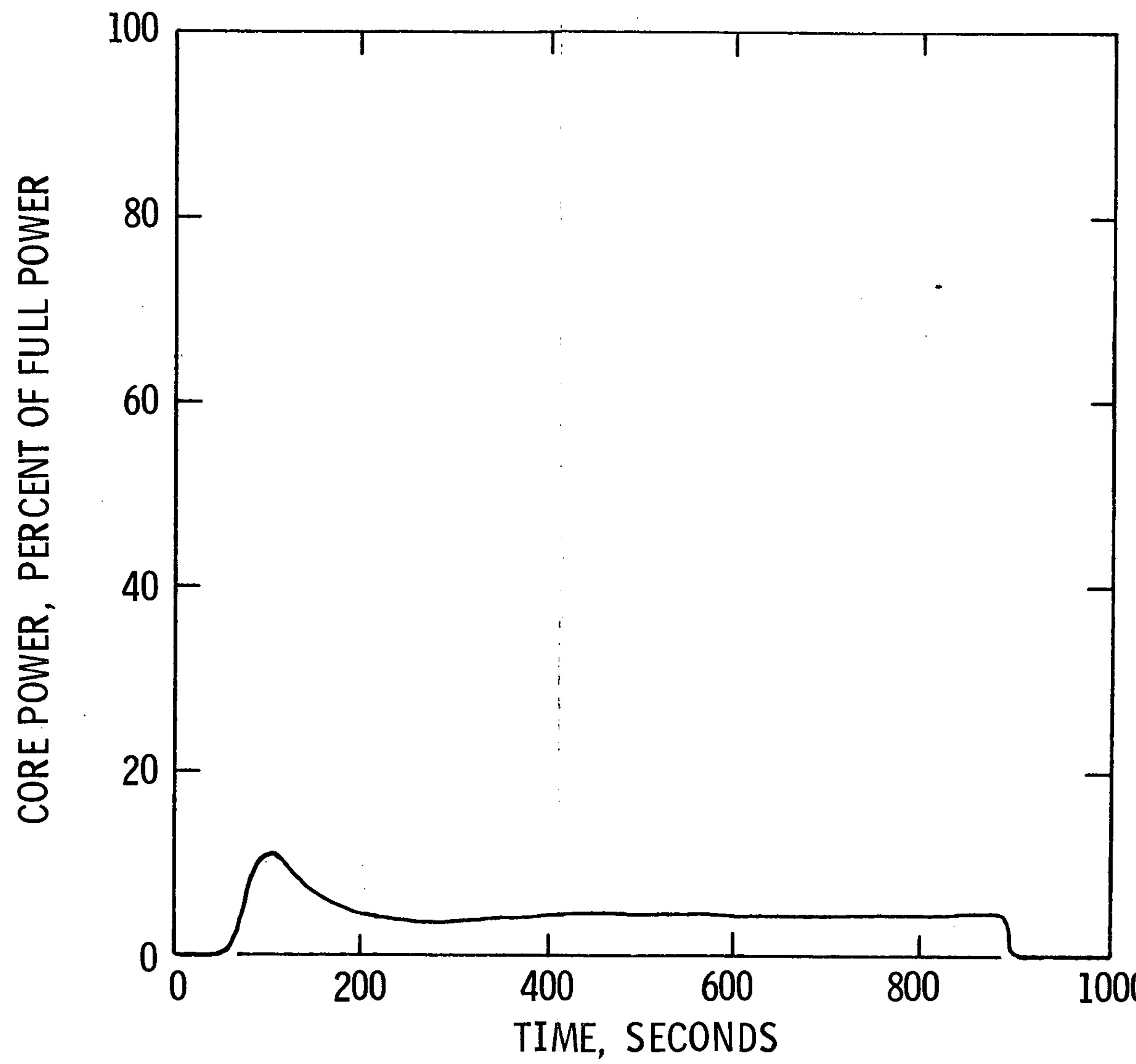
Units 2 & 3

INCREASED MAIN STEAM FLOW STEAM GENERATOR FLUID MASS vs. TIME

Figure 15.1-11

10079-11D-931
20OC6

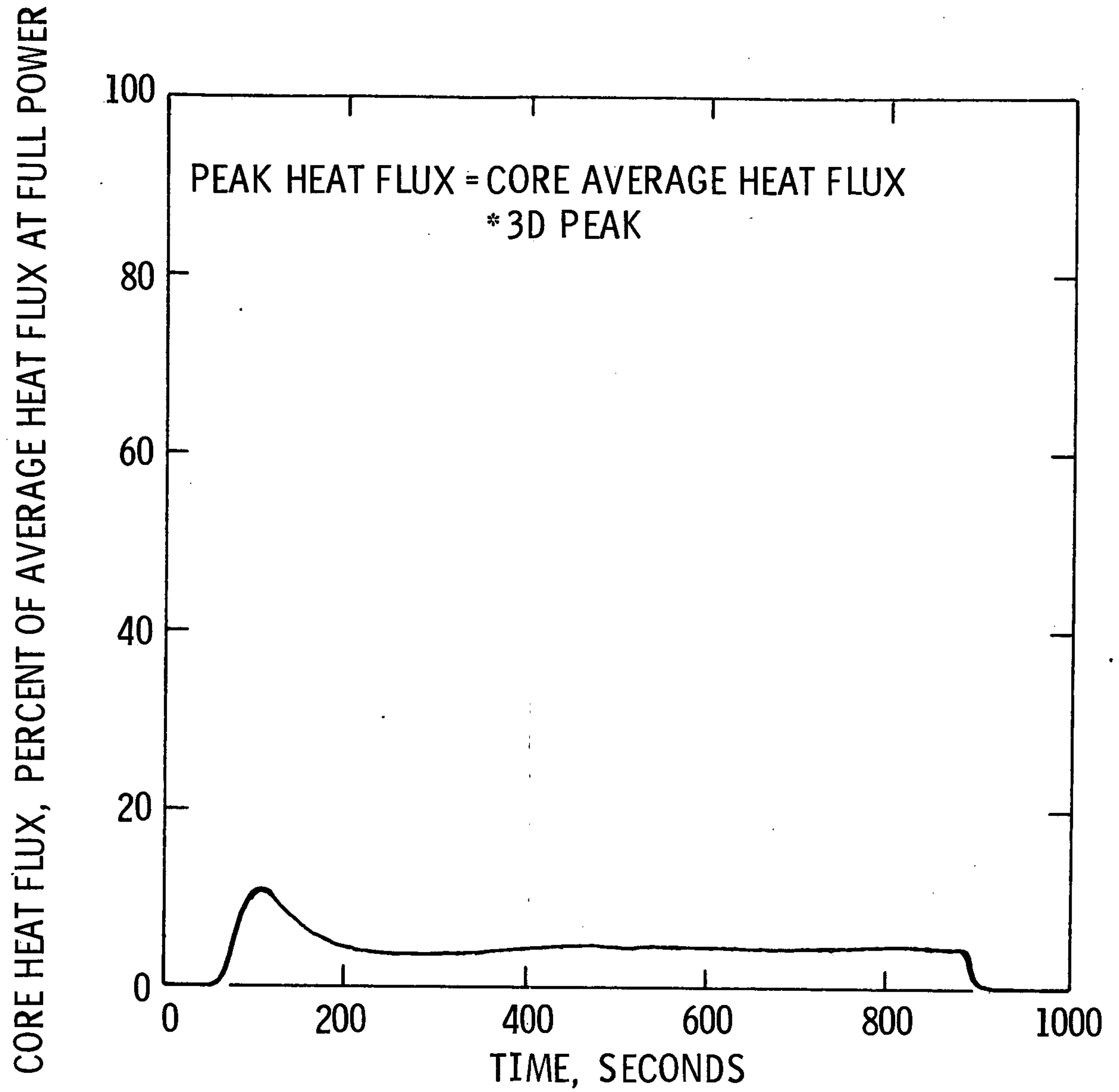


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE CORE POWER vs. TIME

Figure 15.1-12

10079-11D-932
20OC6

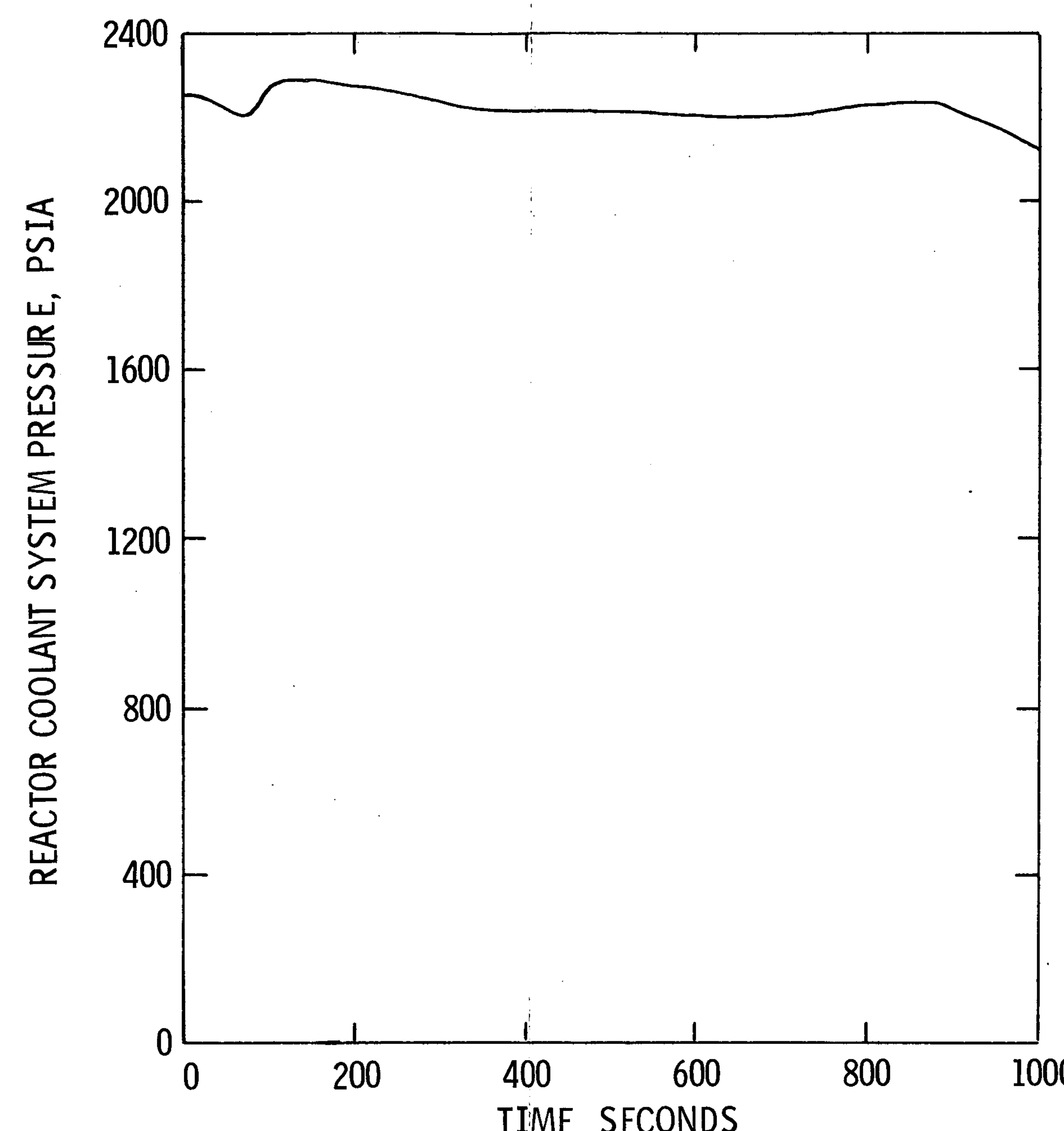


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE CORE HEAT FLUX vs. TIME

Figure 15.1-13

10079·11D-933
200C6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM
GENERATOR ATMOSPHERIC DUMP VALVE
REACTOR COOLANT SYSTEM
PRESSURE vs. TIME

Figure 15.1-14

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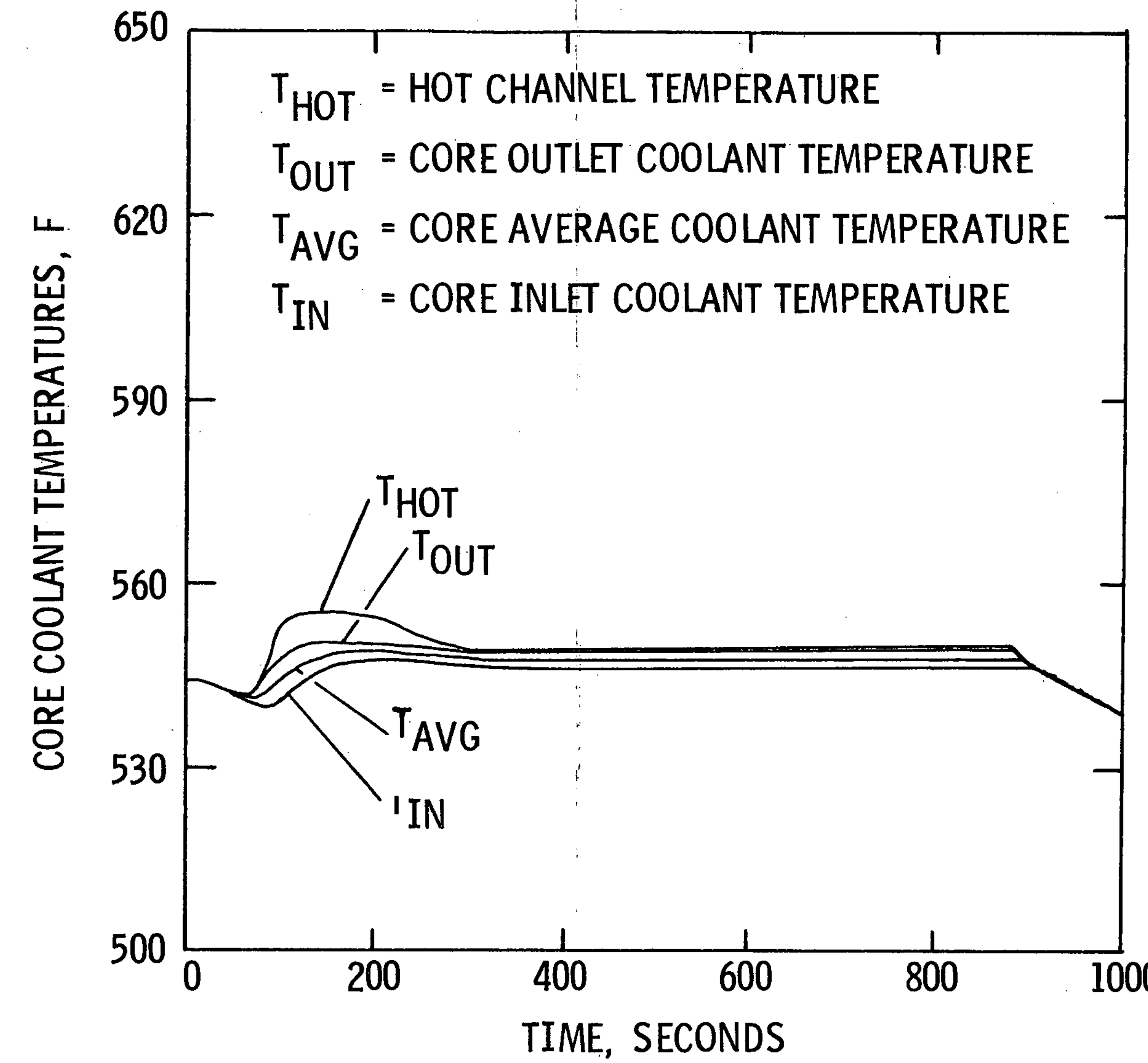
20OC6



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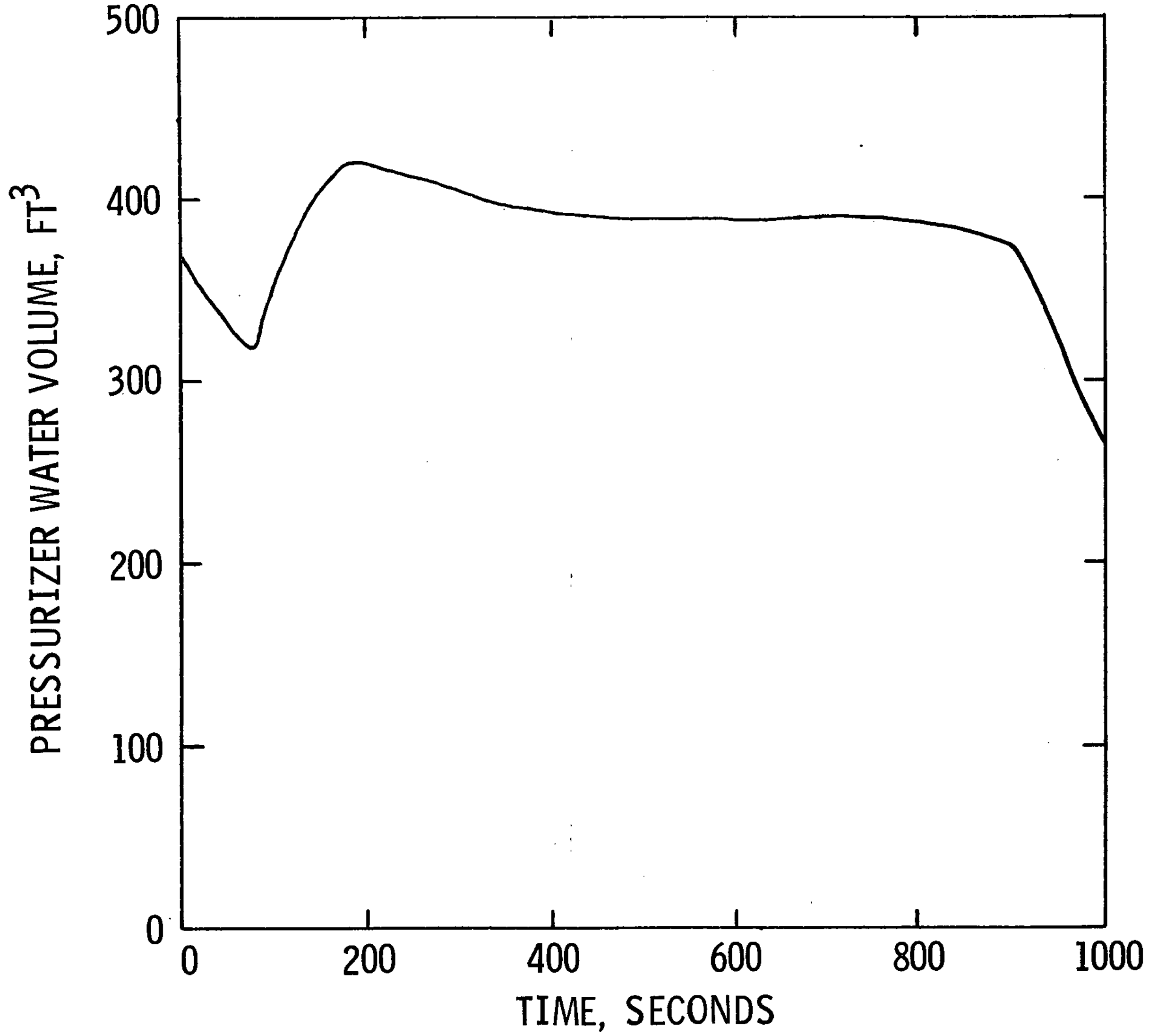
200C6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE
REACTOR COOLANT TEMPERATURE vs. TIME

Figure 15.1-15

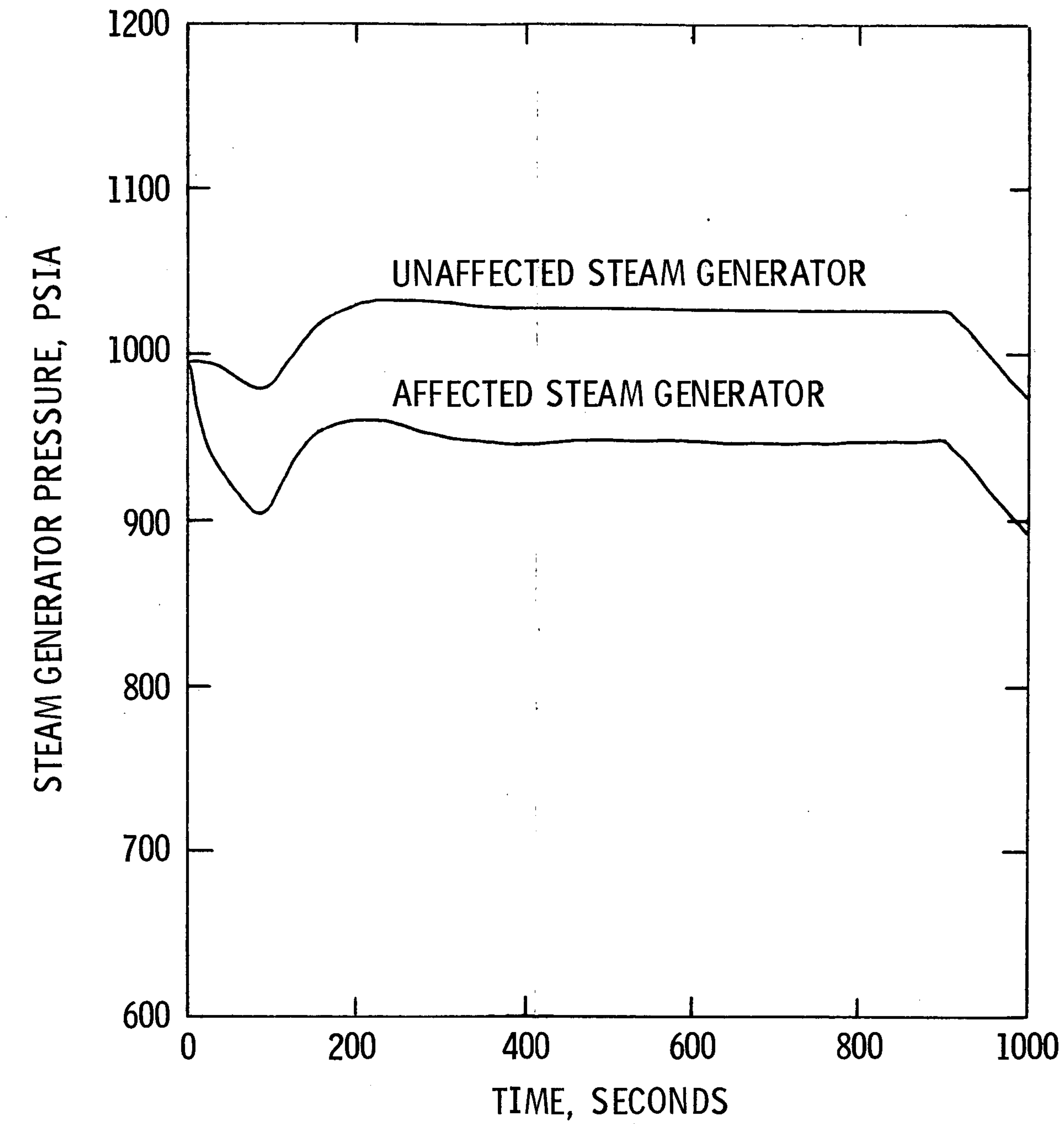


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM
GENERATOR ATMOSPHERIC DUMP VALVE
PRESSURIZER WATER
VOLUME vs. TIME

Figure 15.1-16

10079-11D-936
20OC6

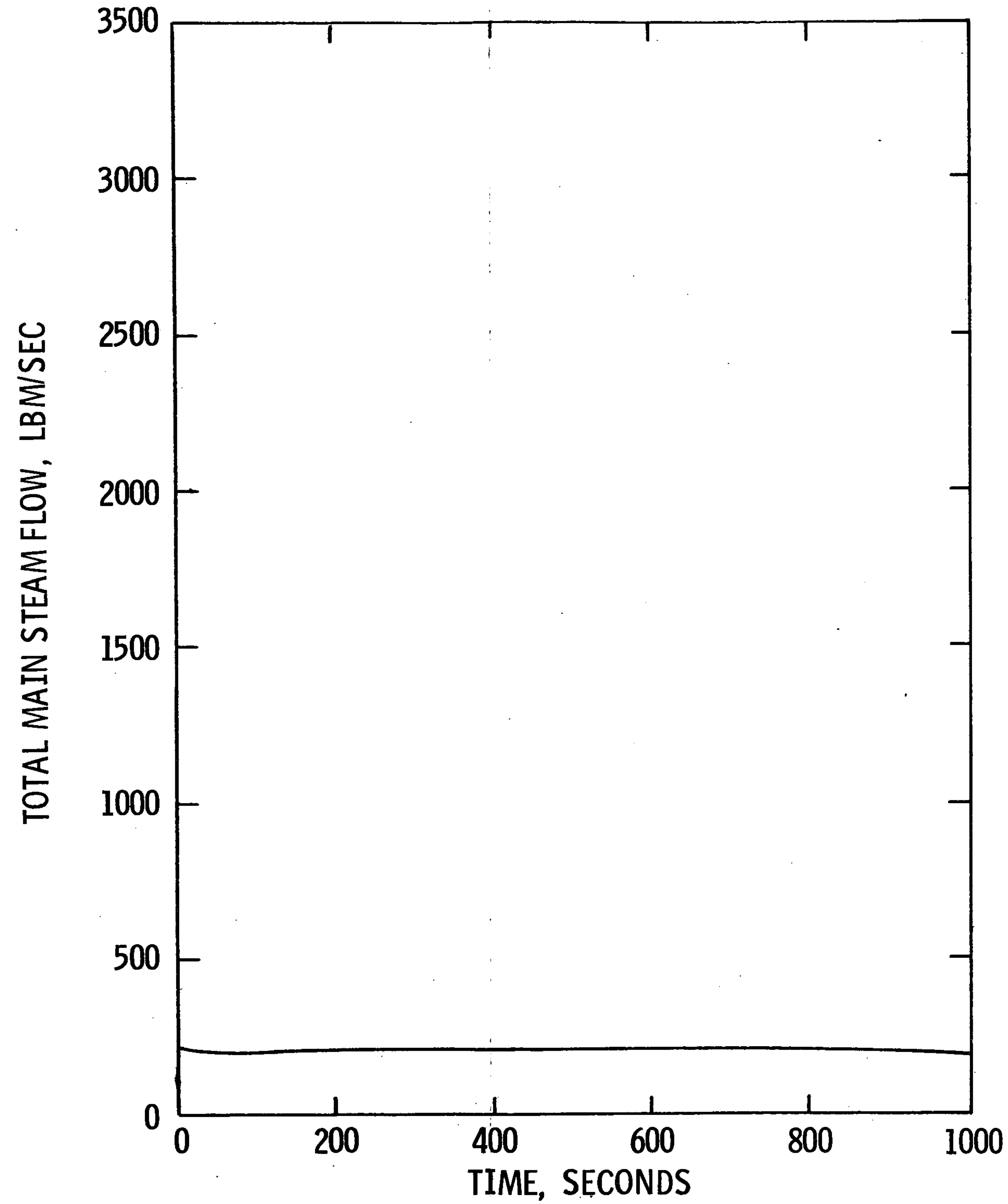


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE STEAM GENERATOR PRESSURE vs. TIME

Figure 15.1-17

10079-11D-937
200C6

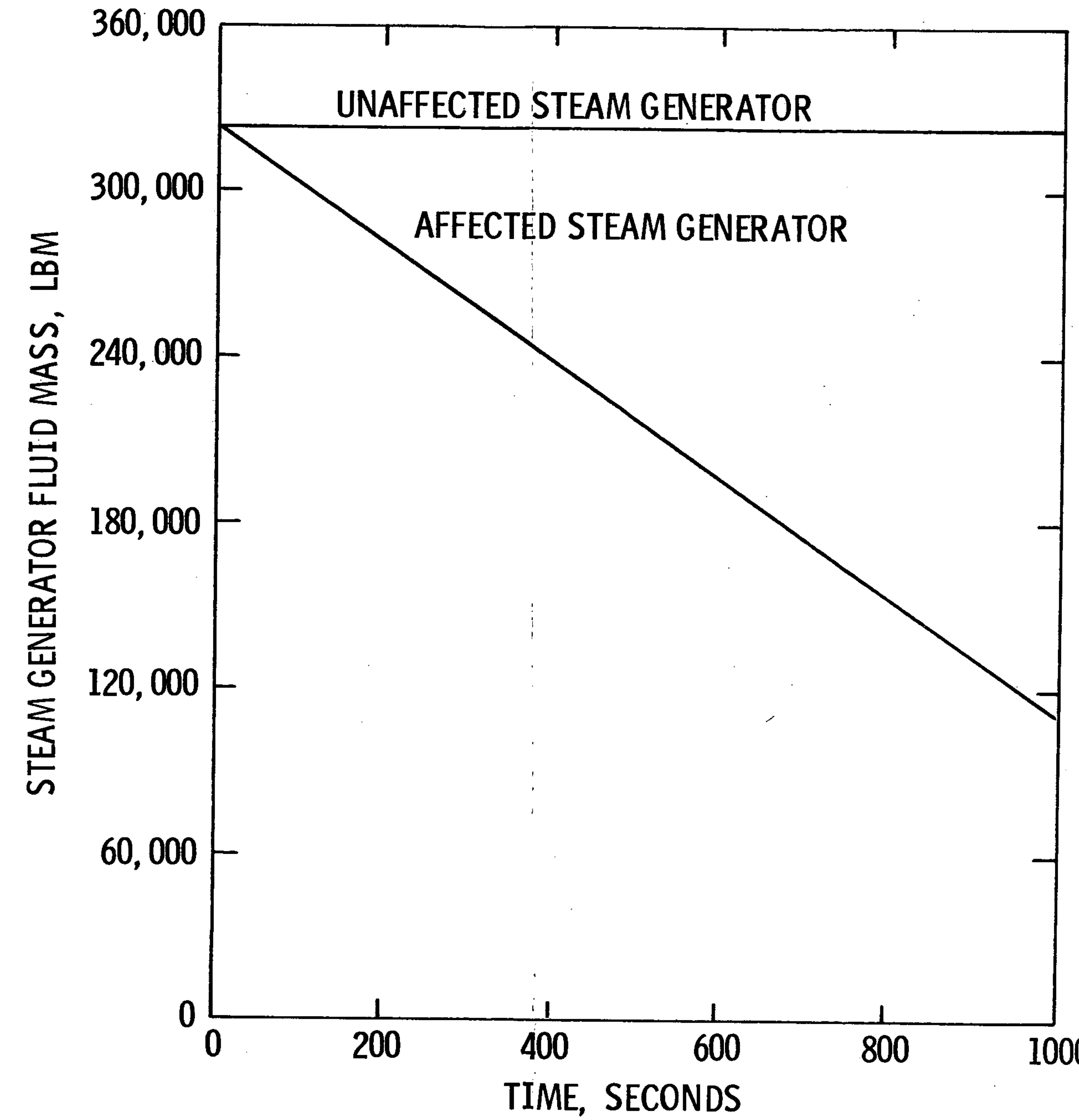


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE TOTAL MAIN STEAM FLOW vs. TIME

Figure 15.1-18

10079-11D-938
200C6



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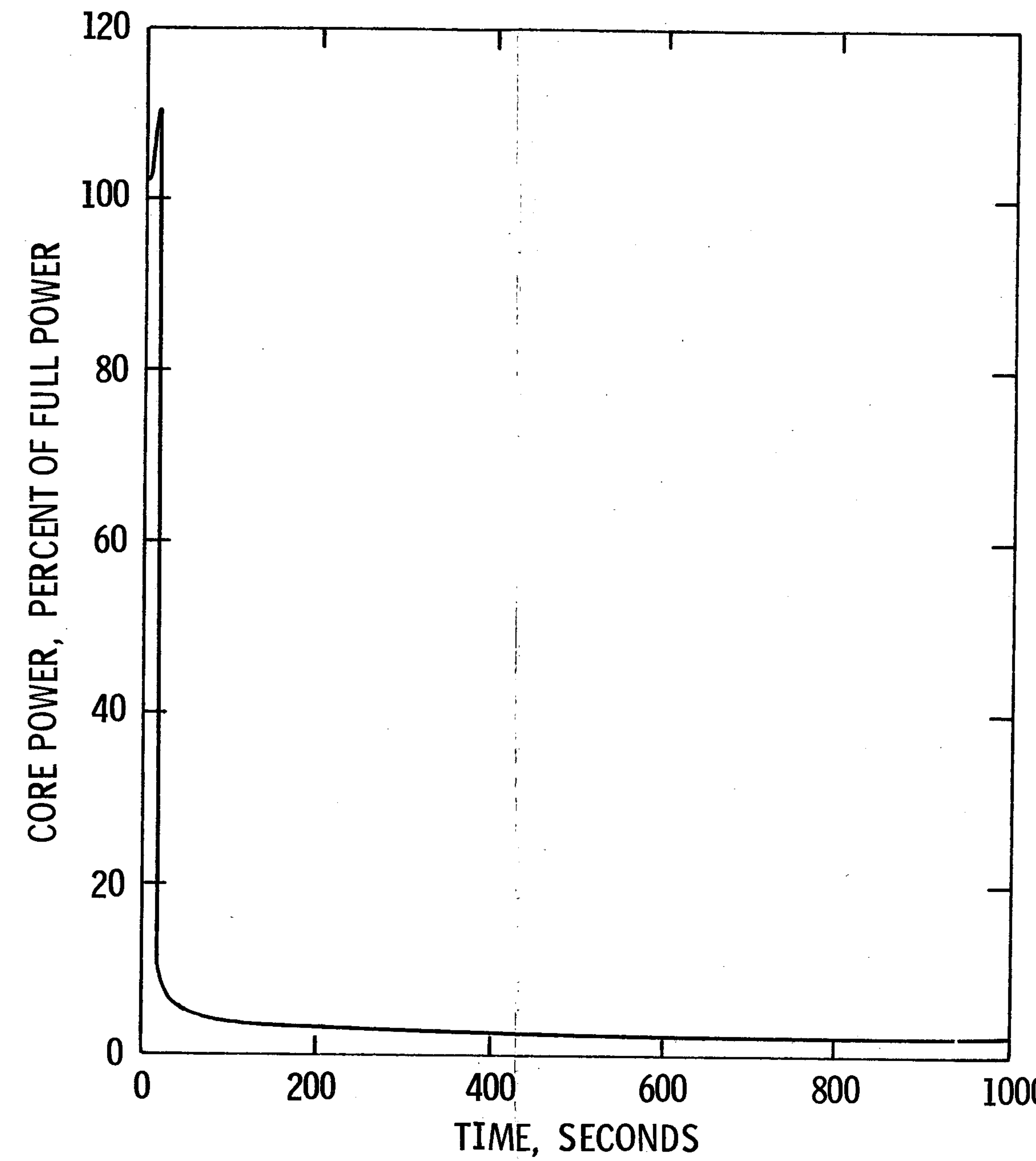
20OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE
STEAM GENERATOR FLUID MASS vs. TIME

Figure 15.1-19

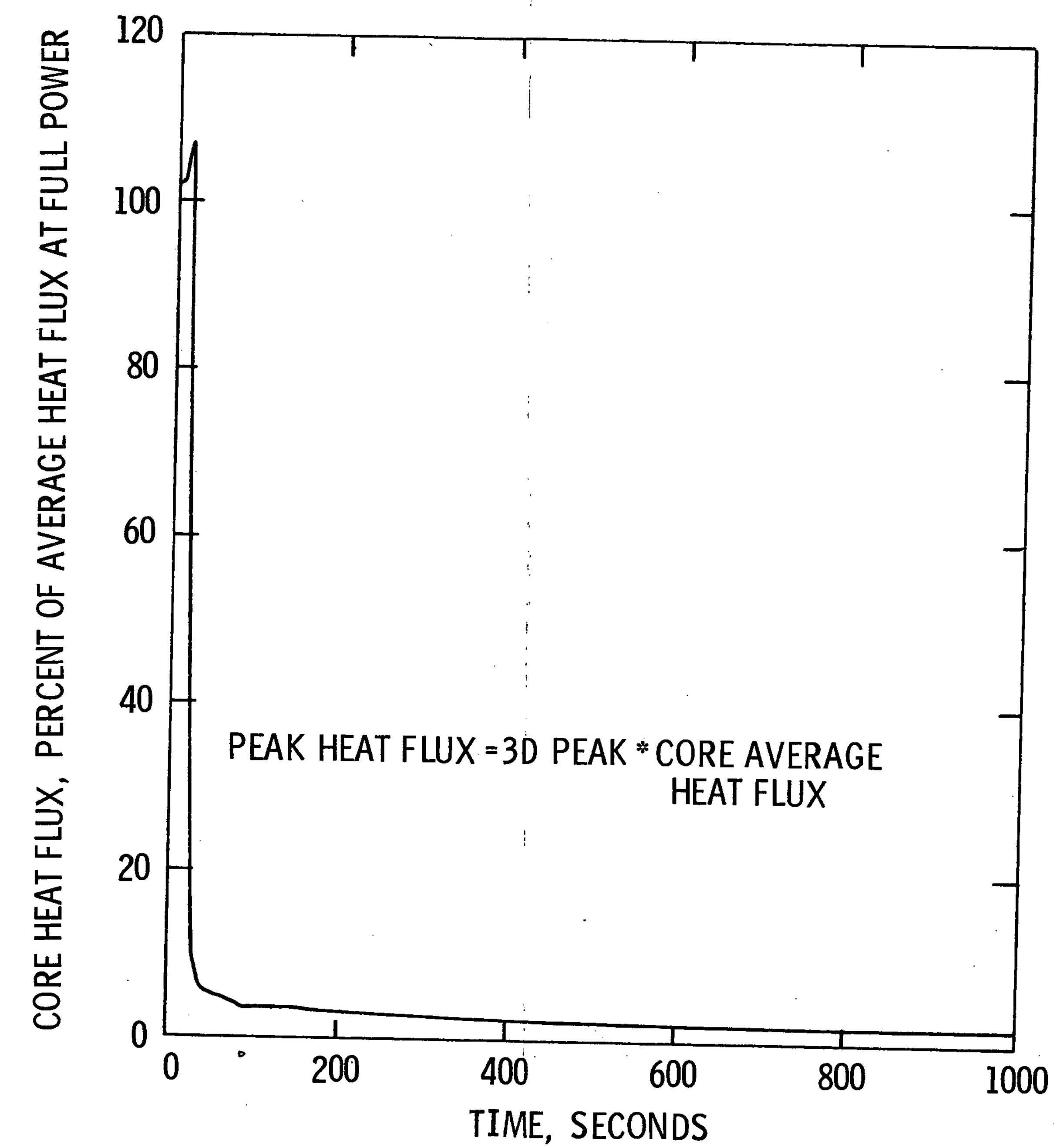


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH
CONCURRENT SINGLE FAILURE
CORE POWER vs. TIME

Figure 15.1-20

10079-11D-940
20OC6

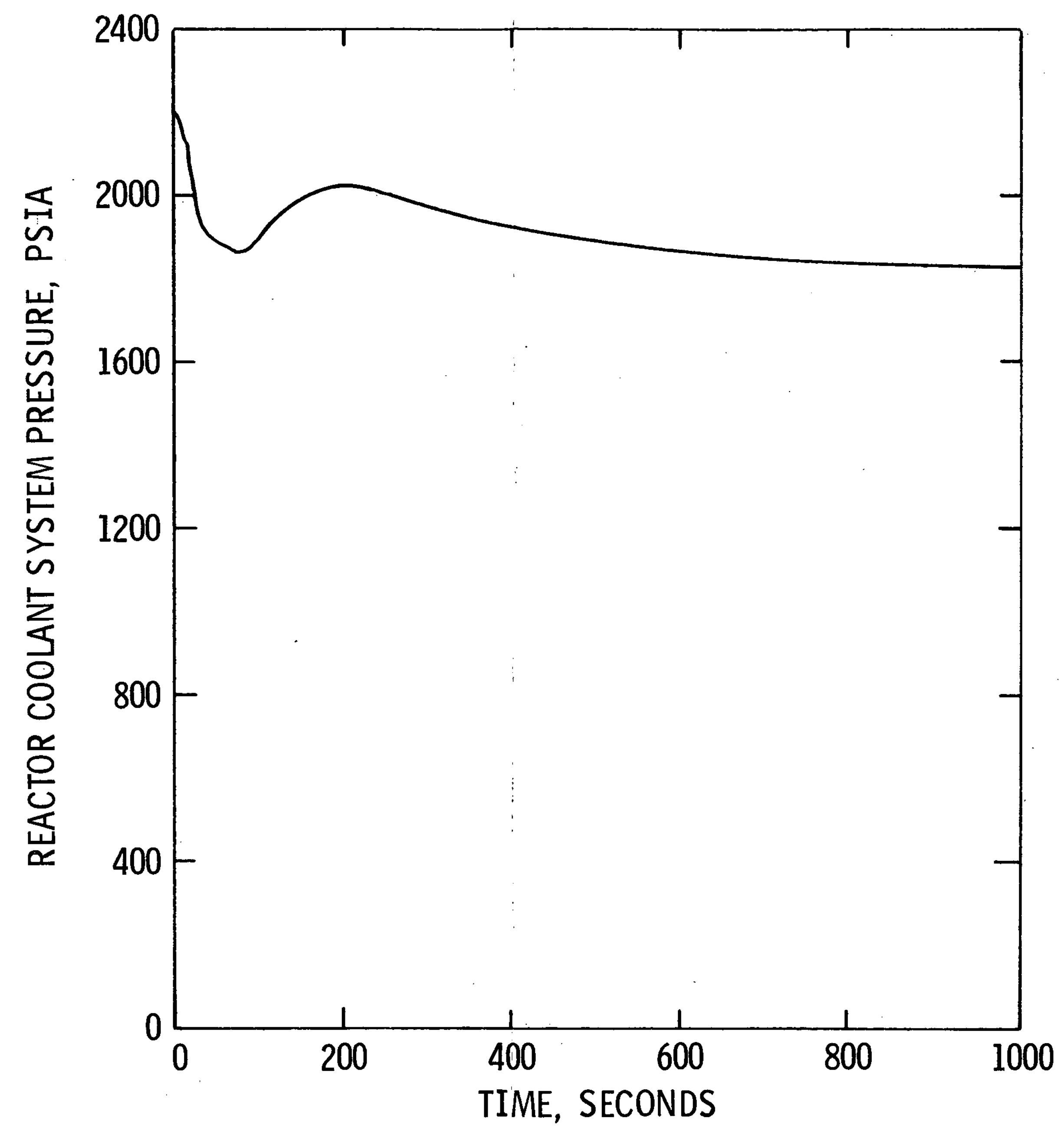


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH
CONCURRENT SINGLE FAILURE
CORE HEAT FLUX vs. TIME

Figure 15.1-21

10079-11D-941
20OC6

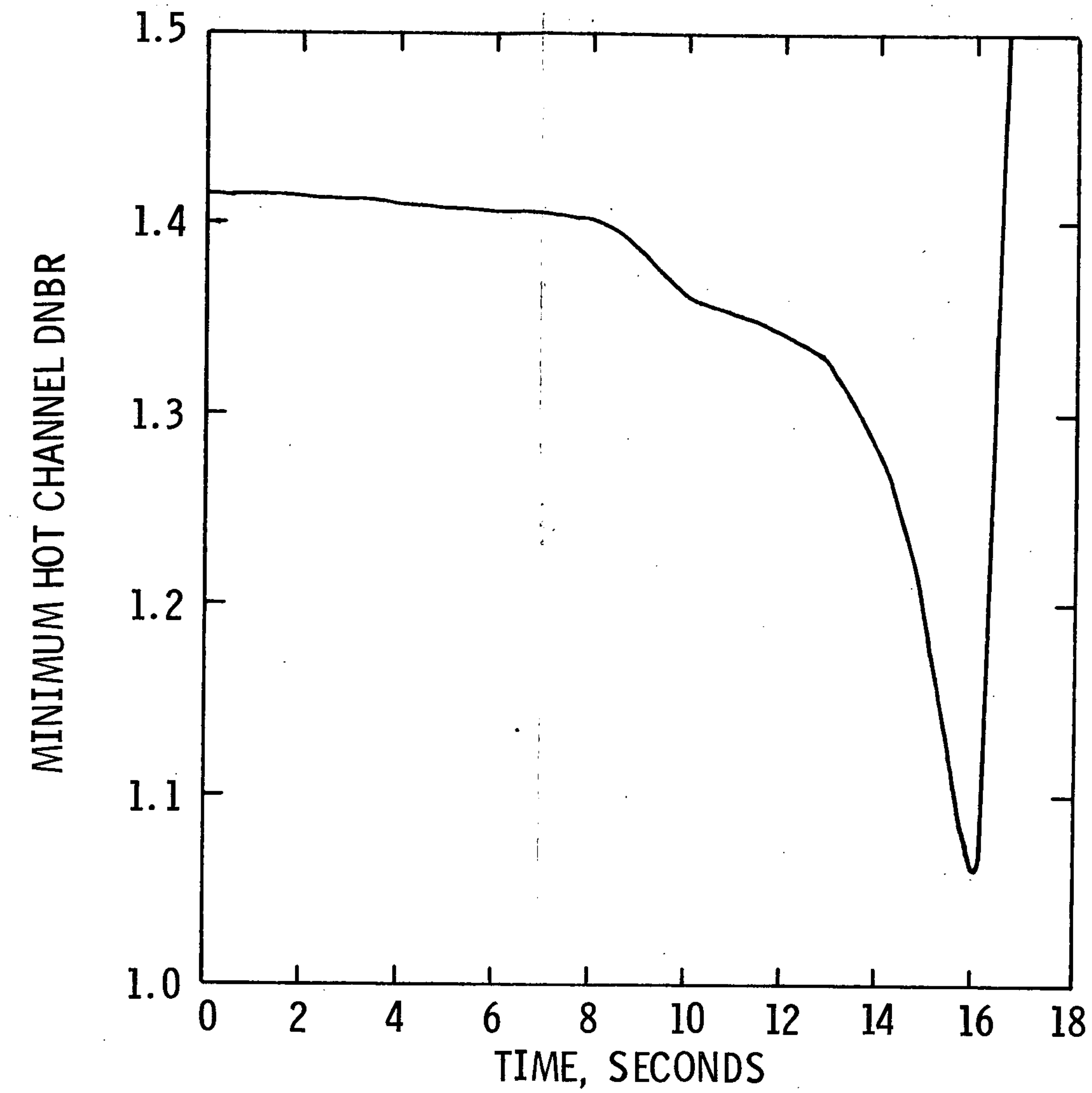


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE REACTOR COOLANT SYSTEM PRESSURE vs. TIME

Figure 15.1-22

10079-11D-942
20OC6

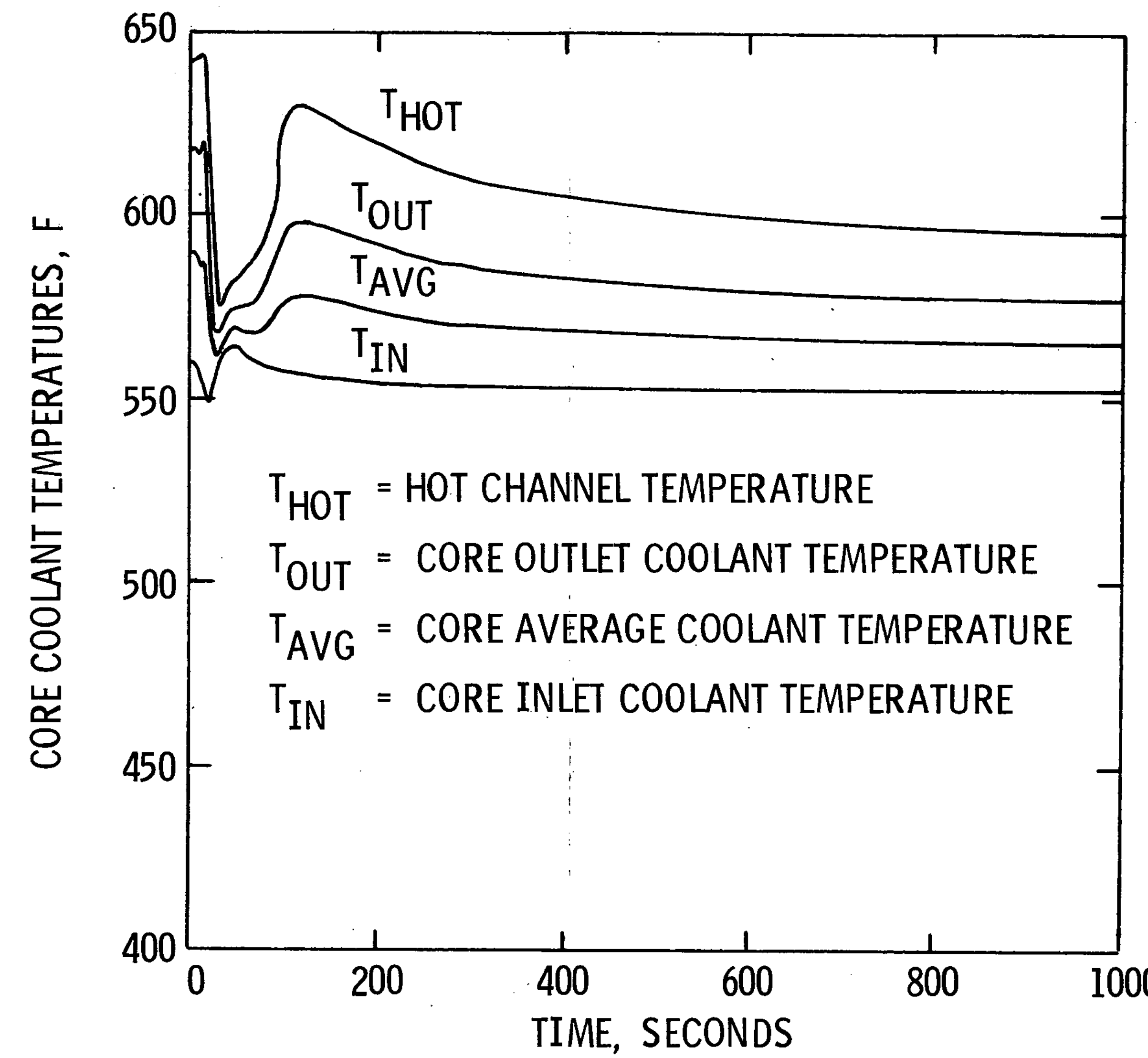


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE MINIMUM HOT CHANNEL DNBR vs. TIME

Figure 15.1-23

10079-11D-943
20OC6



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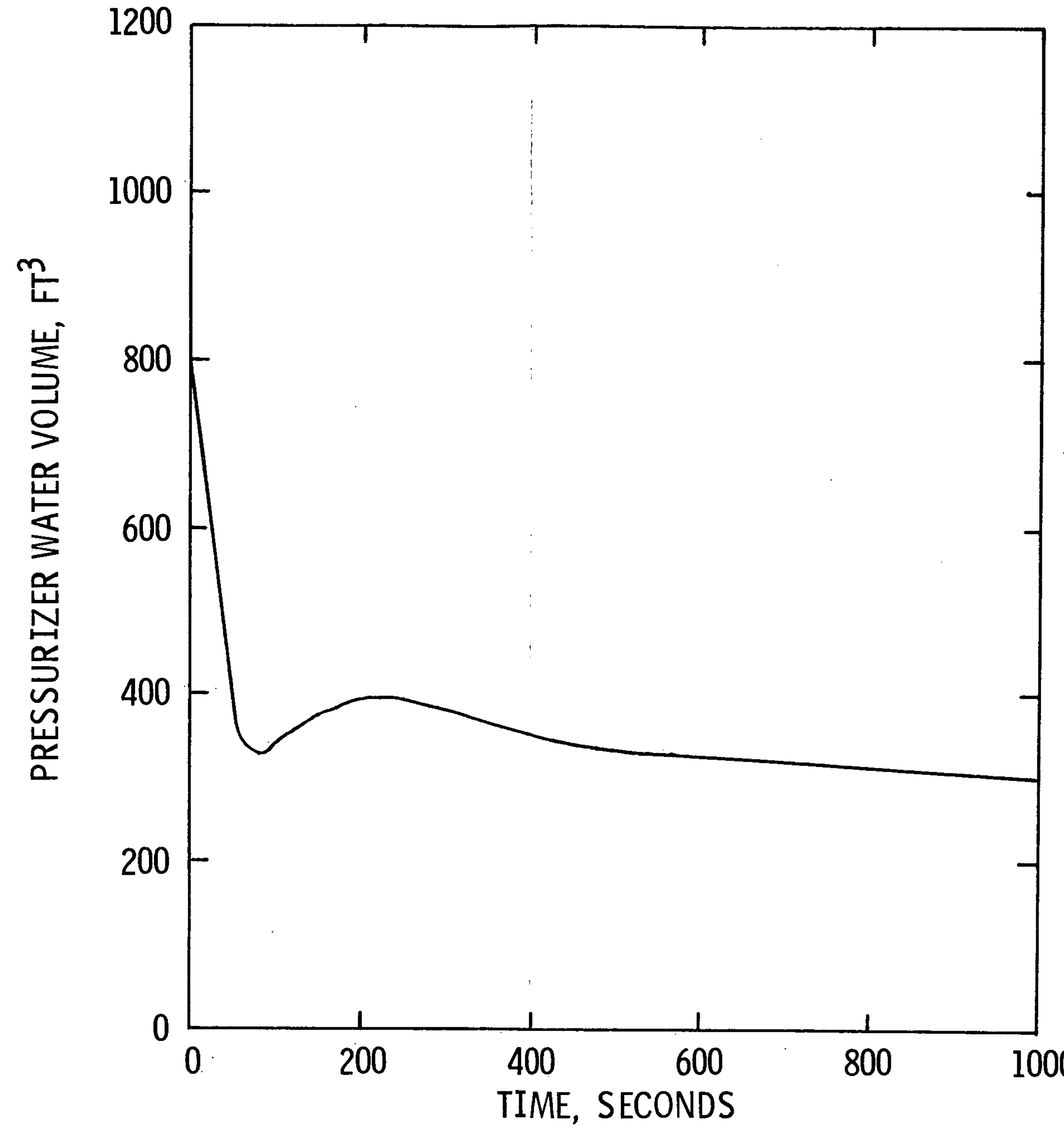
SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE REACTOR COOLANT TEMPERATURE vs. TIME

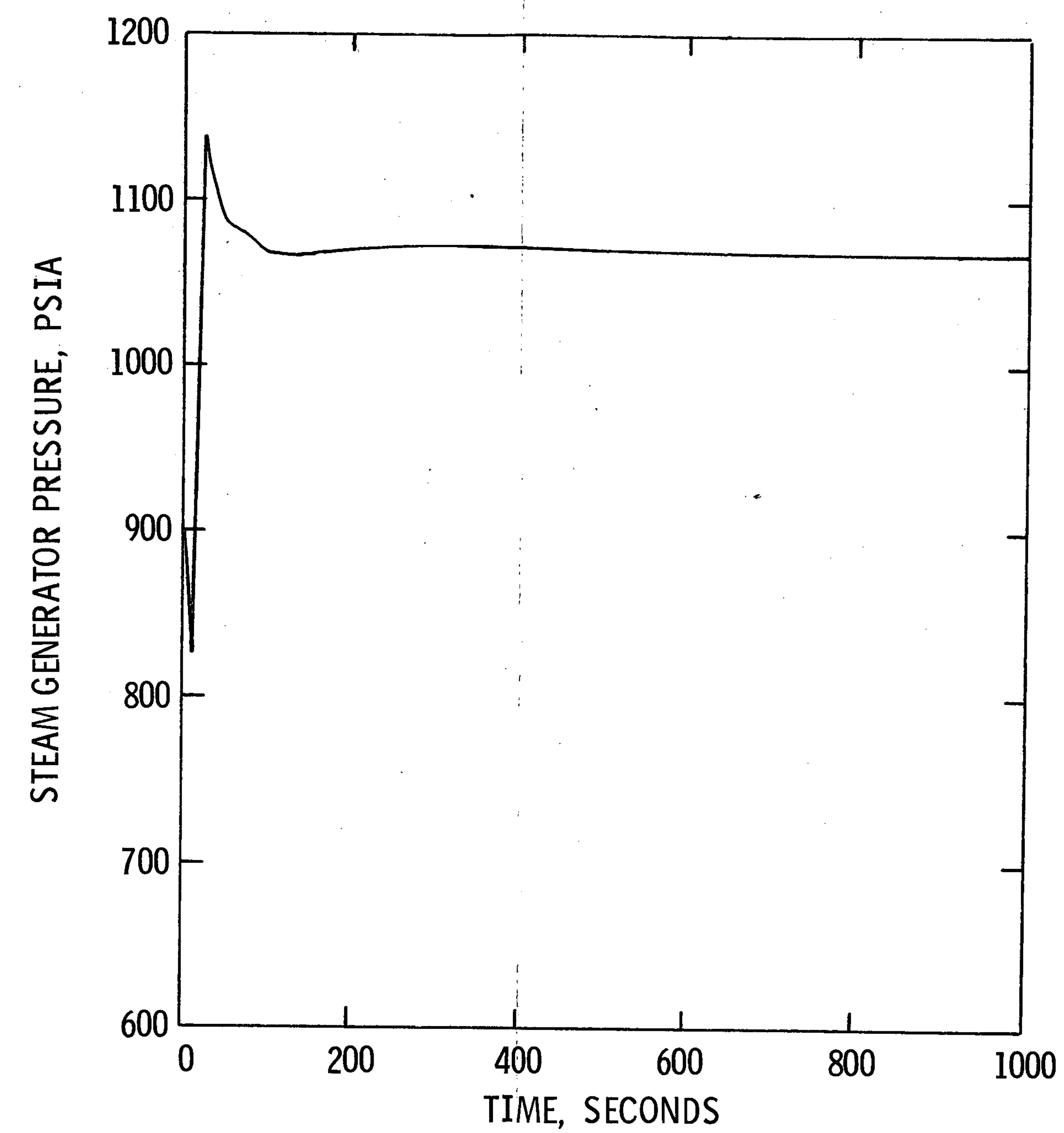
Figure 15.1-24



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH
CONCURRENT SINGLE FAILURE
PRESSURIZER WATER VOLUME vs. TIME

Figure 15.1-25

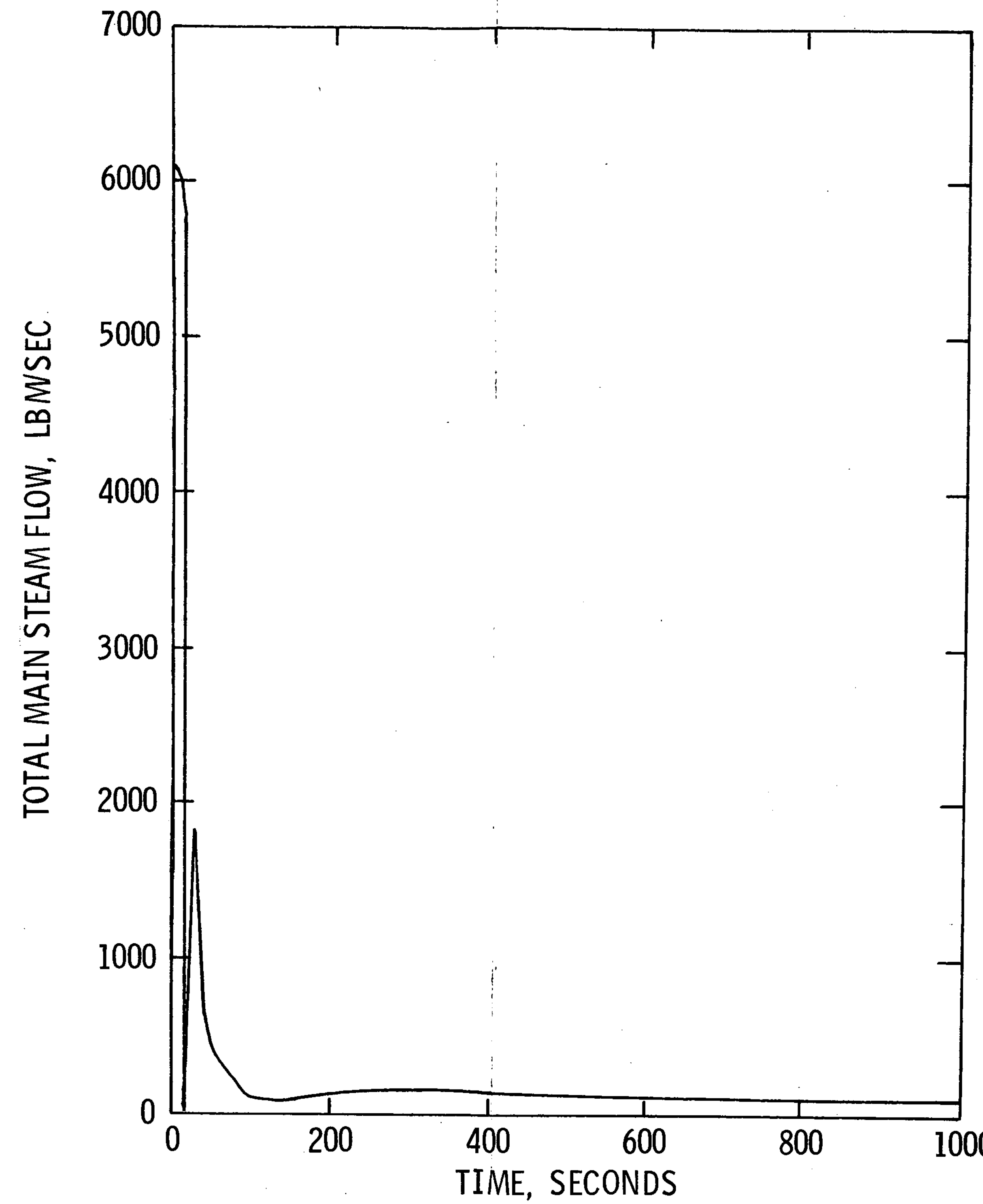


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE STEAM GENERATOR PRESSURE vs. TIME

Figure 15.1-26

10079-11D-946
20OC6

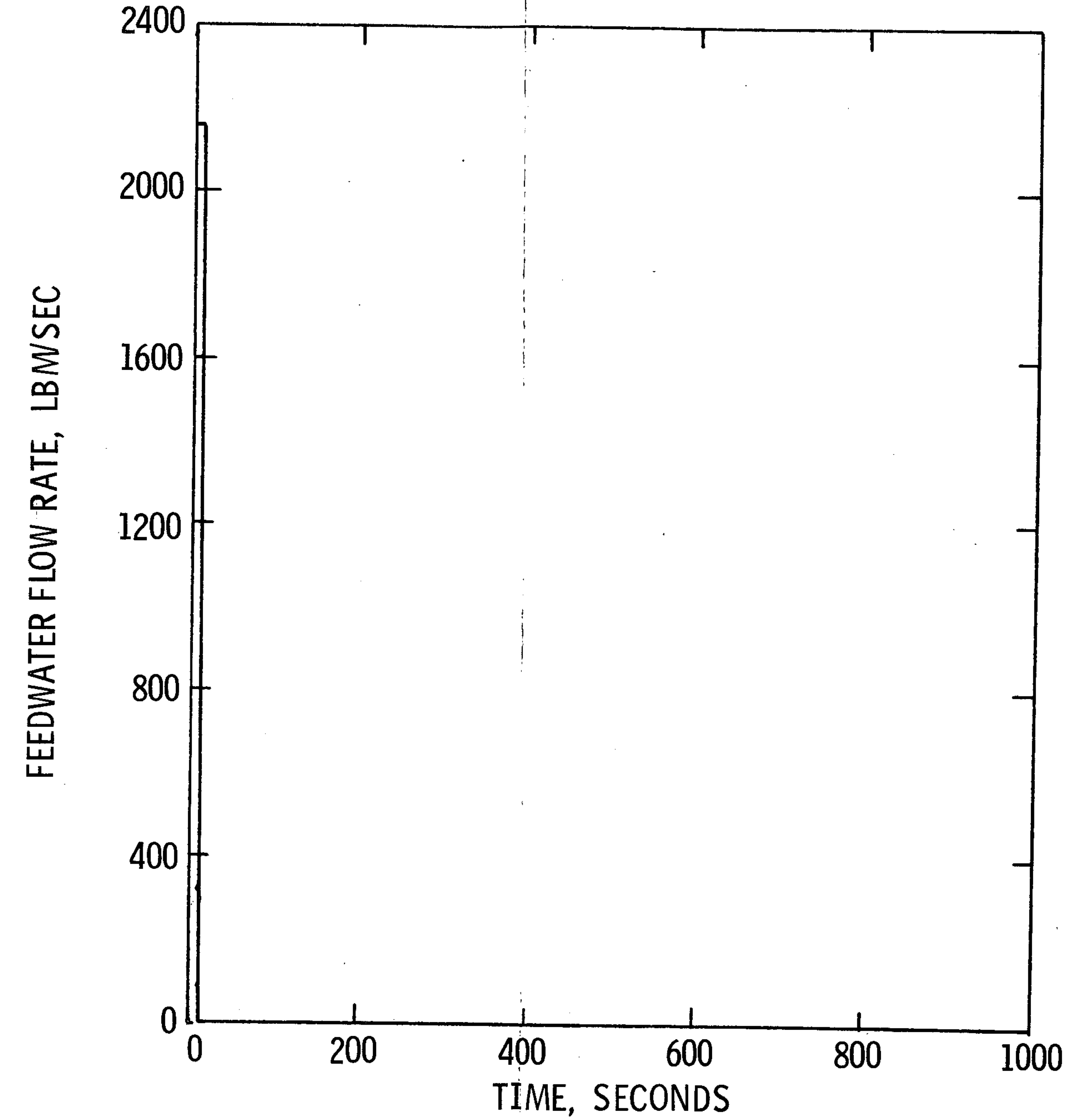


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE TOTAL MAIN STEAM FLOW vs. TIME

Figure 15.1-27

10079-11D-947
20OC6

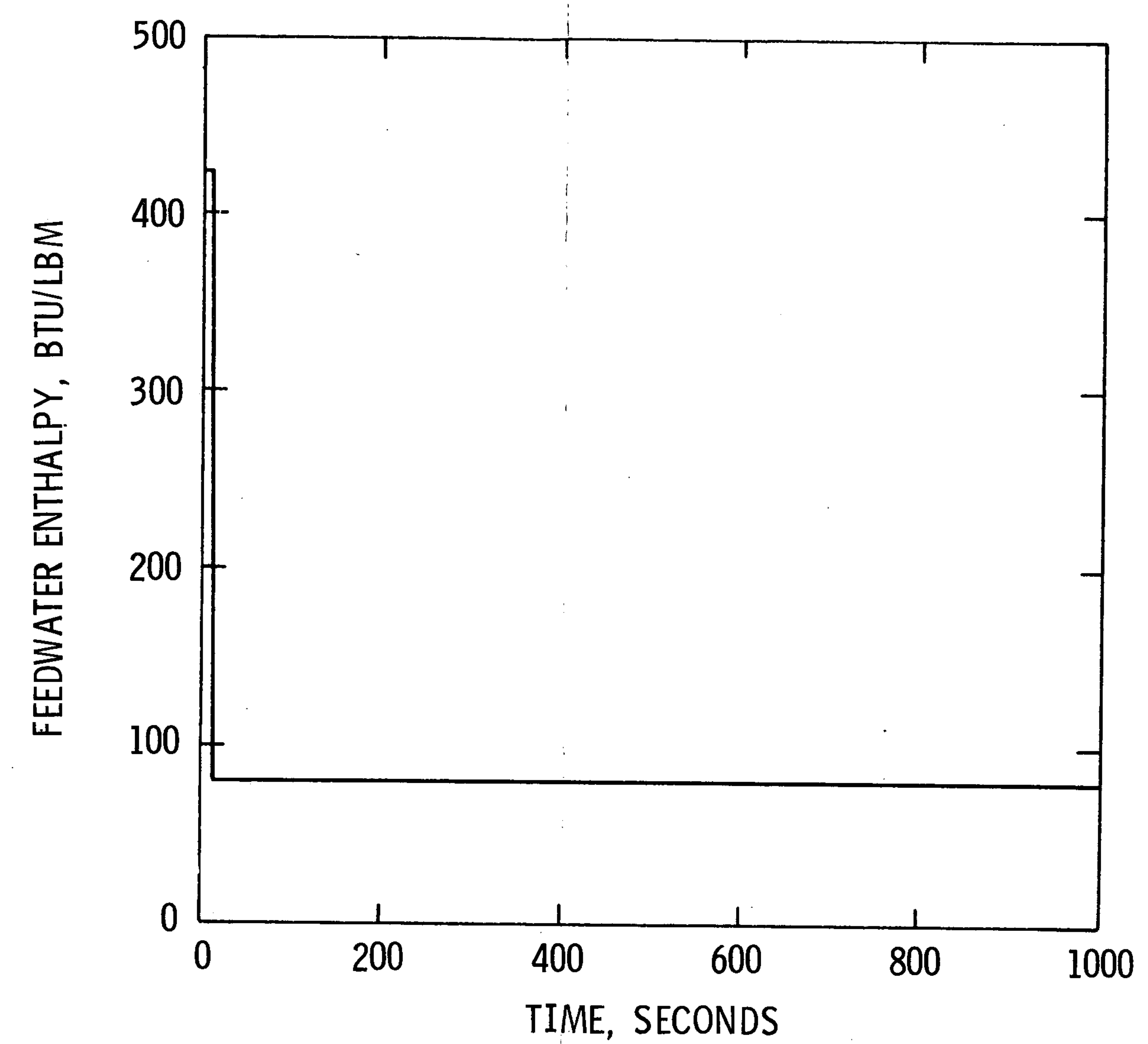


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH
CONCURRENT SINGLE FAILURE
FEEDWATER FLOWRATE vs. TIME

Figure 15.1-28

10079-11D-948
200C6

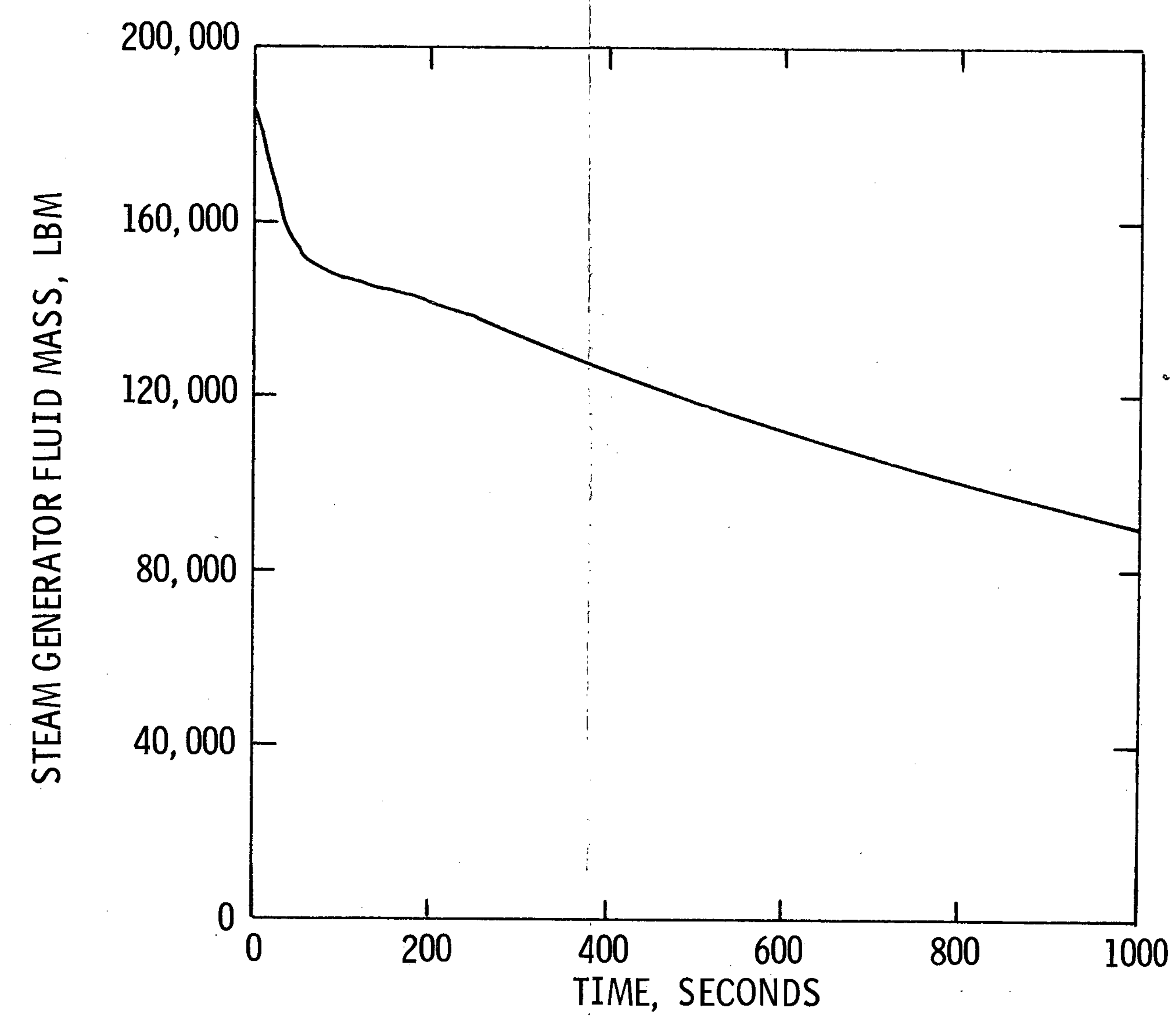


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH
CONCURRENT SINGLE FAILURE
FEEDWATER ENTHALPY vs. TIME

Figure 15.1-29

10079-11D-949
20OC6

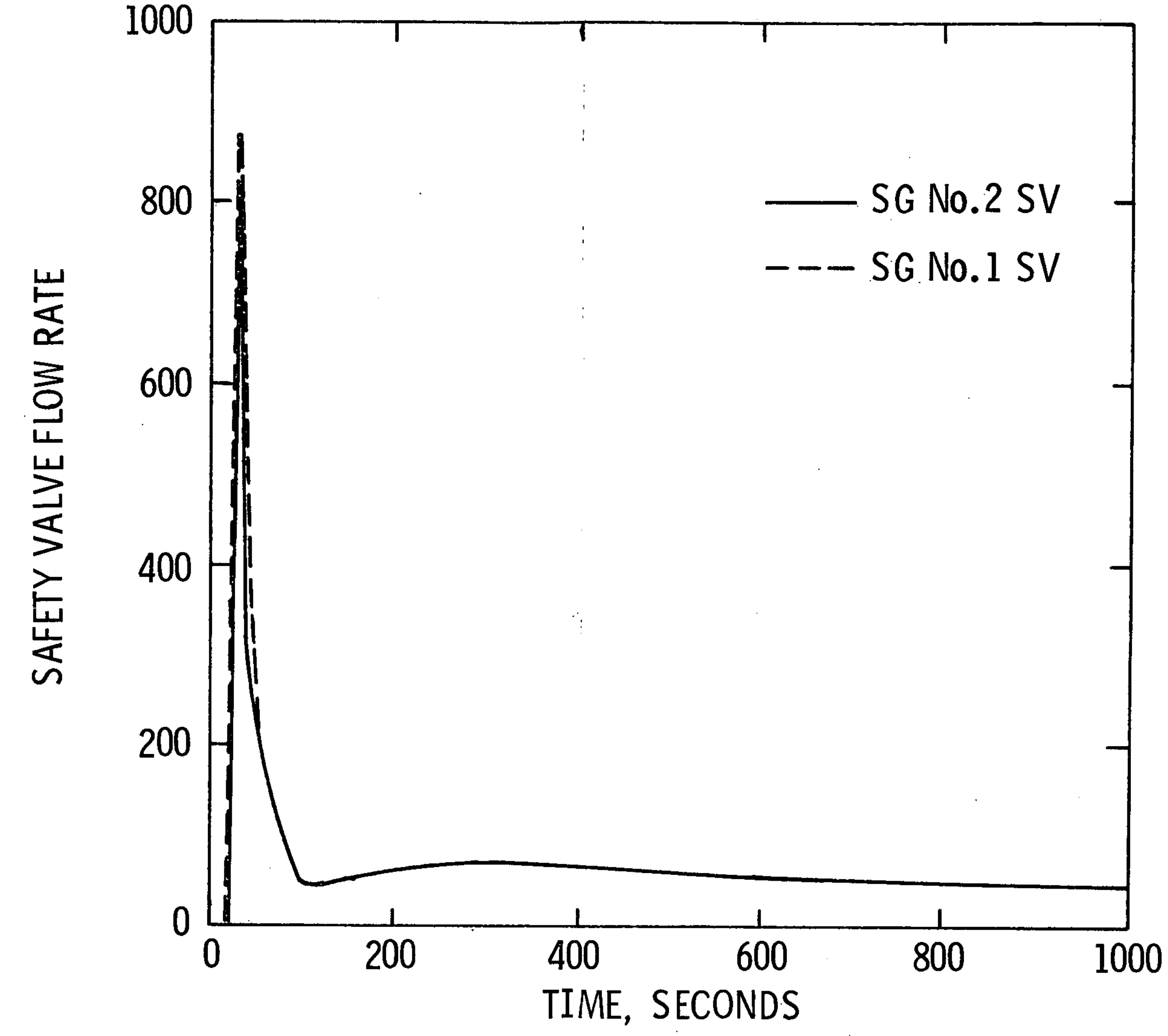


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE STEAM GENERATOR FLUID MASS vs. TIME

Figure 15.1-30

10079-11D-950
200C6

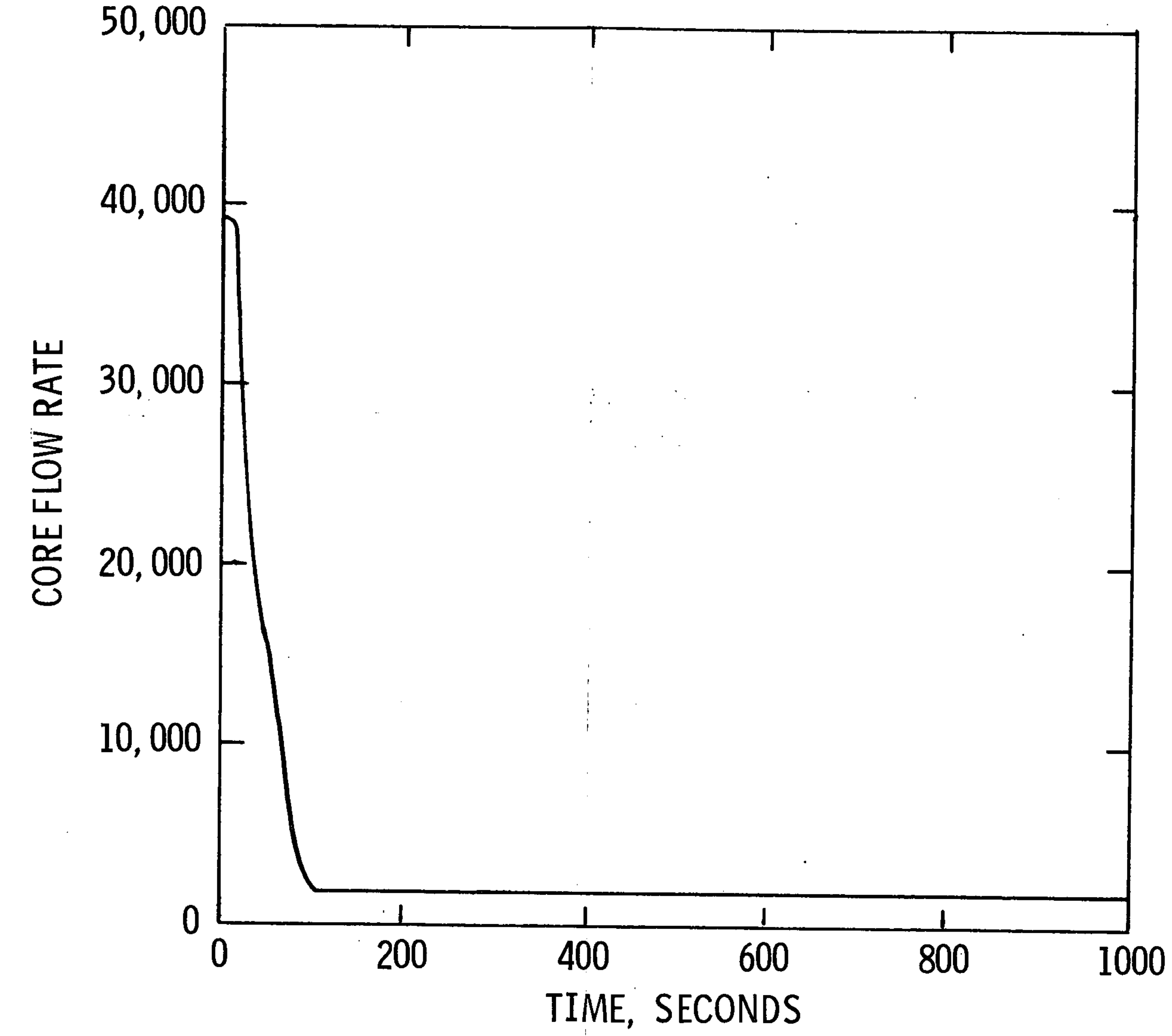


10079-11D-951
20OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE SAFETY VALVE FLOWRATE vs. TIME

Figure 15.1-31

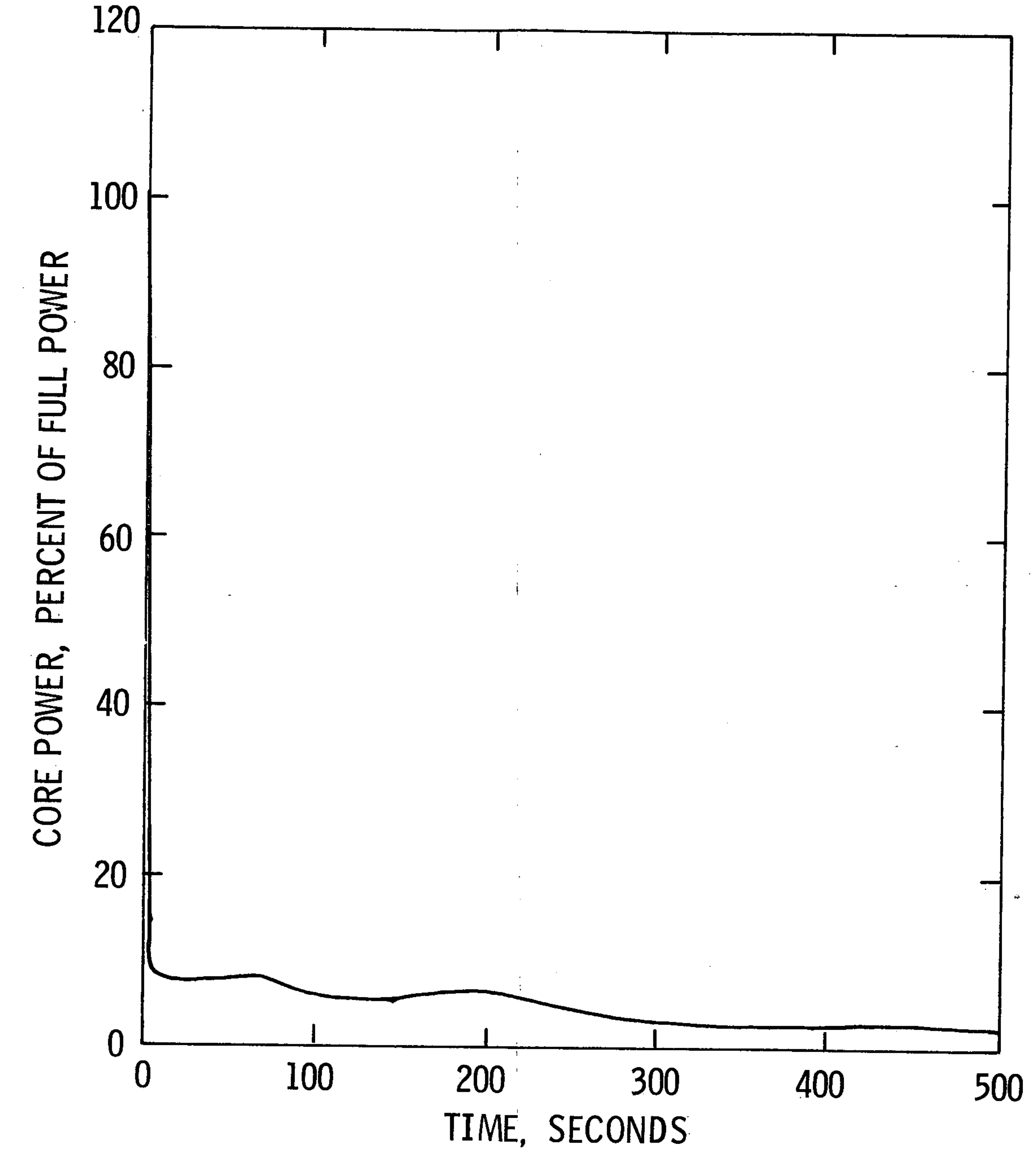


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

INCREASED MAIN STEAM FLOW WITH CONCURRENT SINGLE FAILURE CORE FLOWRATE vs. TIME

Figure 15.1-32

10079-11D-952
20OC6

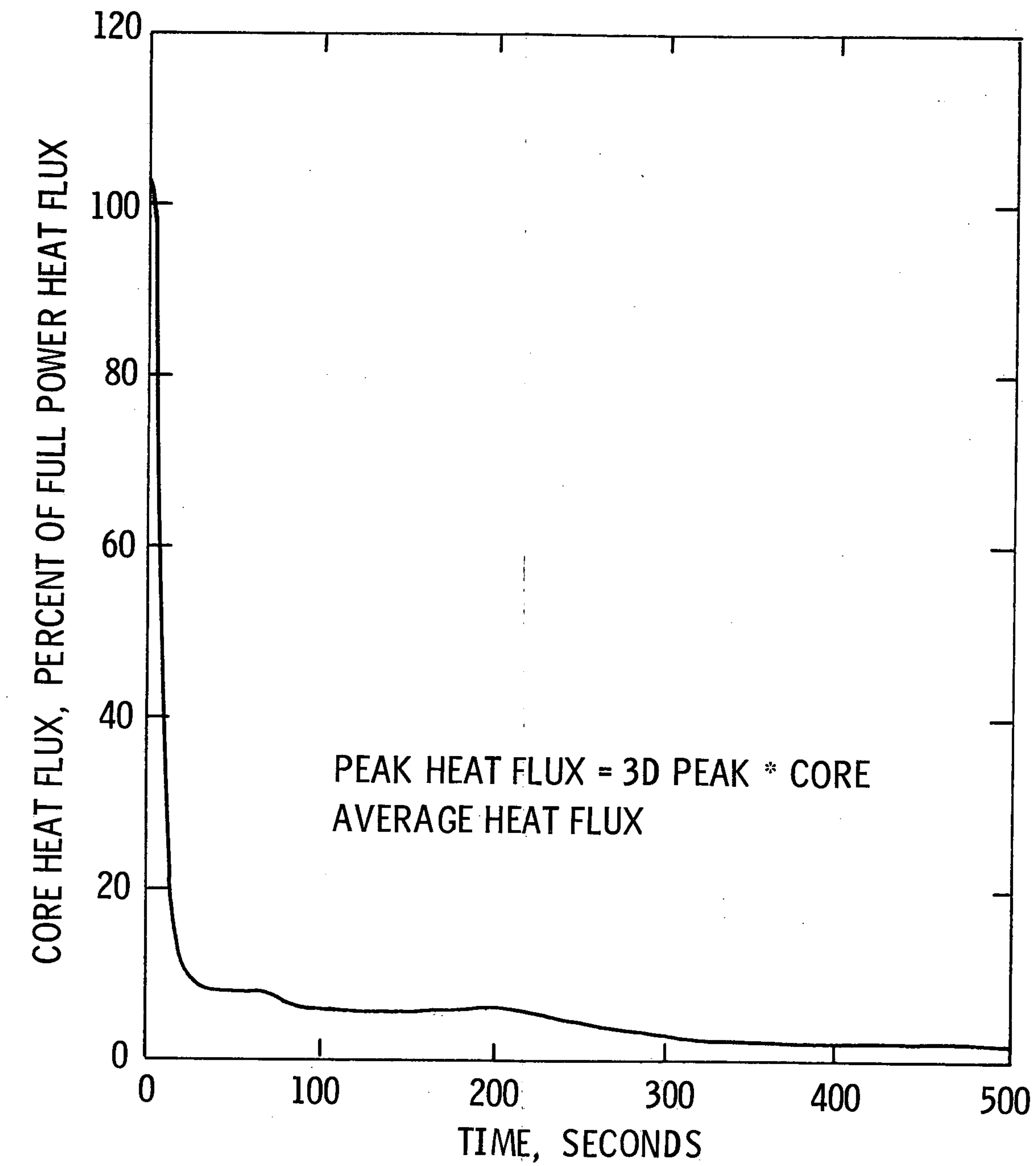


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
CORE POWER vs. TIME

Figure 15.1-33

10079-11D-953
200C6

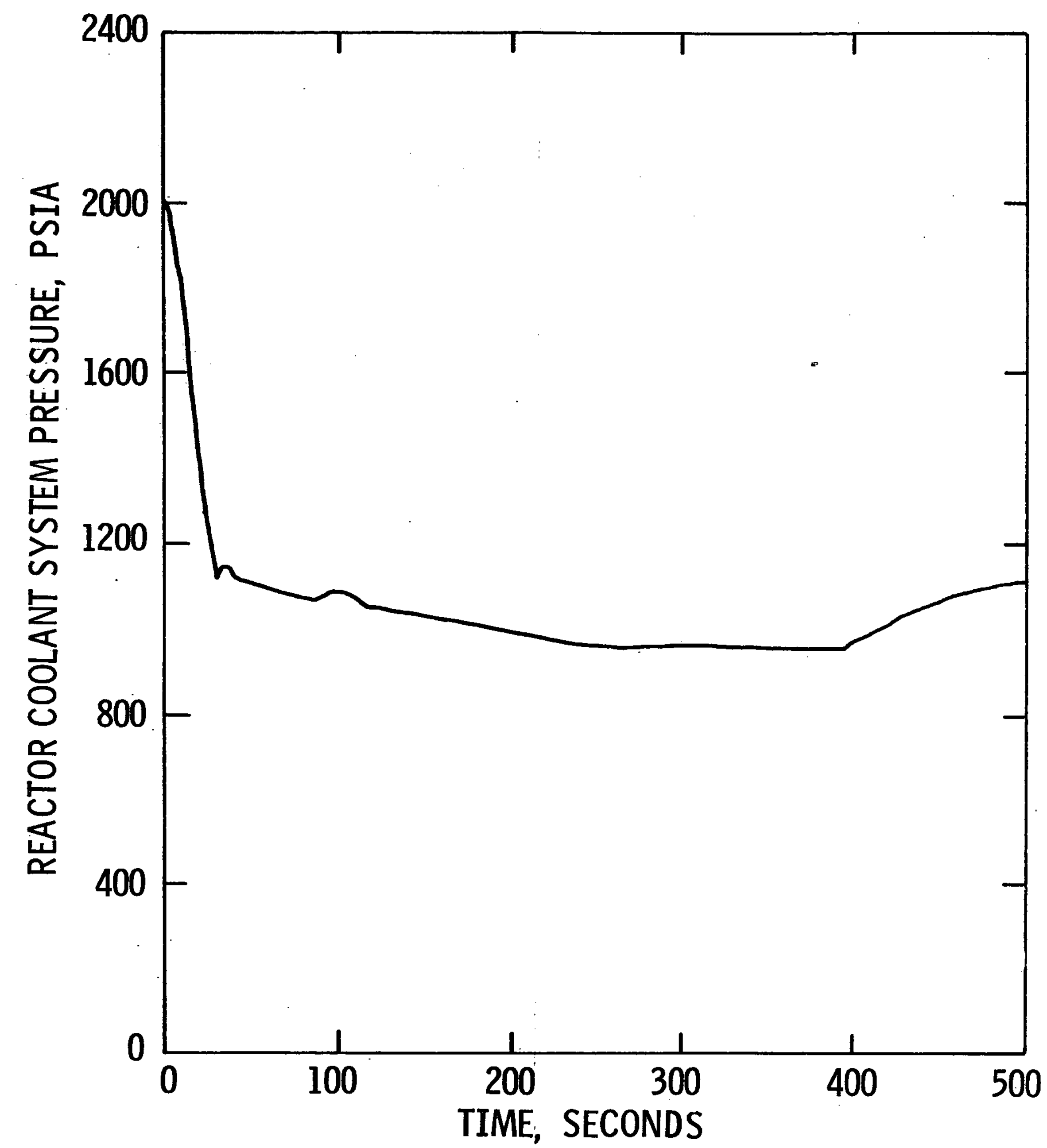


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
CORE HEAT FLUX vs. TIME

Figure 15.1-34

10079-11D-954
20OC6

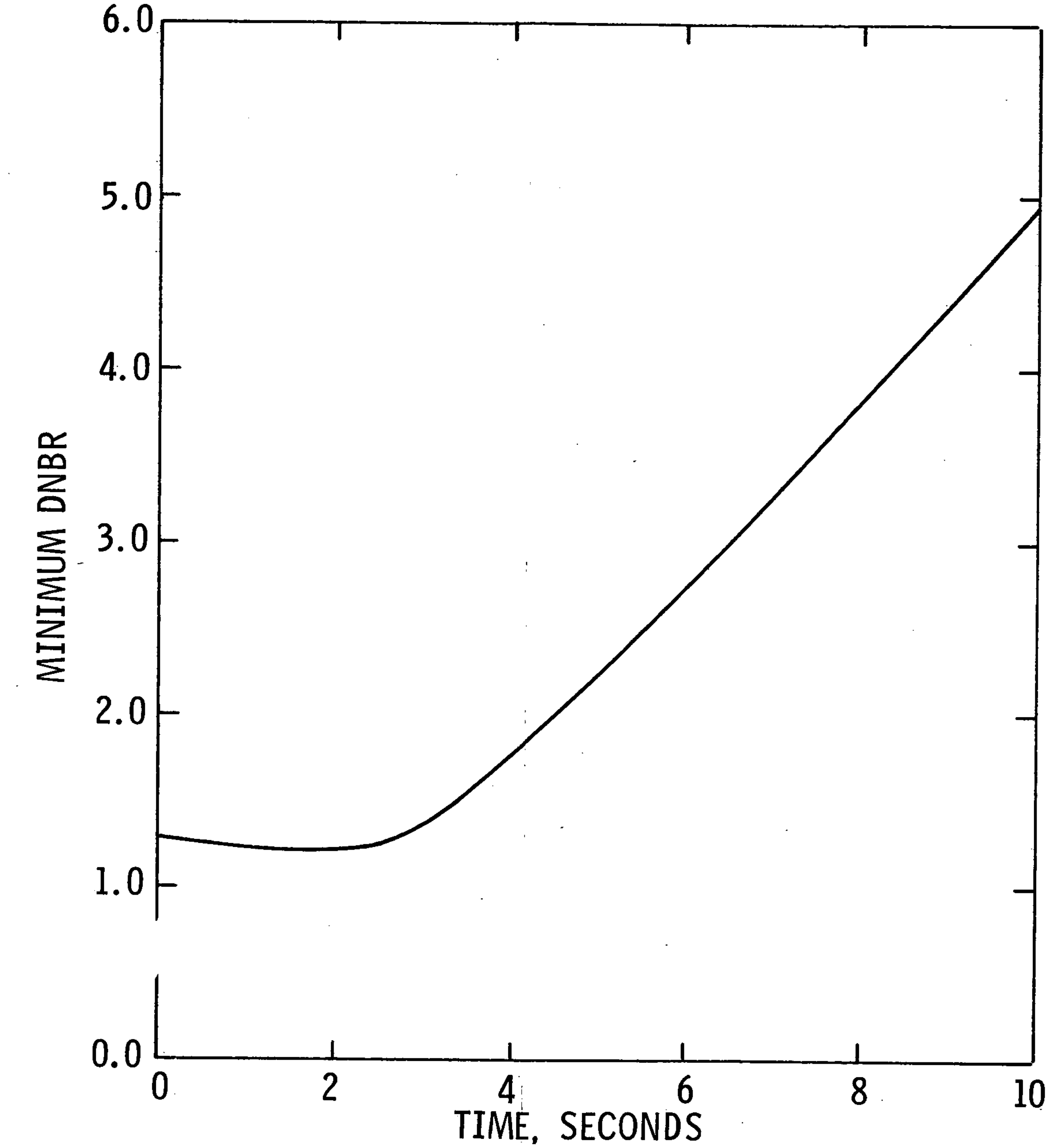


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER REACTOR COOLANT SYSTEM PRESSURE vs. TIME

Figure 15.1-35

10079-11D-955
200C6

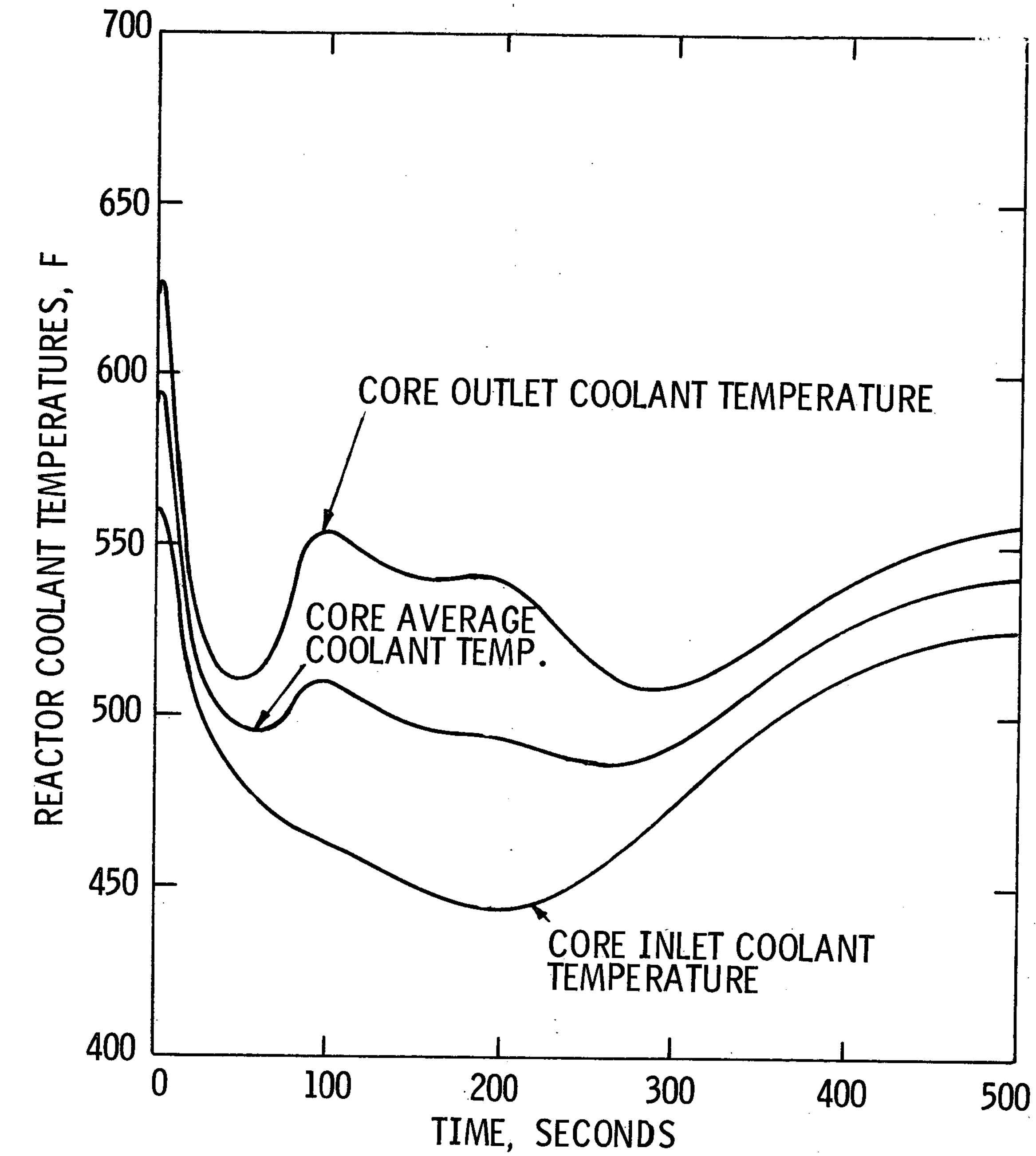


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
DNBR vs. TIME

Figure 15.1-36

10079-11D-956
20OC6

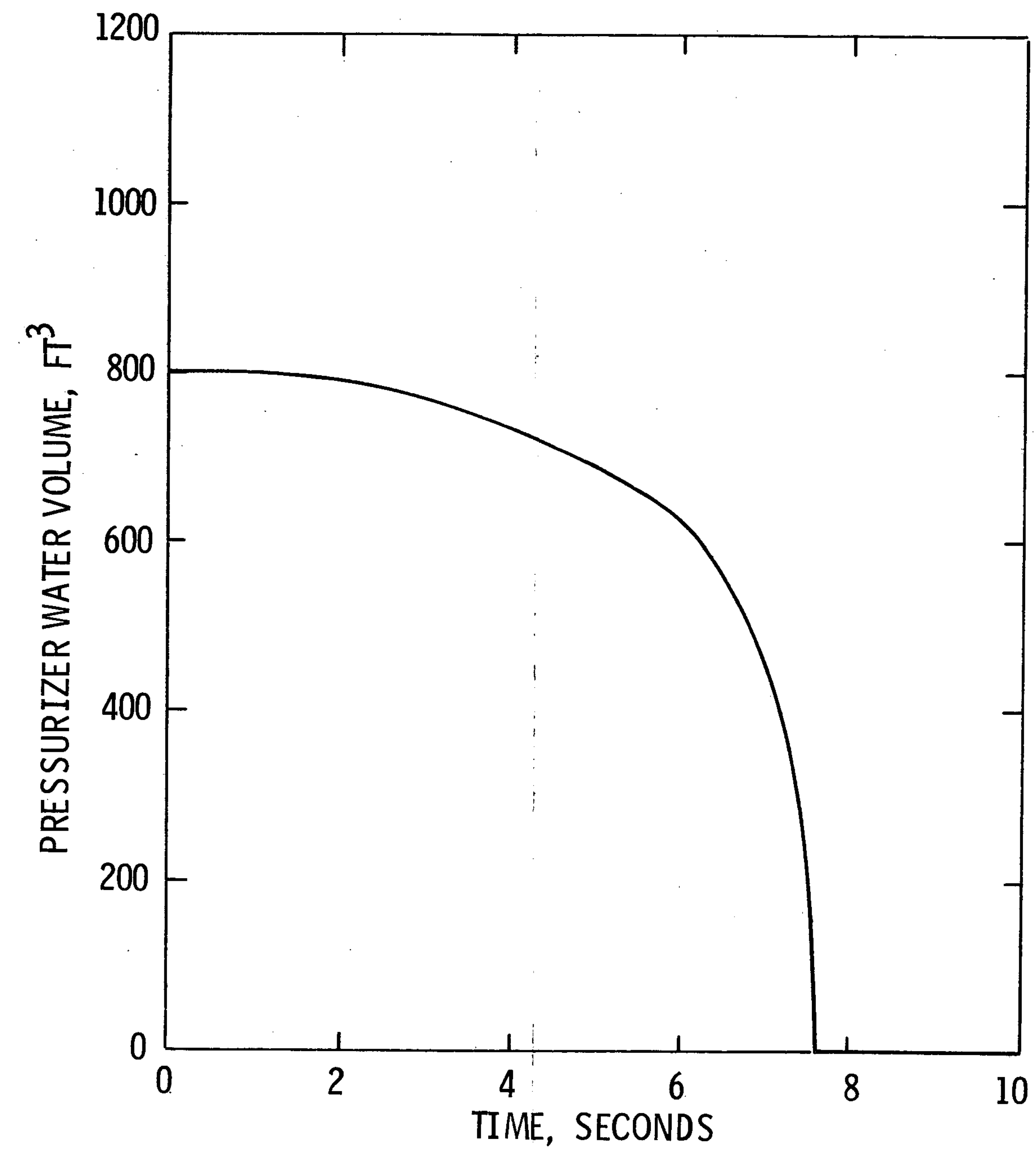


SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3

FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER REACTOR COOLANT TEMPERATURE vs. TIME

Figure 15.1-37

10079-11D-957
20OC6

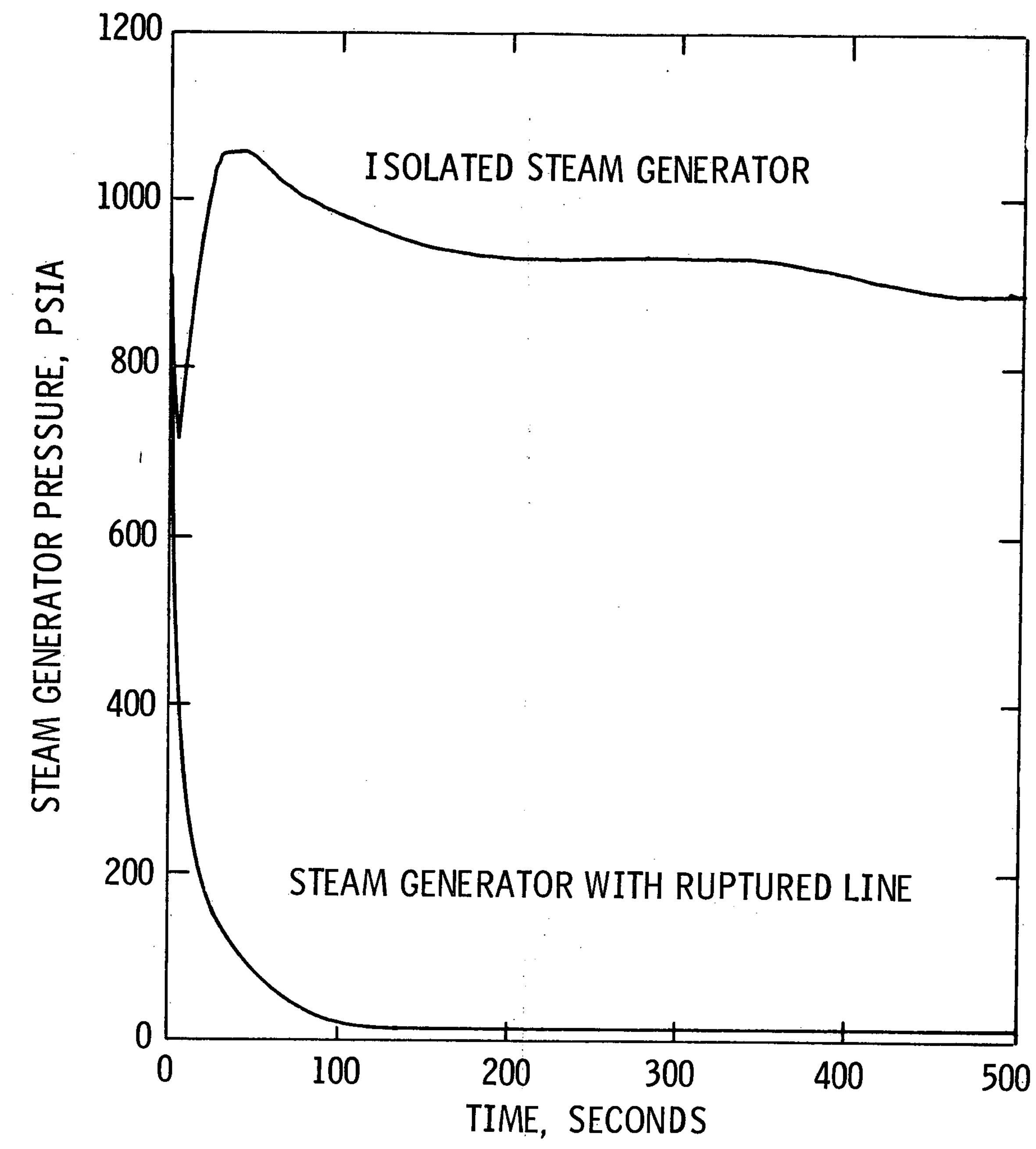


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER PRESSURIZER WATER VOLUME vs. TIME

Figure 15.1-38

10079-11D-958
20OC6

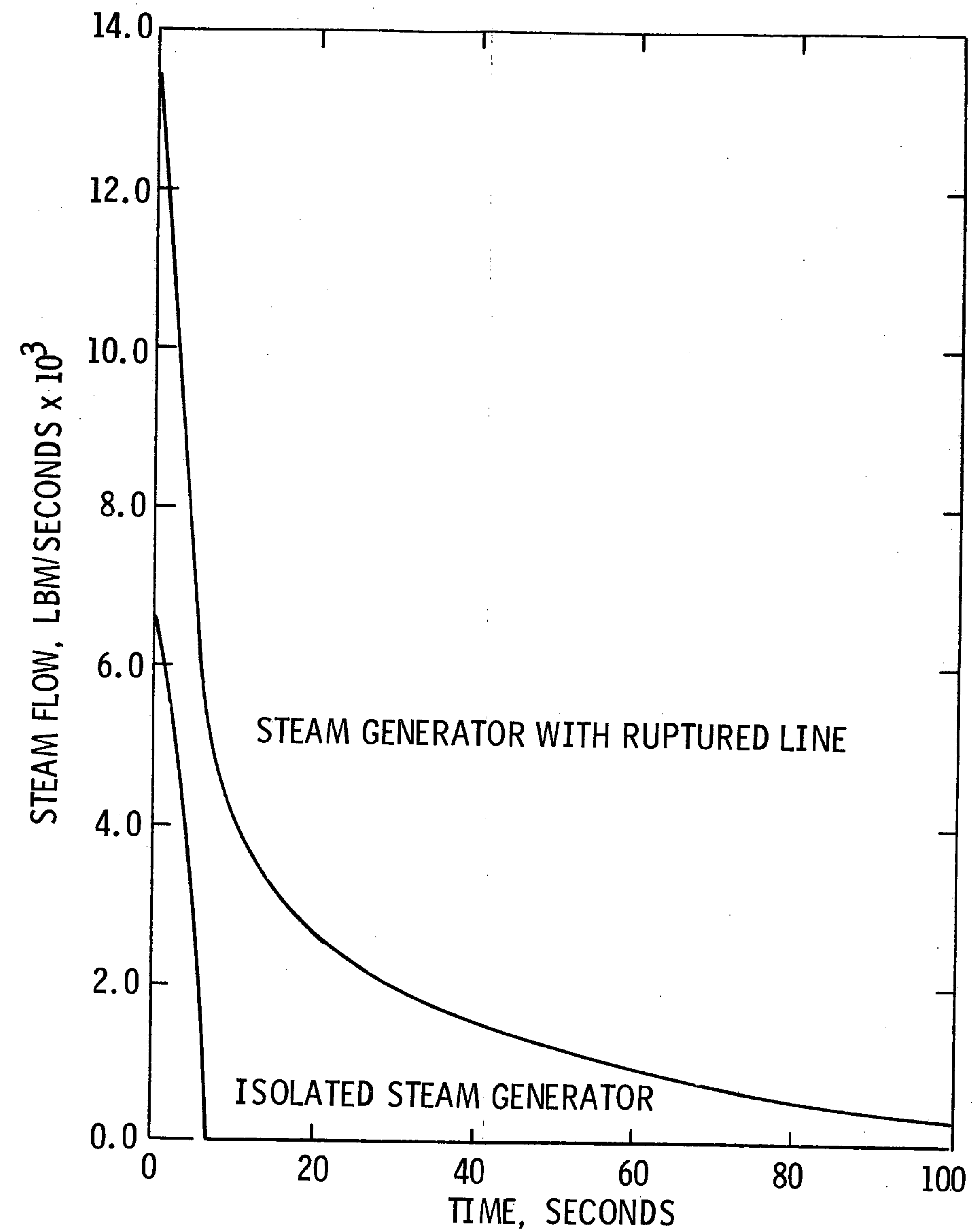


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER STEAM GENERATOR PRESSURE vs. TIME

Figure 15.1-39

10079-11D-959
20OC6

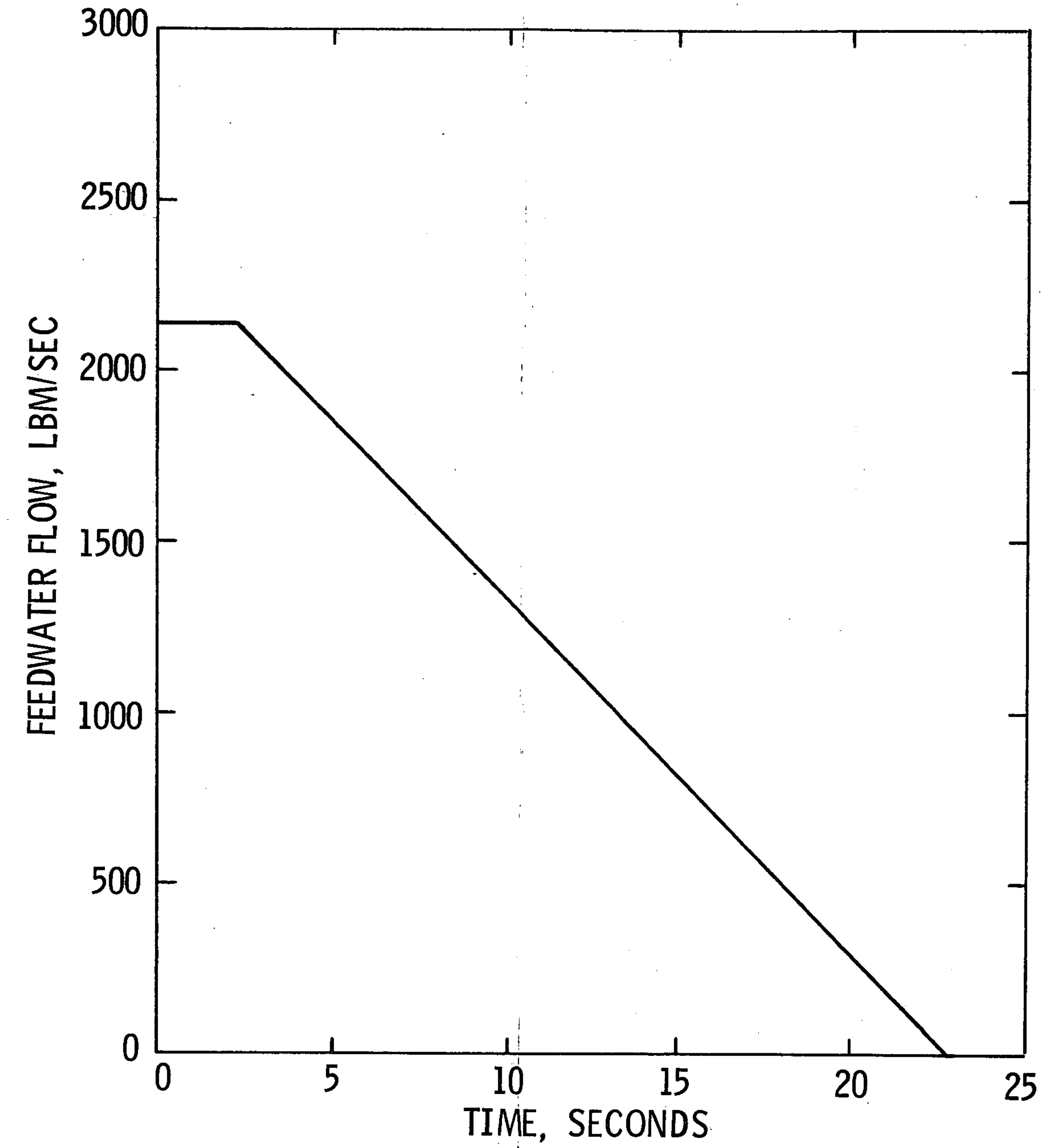


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK WITH
LOSS OF AC POWER STEAM MASS
RELEASE FROM BREAK vs. TIME

Figure 15.1-40

10079-11D-960
20OC6

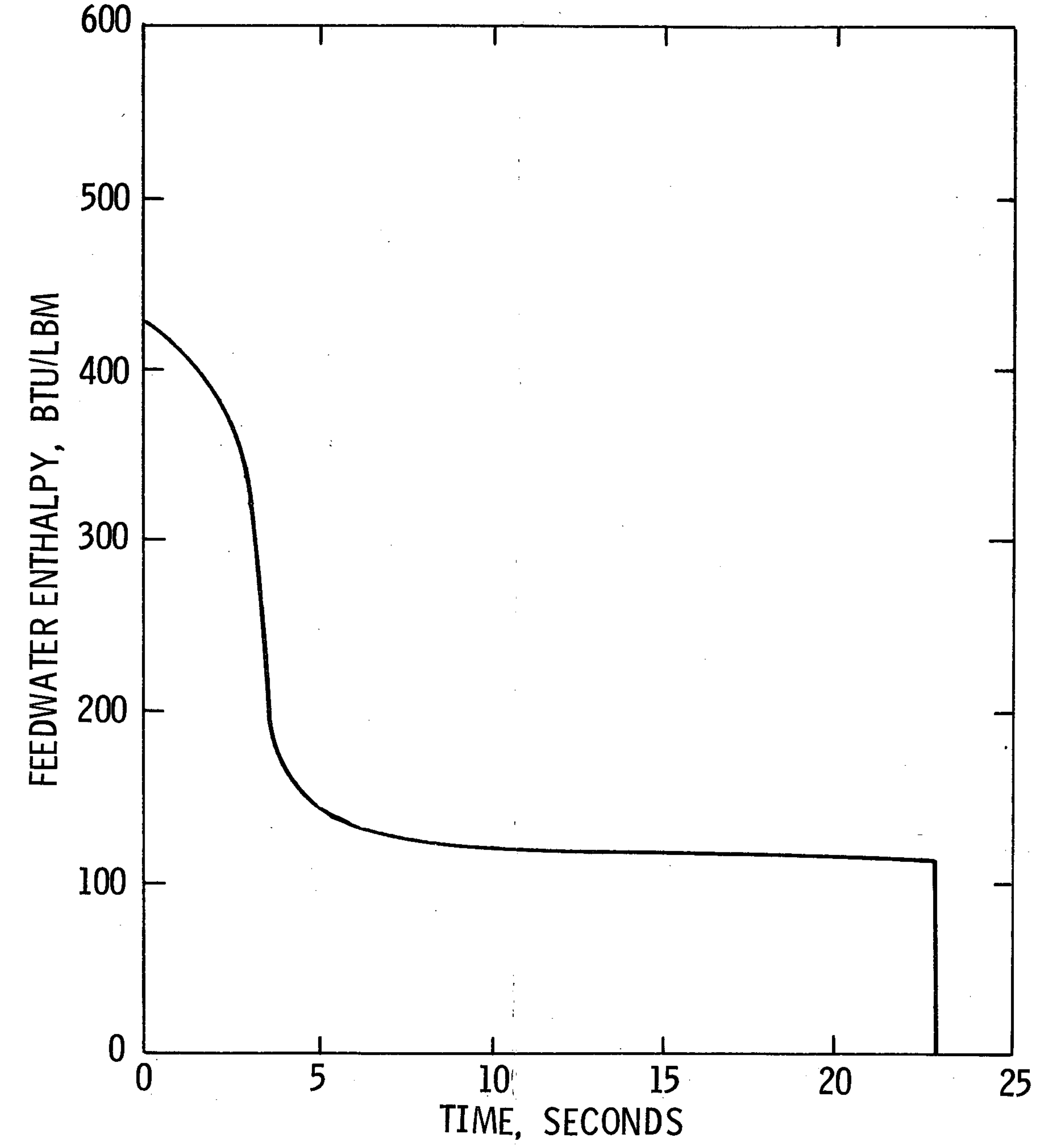


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
FEEDWATER FLOW vs. TIME

Figure 15.1-41

10079-11D-961
20OC6

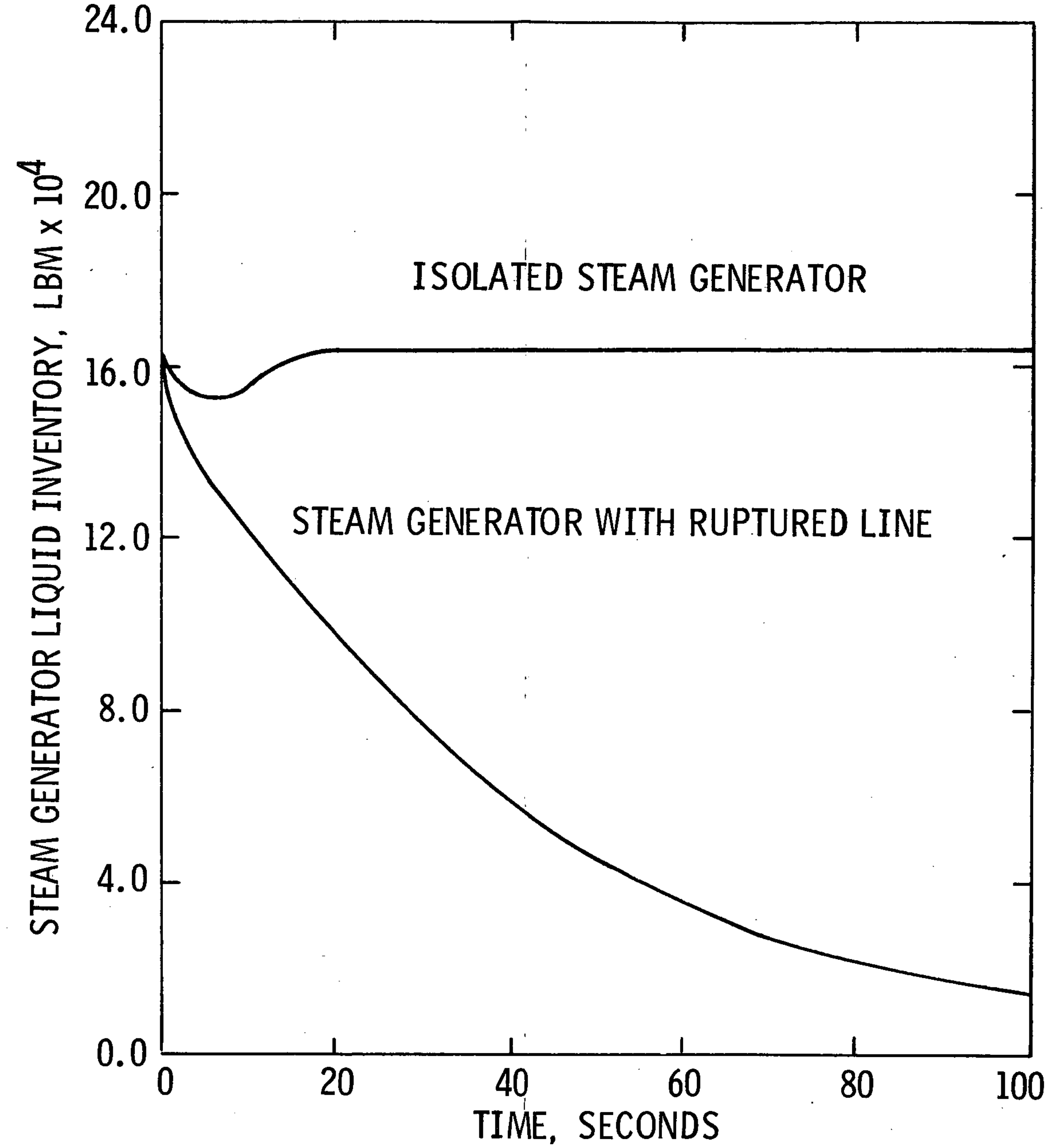


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
FEEDWATER ENTHALPY vs. TIME

Figure 15.1-42

10079-11D-962
200C6

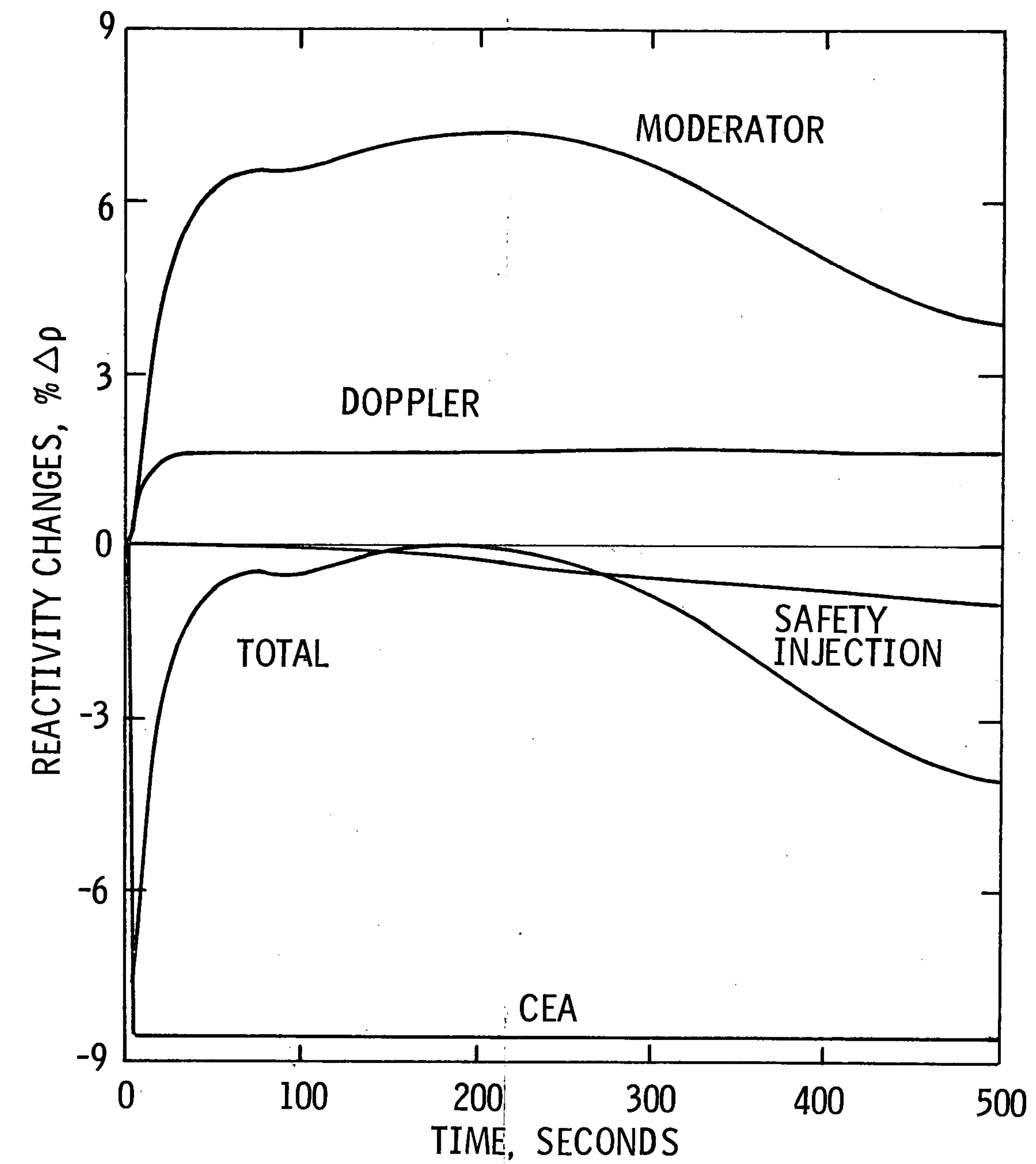


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER STEAM GENERATOR LIQUID INVENTORY vs. TIME

Figure 15.1-43

10079-11D-963
20OC6

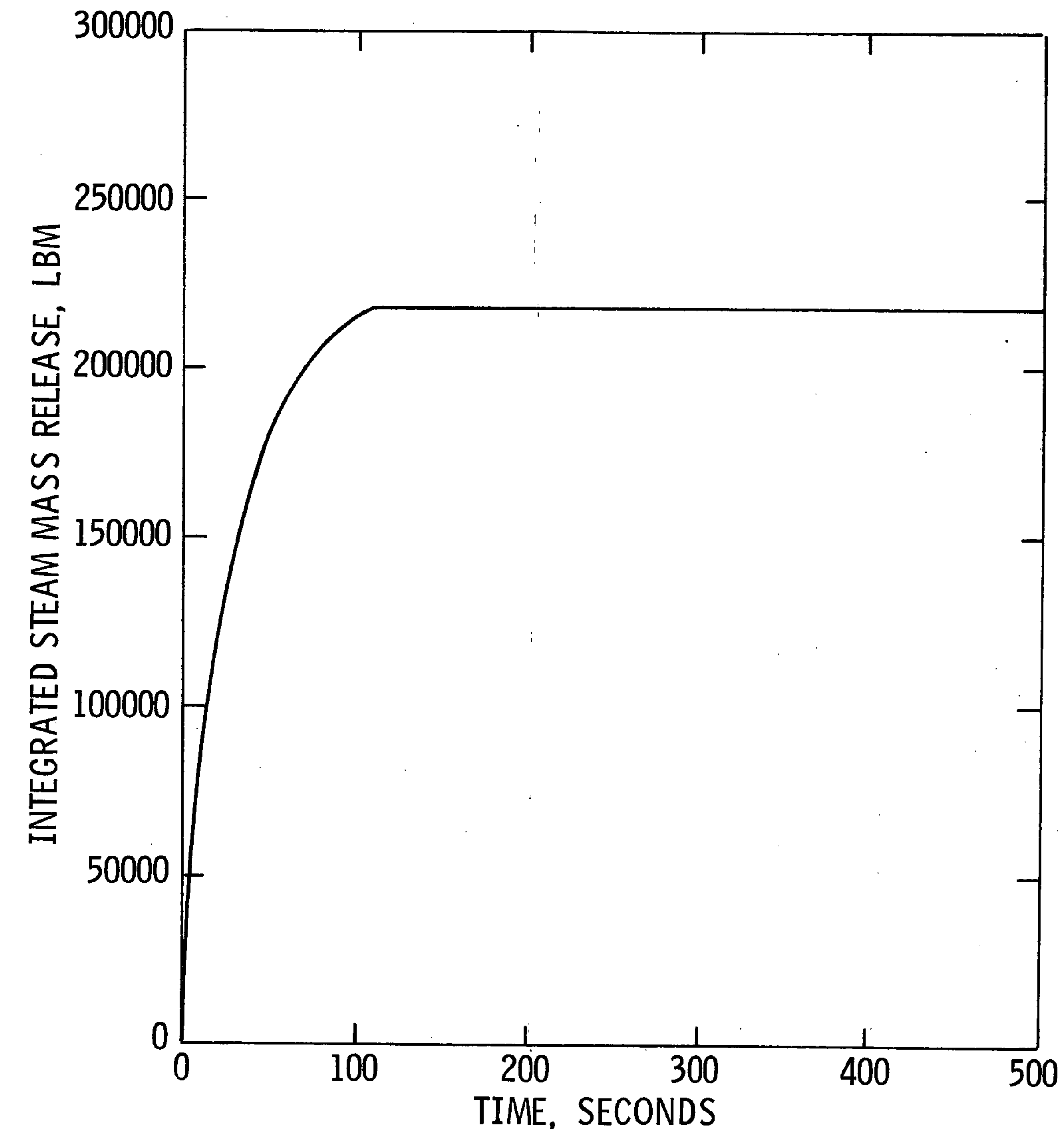


10079-11D-964
20OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
WITH LOSS OF AC POWER
REACTIVITY vs. TIME

Figure 15.1-44

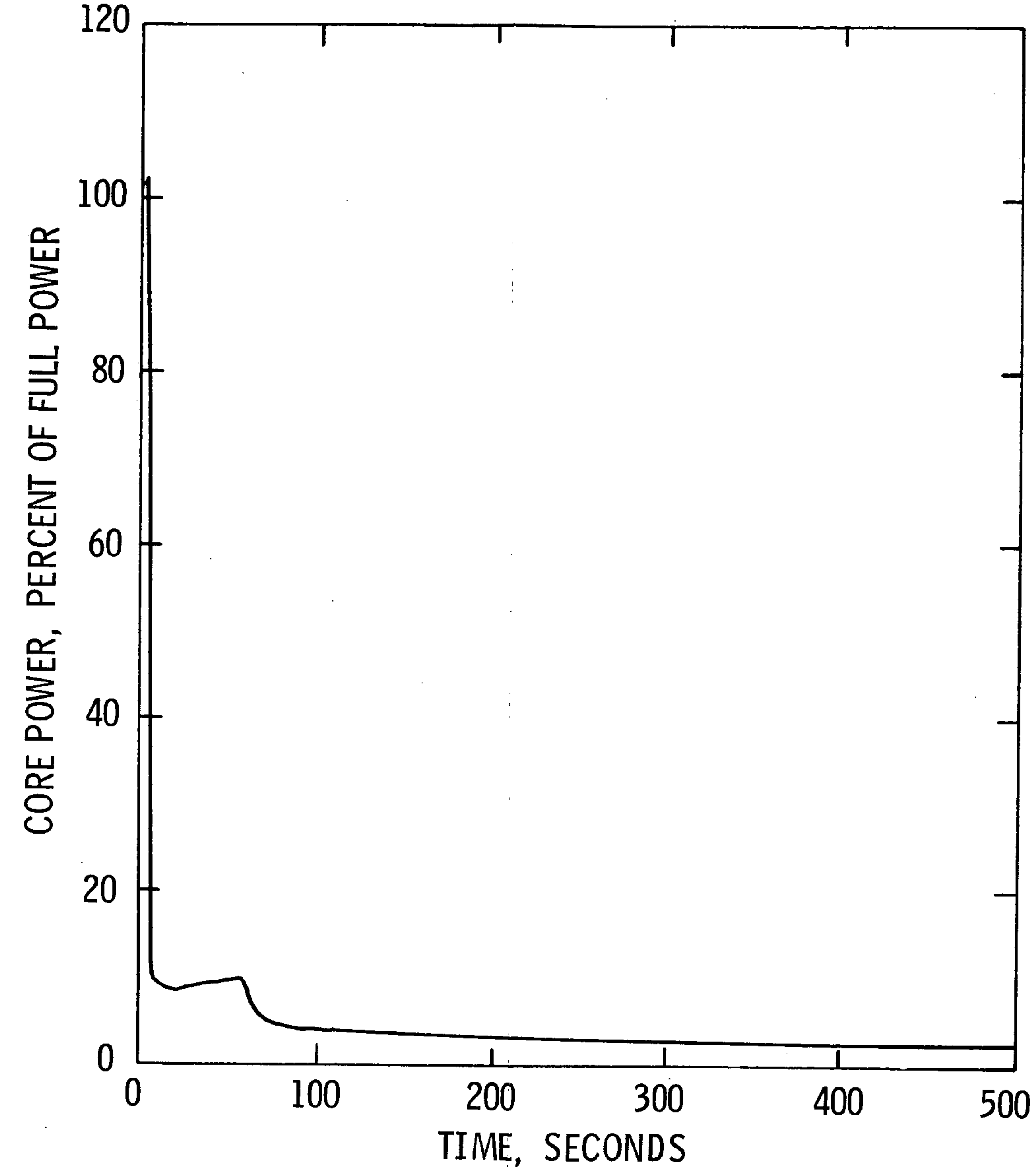


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK WITH LOSS OF AC POWER - INTEGRATED STEAM MASS RELEASE FROM BREAK vs. TIME

Figure 15.1-45

10079-11D-965
200C6

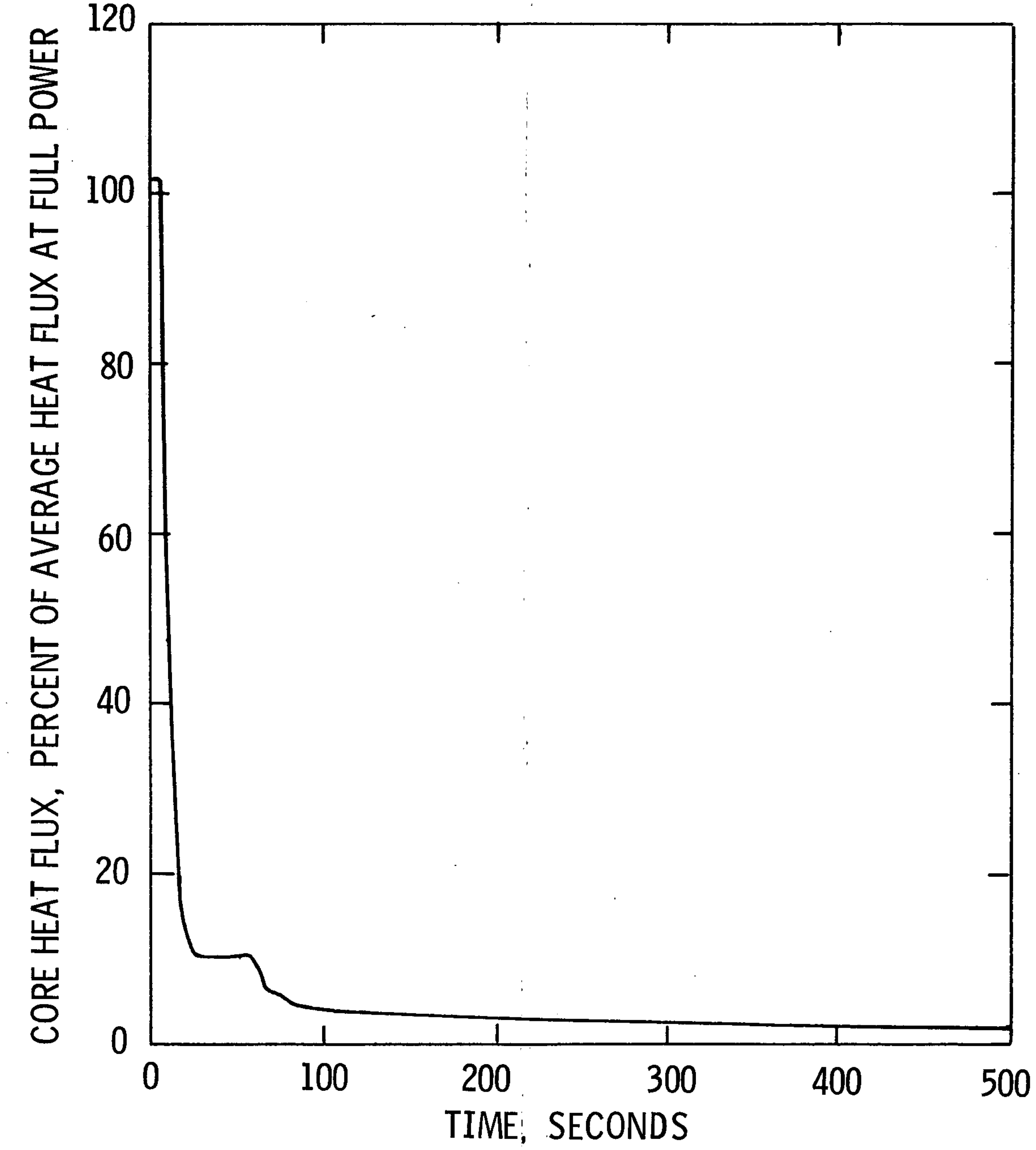


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
CORE POWER vs. TIME

Figure 15.1-46

10079-11D-966
20OC6

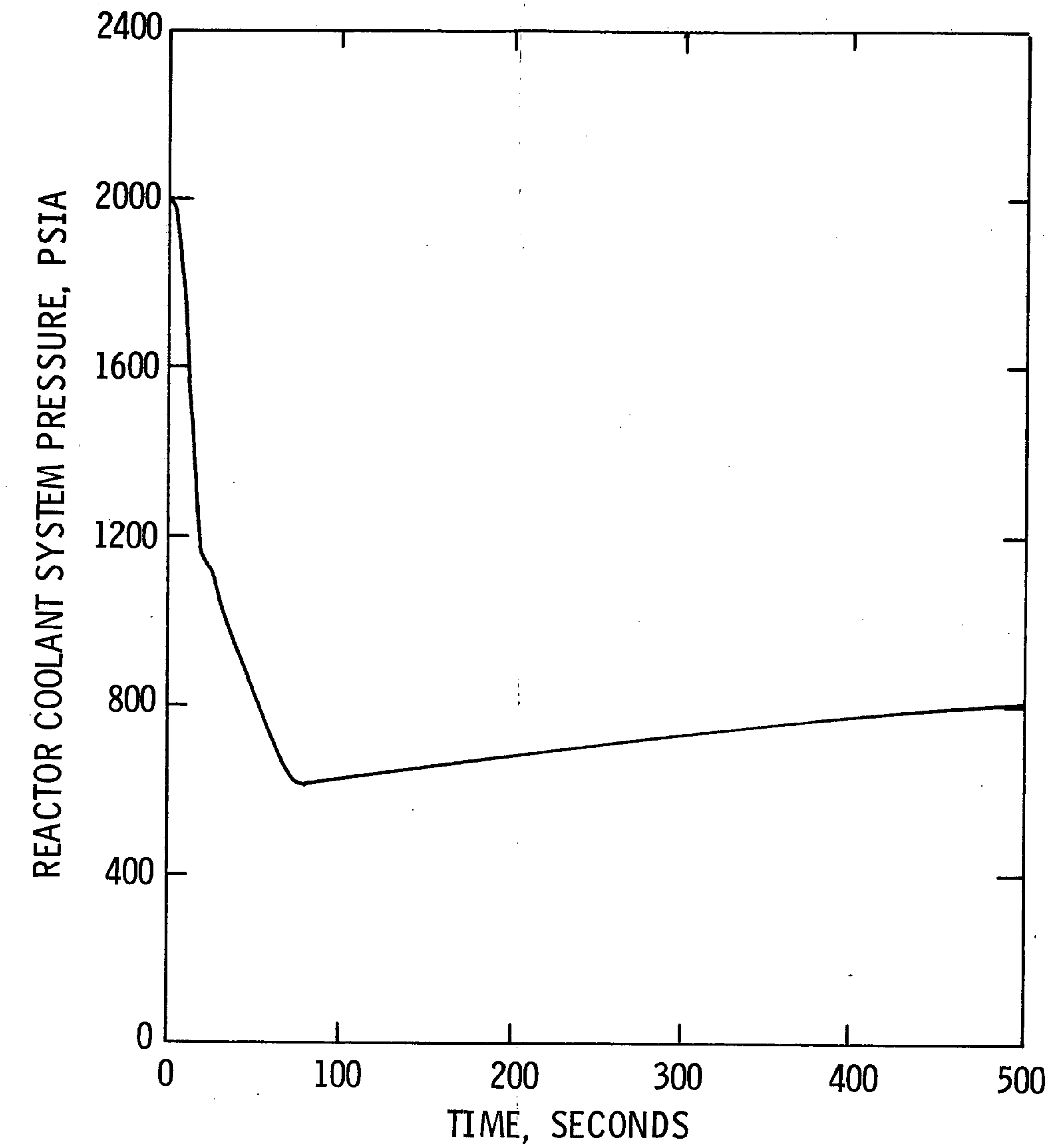


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
CORE HEAT FLUX vs. TIME

Figure 15.1-47

10079-11D-967
20OC6

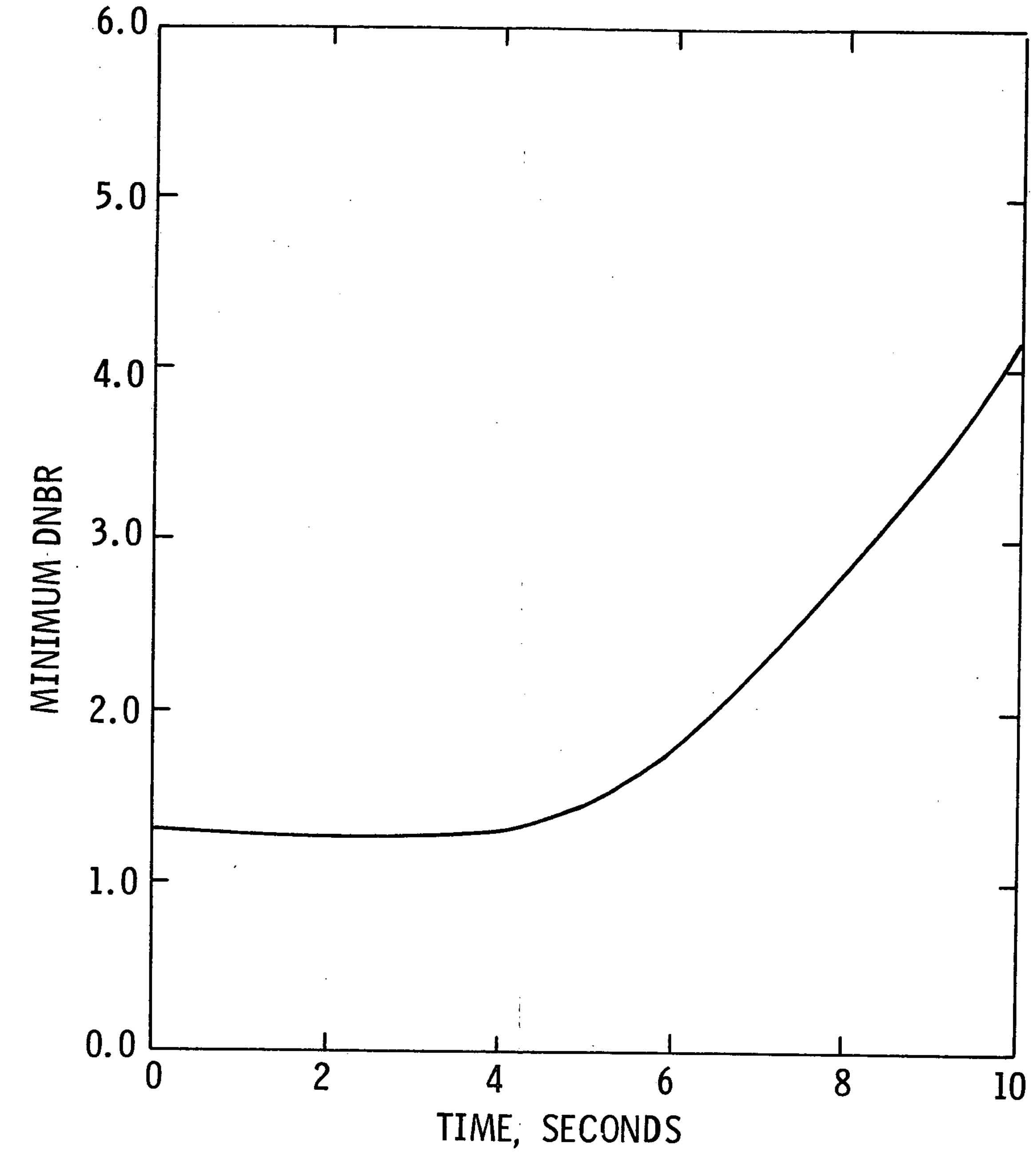


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
REACTOR COOLANT SYSTEM
PRESSURE vs. TIME

Figure 15.1-48

10079-11D-968
20OC6

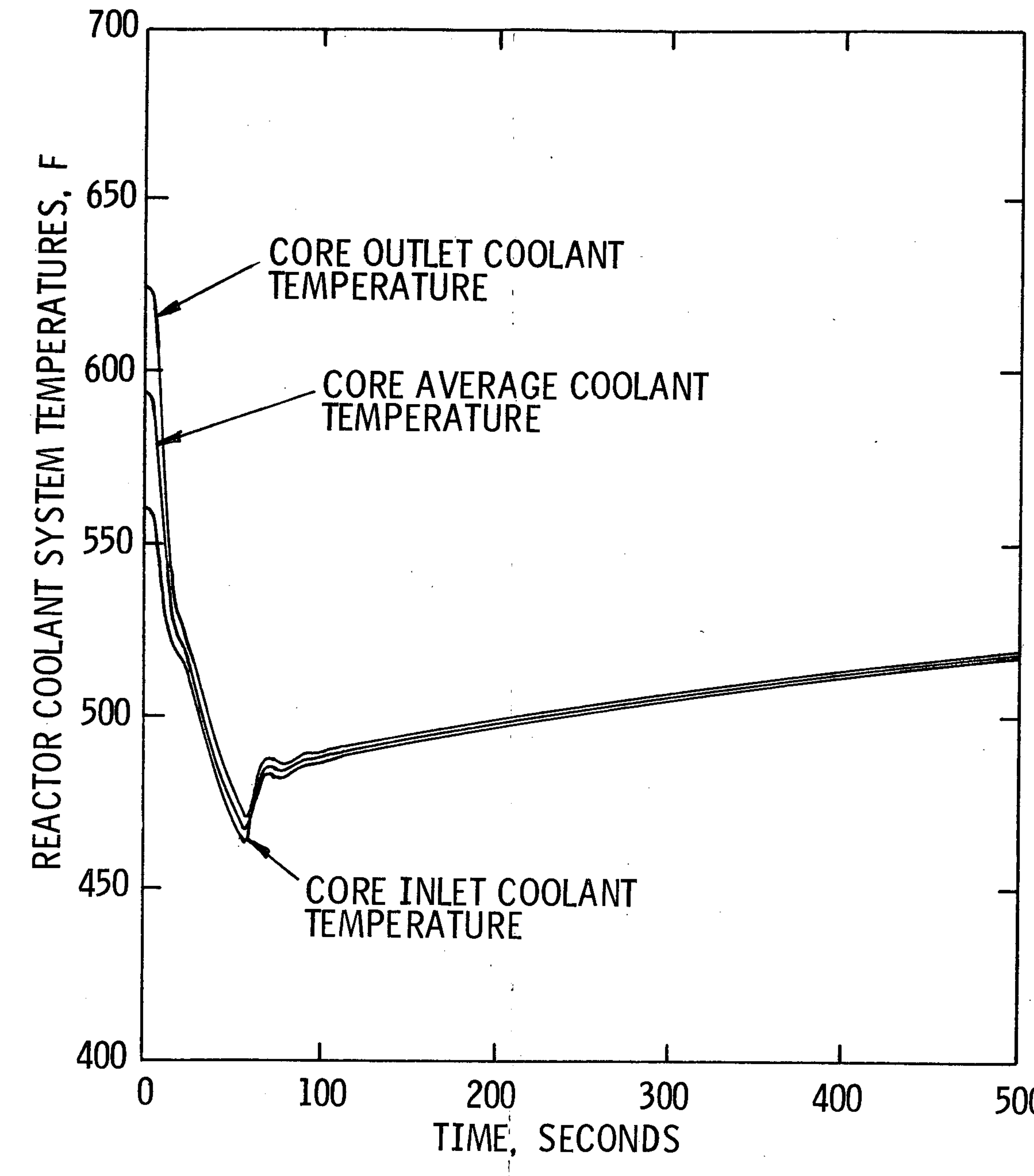


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
DNBR vs. TIME

Figure 15.1-49

10079-11D-969
20OC6

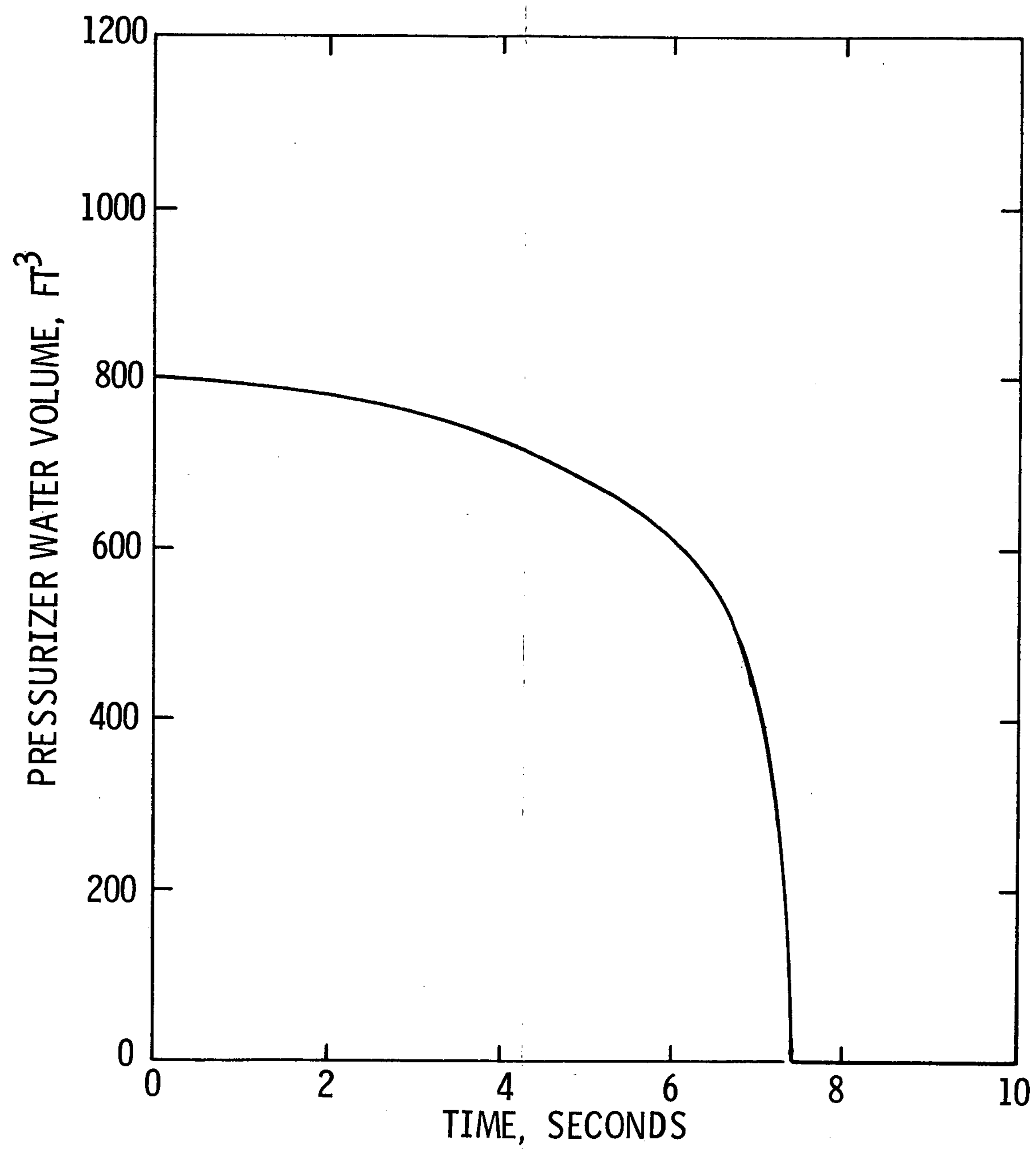


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
REACTOR COOLANT SYSTEM
TEMPERATURE vs. TIME

Figure 15.1-50

10079-11D-970
200C6

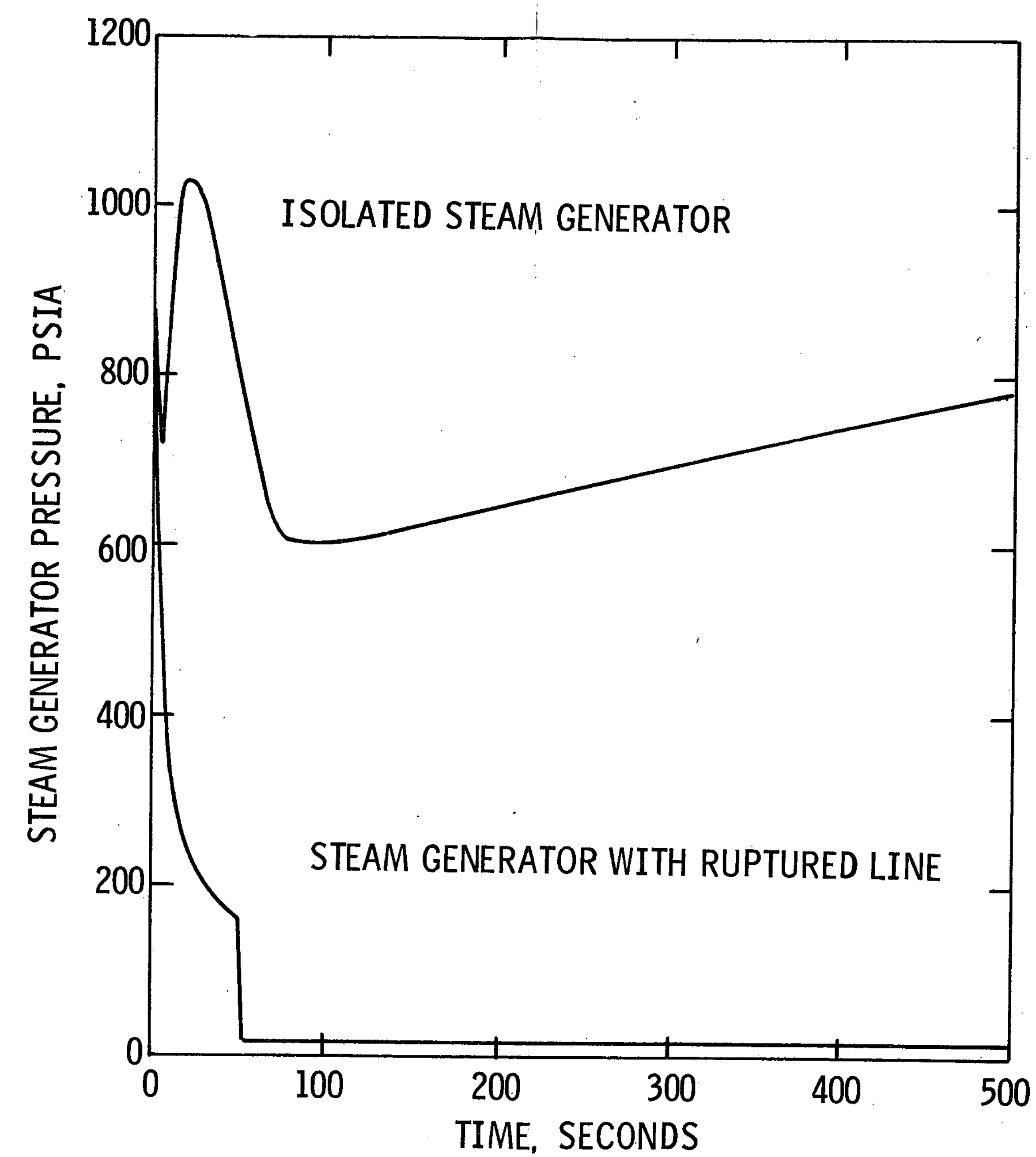


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
PRESSURIZER WATER
VOLUME vs. TIME

Figure 15.1-51

10079-11D-971
20OC6

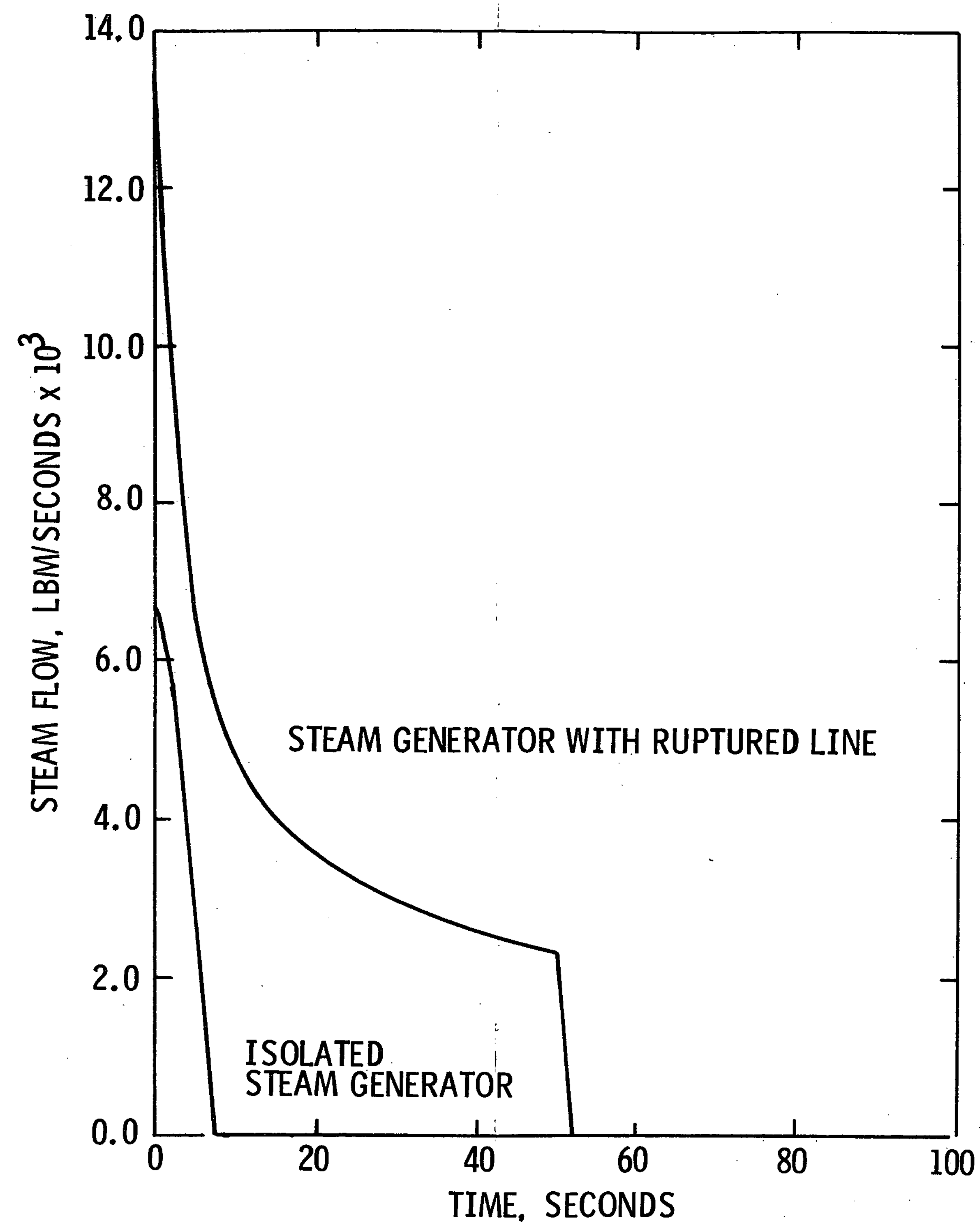


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
STEAM GENERATOR
PRESSURE vs. TIME

Figure 15.1-52

10079-11D-972
200C6

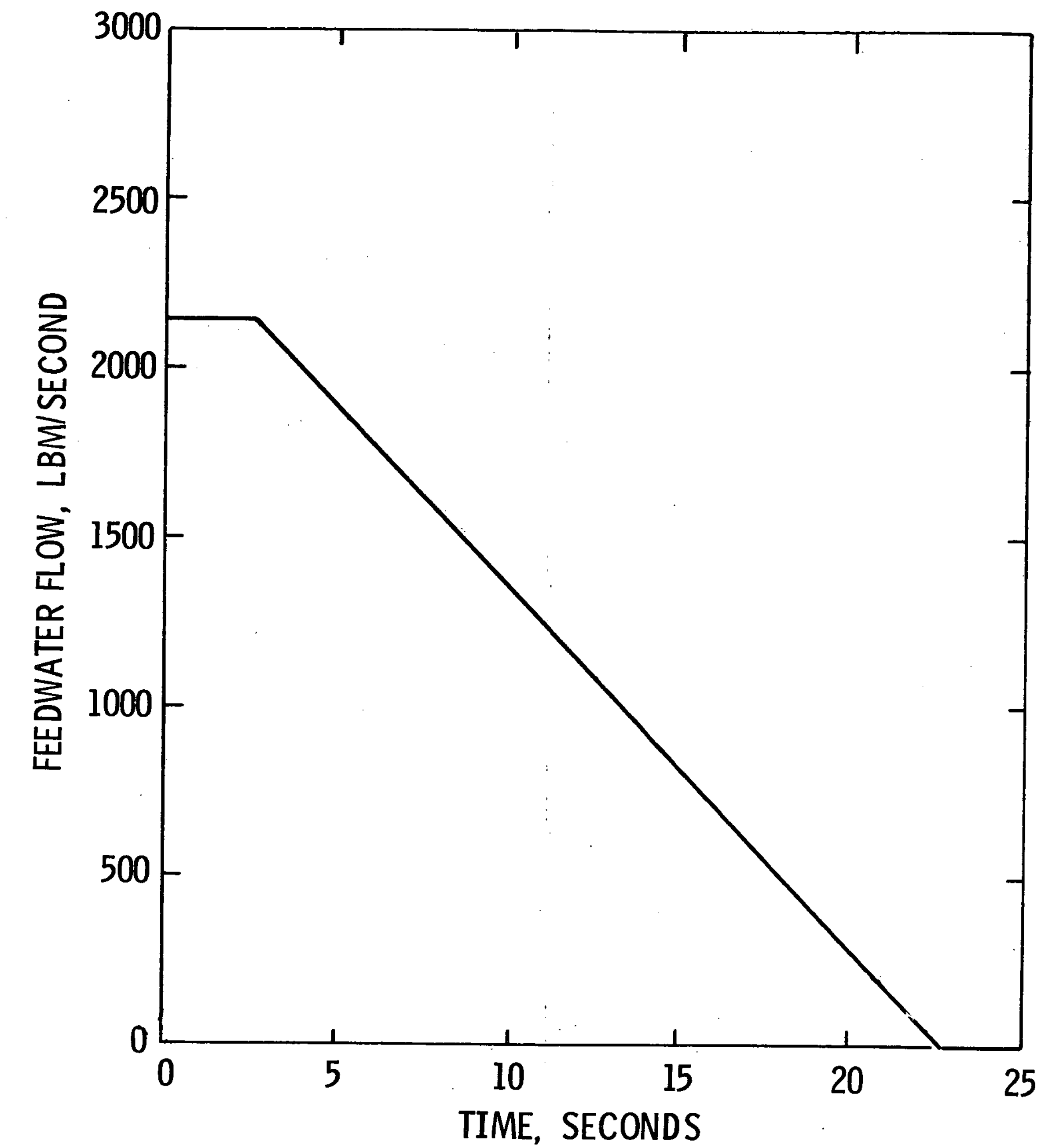


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
STEAM MASS RELEASE FROM
BREAK vs. TIME

Figure 15.1-53

10079-11D-973
20OC6

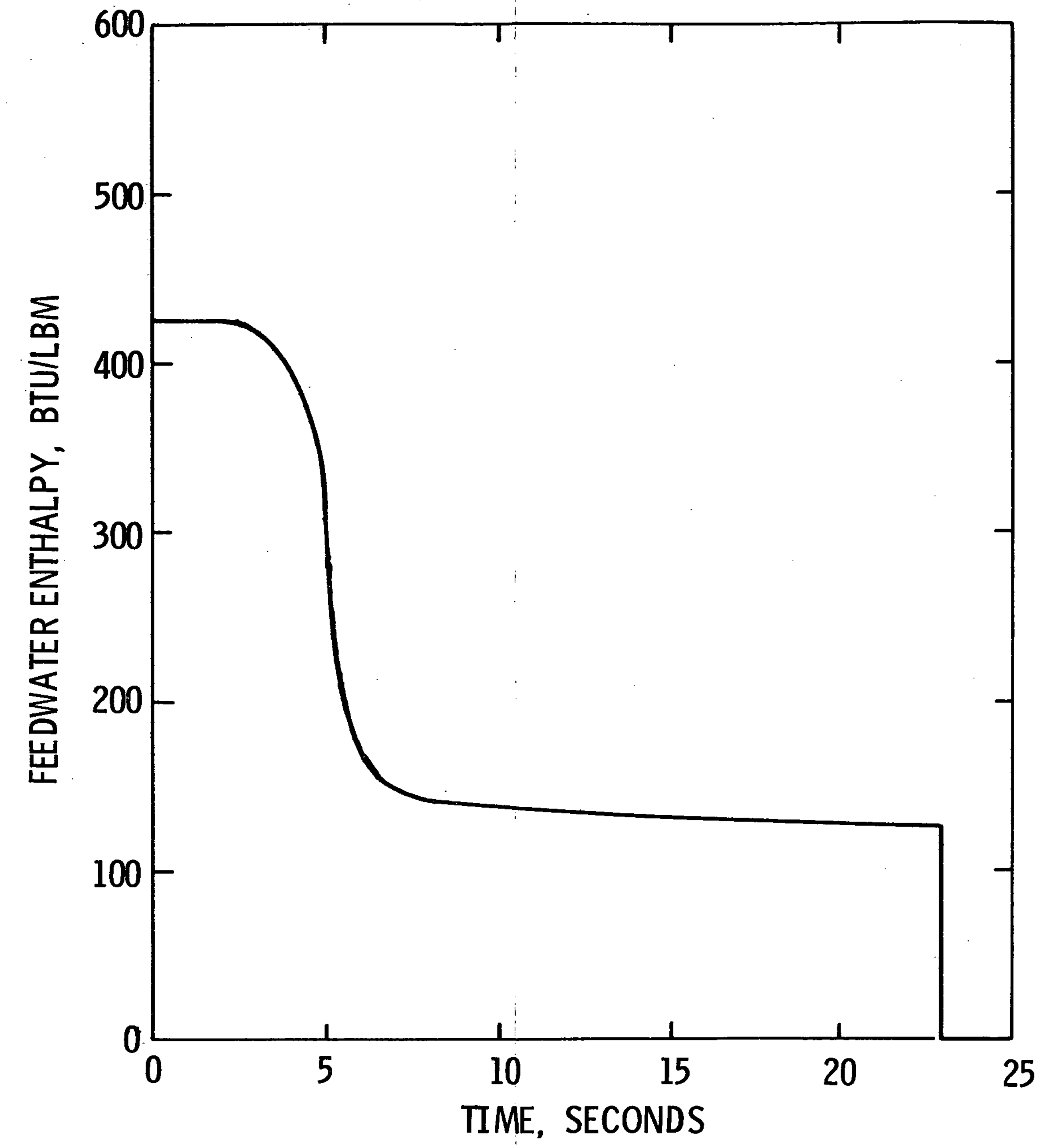


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
FEEDWATER FLOW vs. TIME

Figure 15.1-54

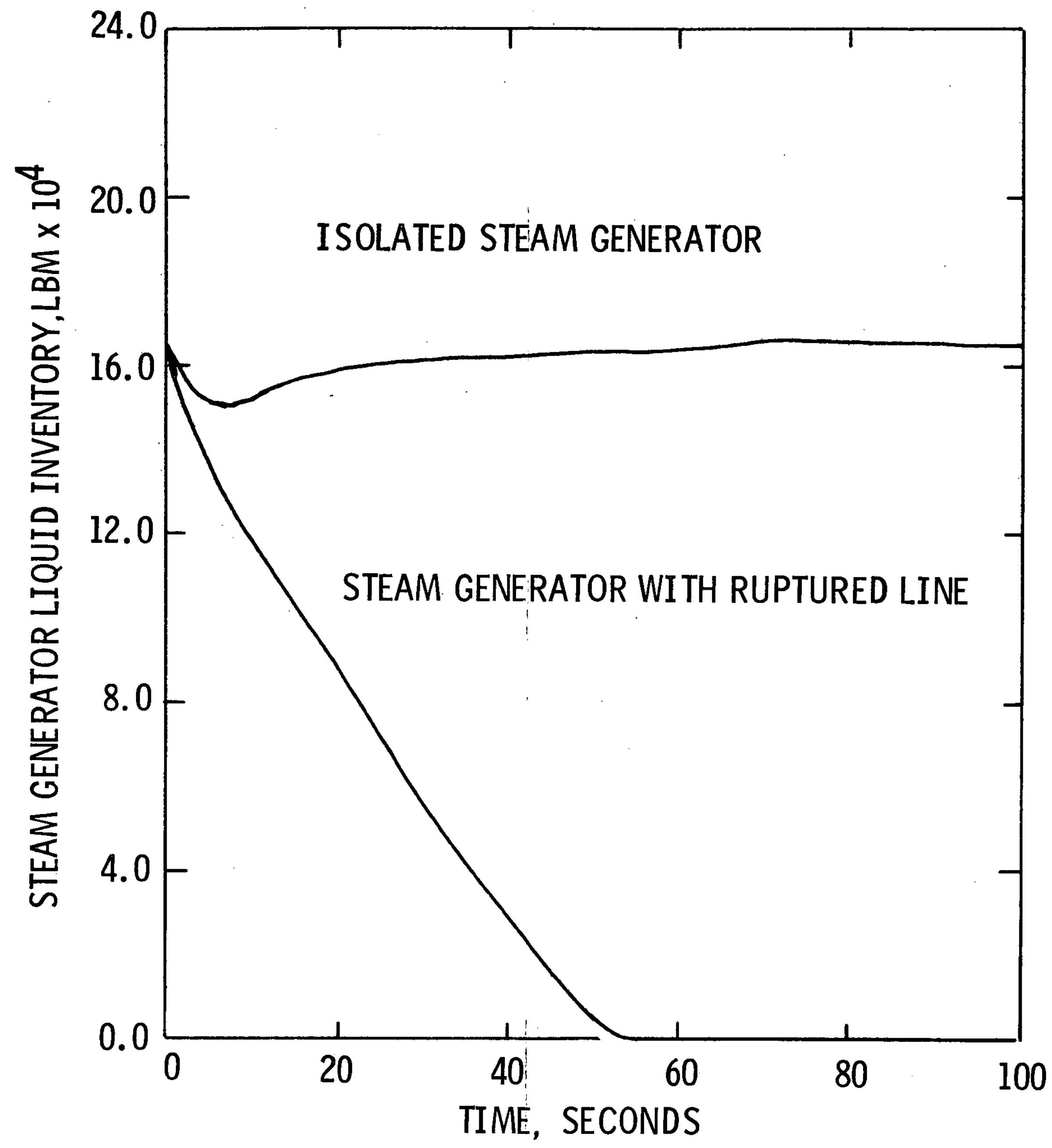
10079-11D-974
20OC6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
FEEDWATER ENTHALPY vs. TIME

Figure 15.1-55

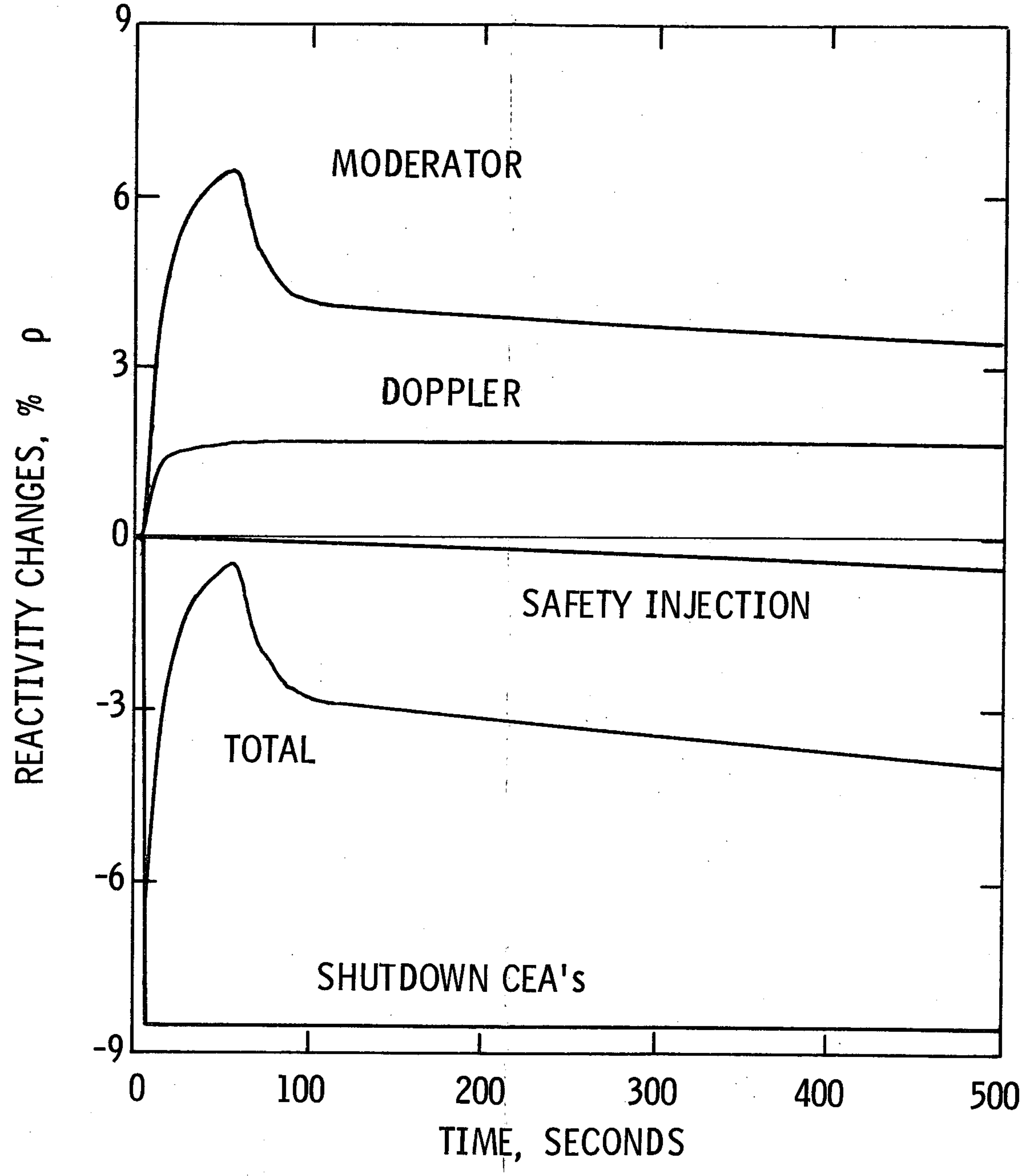


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
STEAM GENERATOR LIQUID
INVENTORY vs. TIME

Figure 15.1-56

10079-11D-976
200C6

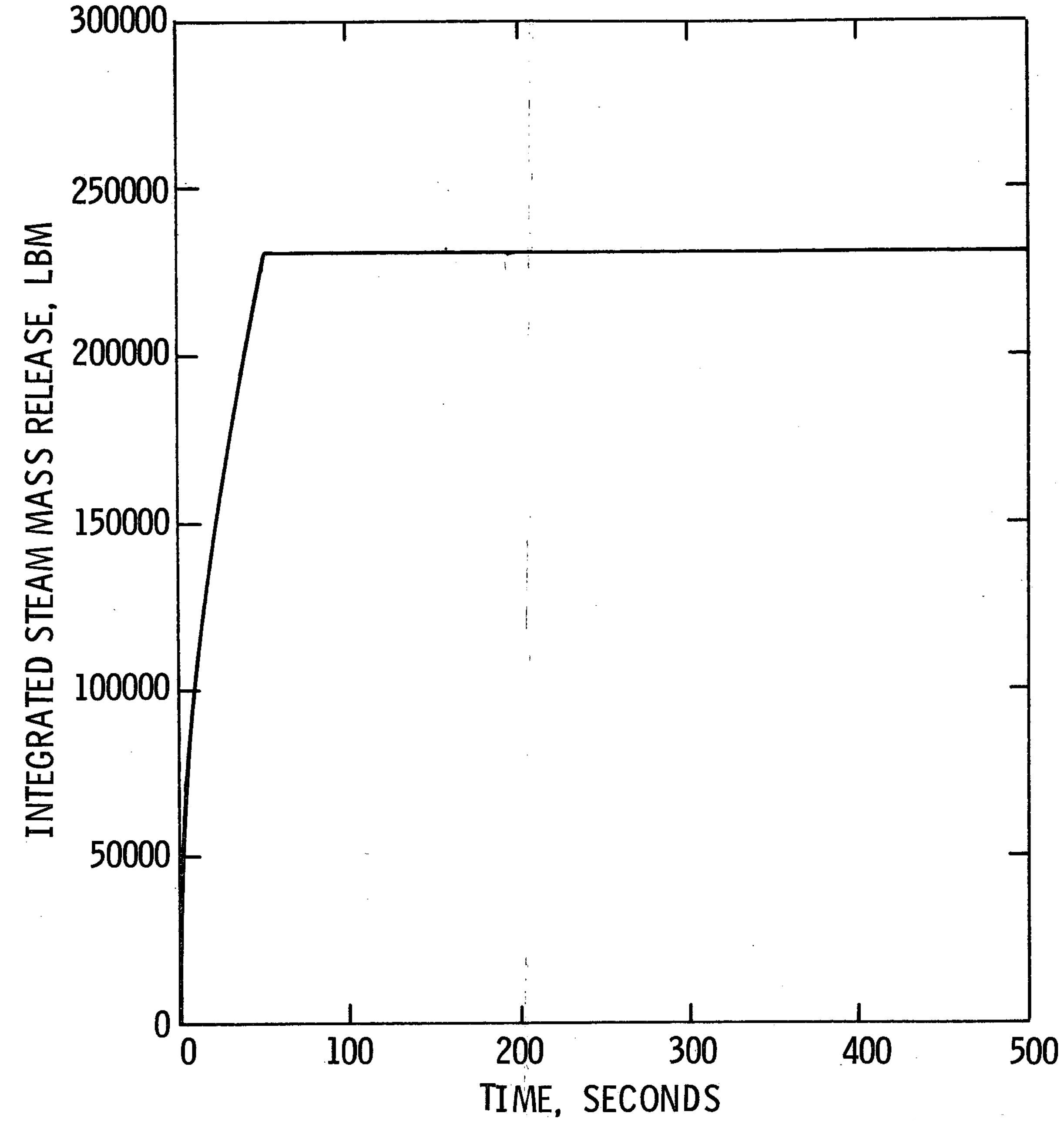


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
REACTIVITY vs. TIME

Figure 15.1-57

10079-11D-977
20OC6

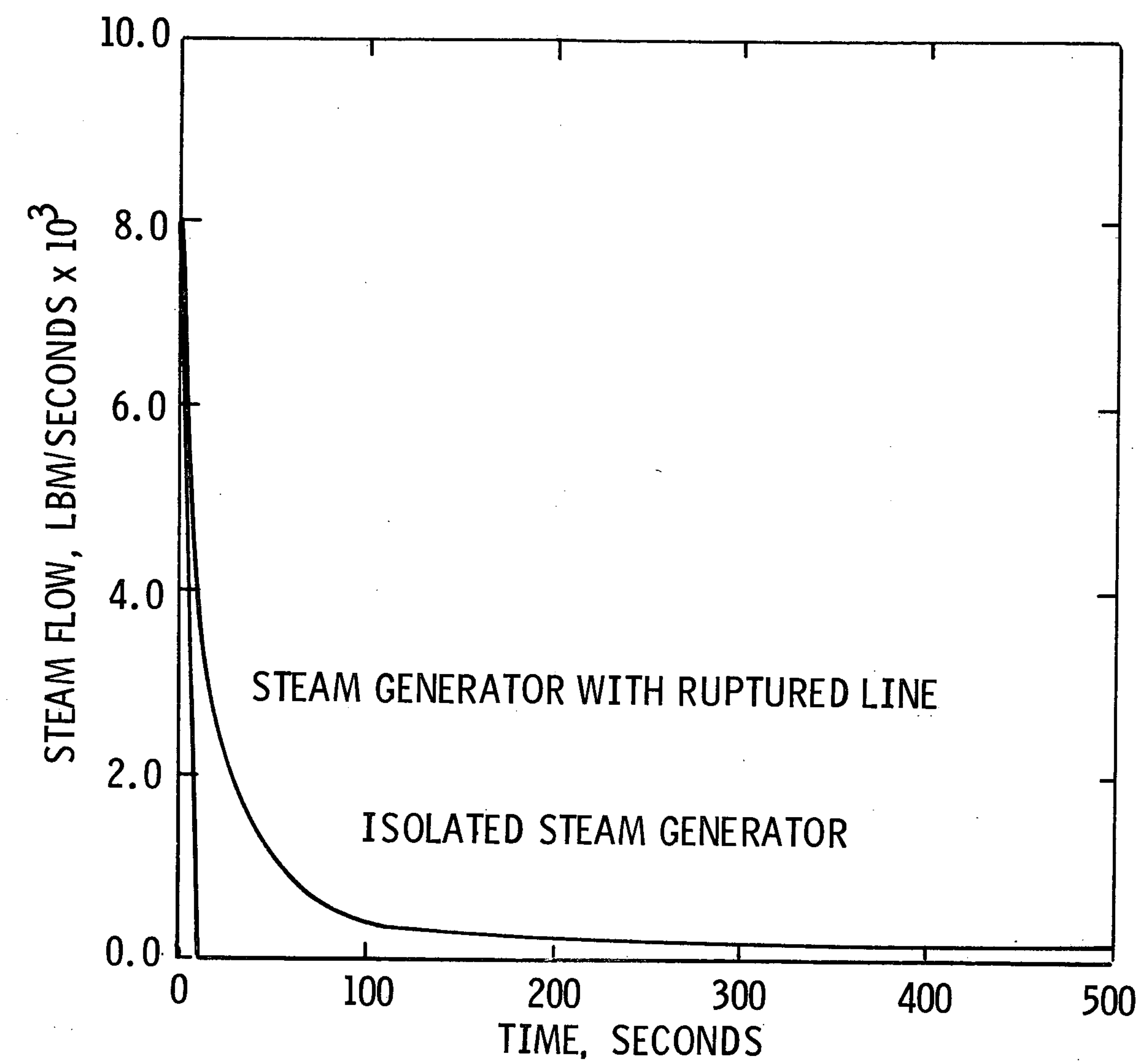


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FULL POWER STEAM LINE BREAK
INTEGRATED STEAM MASS RELEASE
FROM BREAK vs. TIME

Figure 15.1-58

10079-11D-978
20OC6

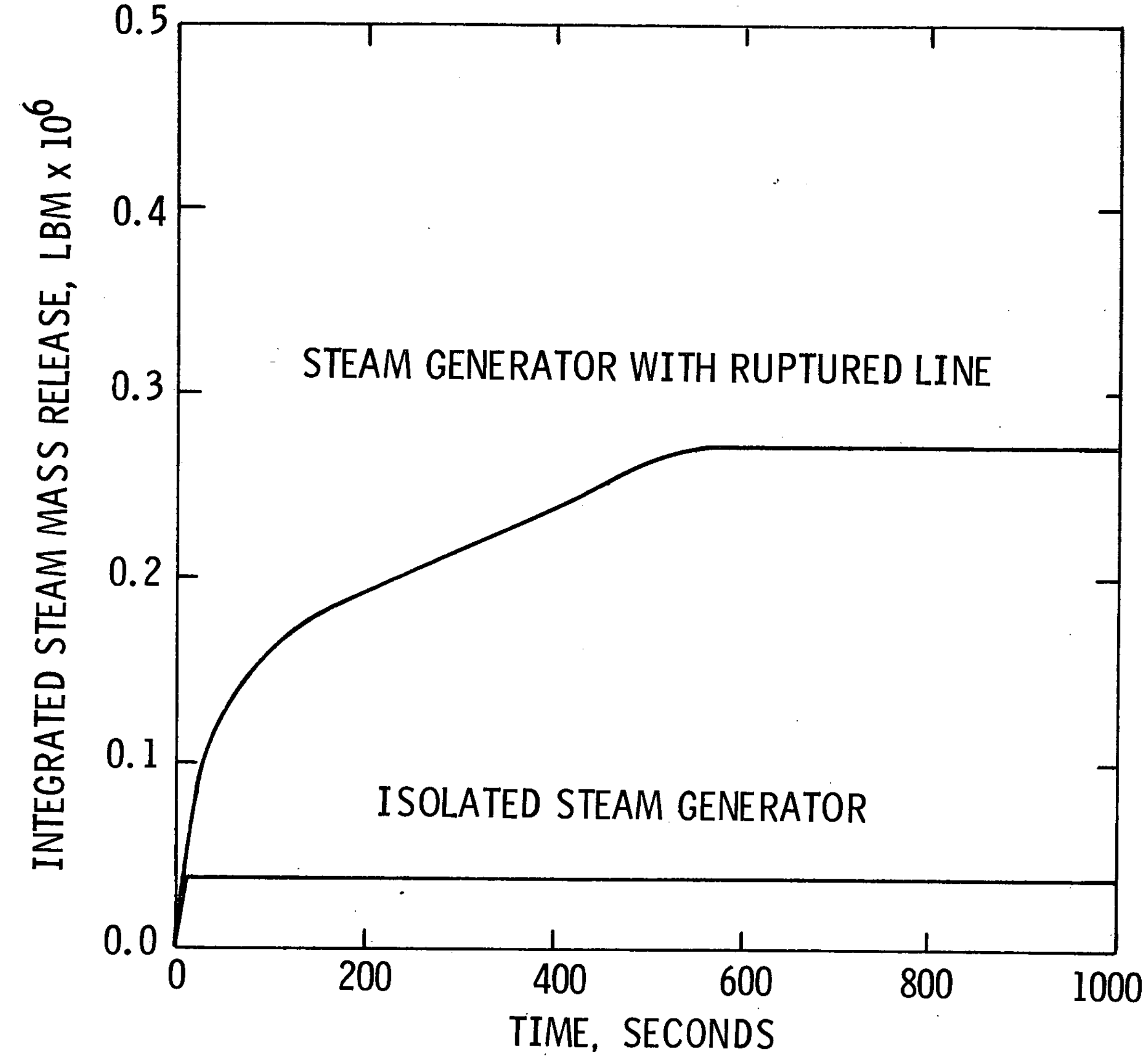


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

HOT ZERO POWER STEAM LINE BREAK WITH LOSS OF AC POWER STEAM MASS RELEASE FROM BREAK vs. TIME

Figure 15.1-59

10079-11D-979
200C6

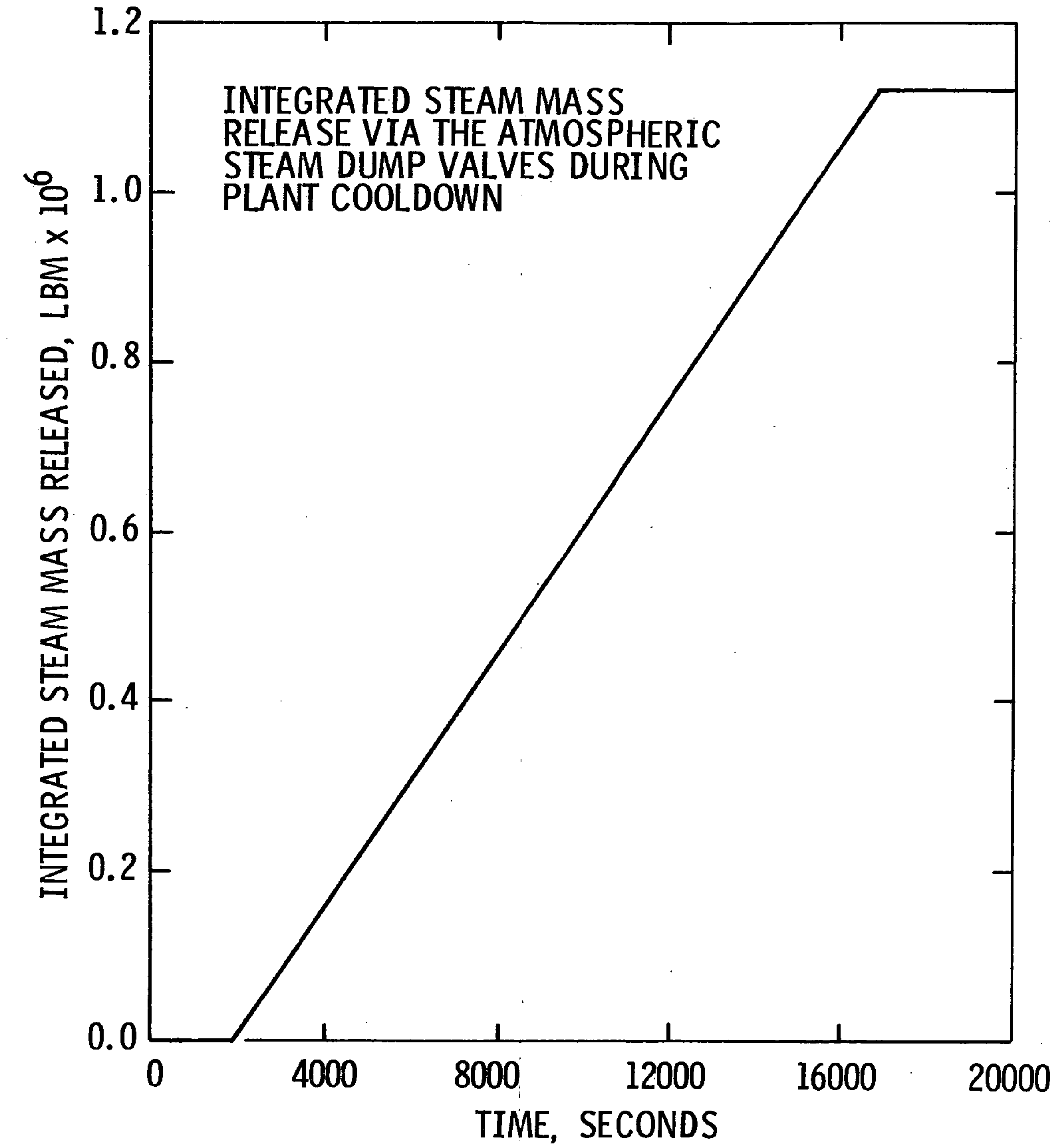


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

HOT ZERO POWER STEAM LINE BREAK WITH LOSS OF AC POWER - INTEGRATED STEAM MASS RELEASE FROM BREAK vs. TIME

Figure 15.1-60

10079-11D-980
20OC6

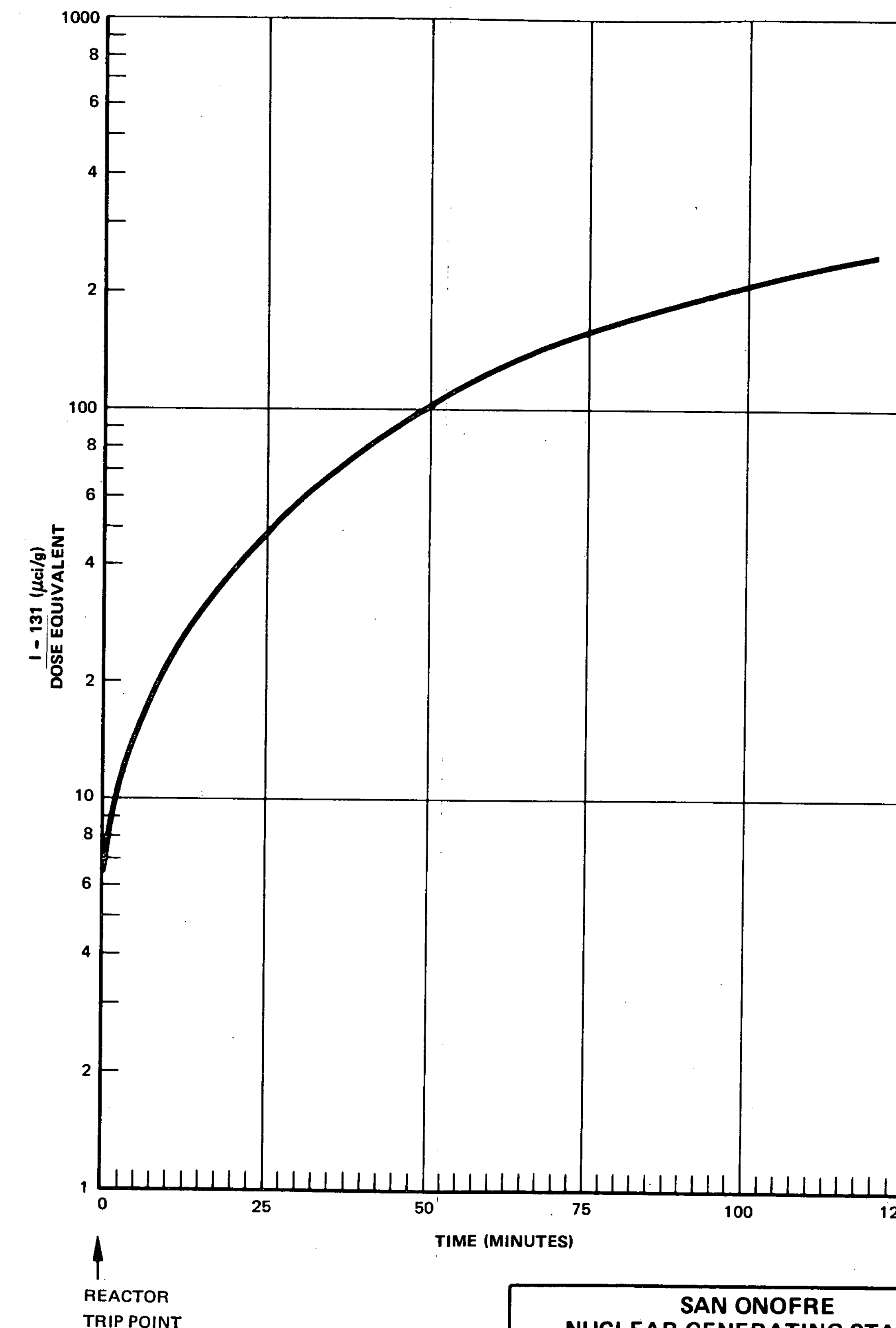


SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3

HOT ZERO POWER STEAM LINE BREAK WITH LOSS OF AC POWER - INTEGRATED STEAM MASS RELEASE FROM STEAM DUMP VALVES vs. TIME

Figure 15.1-61

10079-11D-981
20OC6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

REACTOR COOLANT SYSTEMS DOSE EQUIVALENT IODINE CONCENTRATION vs. TIME FOLLOWING REACTOR TRIP (SPIKING FACTOR = 500)

Figure 15.1-62

10079-11D-982
200C6

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM (TURBINE PLANT)

15.2.1 MODERATE FREQUENCY INCIDENTS

15.2.1.1 Loss of External Load

15.2.1.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of external load classifies it as a moderate frequency incident as defined in reference 1 of section 15.0. A loss of external load is caused by abnormal events in the electrical distribution network.

15.2.1.1.2 Sequence of Events and Systems Operation

A loss of external load produces a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves. A loss of external load would generate a turbine trip which normally produces an immediate reactor trip signal from the turbine master trip relay. The steam bypass control system is normally in automatic mode and would be available upon turbine trip. In the event that the turbine stop valves were to close and the steam bypass control system were in the manual mode, and credit is not taken for reactor trip on turbine trip, reactor trip would occur as a result of high pressurizer pressure. If the bypass system is in the manual mode and no credit is taken for immediate operator action, the steam generator safety valves open to relieve steam and provide an ultimate heat sink for the NSSS. Following a loss of external load, offsite power is available to provide ac power to the auxiliaries. The case of loss of all normal ac power is presented in paragraph 15.2.1.4. The operator can initiate a controlled system cooldown using the turbine bypass valves any time after reactor trip occurs.

The systems operations described above and the resulting sequence of events would produce consequences no more adverse than those following a loss of condenser vacuum, which is described in paragraph 15.2.1.3, since the condenser is available to cool the plant for the loss of external load transient when operator action is assumed after 30 minutes. The consequences of a single malfunction of an active component or system following a loss of external load are discussed in paragraph 15.2.2.1.

15.2.1.1.3 Core and System Performance

The core and system performance parameters following a loss of external load would be no more adverse than those following a loss of condenser vacuum, which is described in paragraph 15.2.1.3.

15.2.1.1.4 Barrier Performance

The barrier performance parameters following a loss of external load would be less adverse than those following a loss of condenser vacuum (see paragraph 15.2.1.3), because the steam bypass control system would be available to remove steam to the condenser rather than using the atmospheric dump valves.

15.2.1.1.5 Radiological Consequences

The radiological consequences due to steam releases from the secondary system and less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in paragraph 15.1.1.4.

15.2.1.2 <u>Turbine Trip</u>

15.2.1.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a turbine trip classifies it as a moderate frequency incident as defined in reference 1 of section 15.0. A turbine trip can be produced by any of the following signals:

A. Manual emergency trip

B. Low pressure of the turbine lube oil

C. Low vacuum in the condenser (see paragraph 15.2.1.3)

D. High temperature of the stator water

E. Low flow of the stator rectifier cooling water

F. Low differential pressure of the seal oil

G. High temperature of the hydrogen

H. High temperature of the low pressure turbine exhaust

I. Electric governor discrepancy trip

J. Protective trips from the reactor

K. Excessive thrust bearing wear

L. Turbine overspeed trip

M. Moisture separator/reheater drain tank level high

N. Generator differential protection

O. Negative phase sequence

P. Unit differential protection

Q. First zone distance

R. Anti-motoring

S. Volts per cycle high

T. Generator stator earth fault (reverse fault)

U. Main transformer Buchholtz surge

V. Loss of excitation

W. Unit transformer differential protection

X. Unit transformer overcurrent

Y. Unit transformer earth fault

15.2.1.2.2 Sequence of Events and Systems Operation

A turbine trip produces a reduction of steam flow from the steam generators to the turbine due to closure of the turbine stop valves. A turbine trip normally produces an immediate reactor trip signal from the turbine stop valves (through unitized activator pressure monitors). The steam bypass control system is normally in automatic mode and would be available upon turbine trip. In the event that the turbine stop valves were to close and the steam bypass control system were in the manual mode, and credit is not taken for reactor trip or turbine trip, reactor trip would occur as a result of high pressurizer pressure. If the bypass system is in the manual mode and no credit is taken for immediate operator action, the steam generator safety valves will open to relieve steam and provide an ultimate heat sink for the NSSS. Following a turbine trip, offsite power is available to provide ac power to the auxiliaries. The case of loss of all normal ac power is presented in paragraph 15.2.1.4. The operator can initiate a controlled system cooldown using the turbine bypass valves any time after reactor trip occurs.

The systems operations described above, and the resulting sequence of events, would produce consequences no more adverse than those following a loss of condenser vacuum, as described in paragraph 15.2.1.3, since the condenser is available to cool the plant for the turbine trip transient when operator action is assumed after 30 minutes. The consequences of a single malfunction of an active component or system following a turbine trip are discussed in paragraph 15.2.2.2.

15.2.1.2.3 Core and System Performance

The core and system performance parameters following a turbine trip would be no more adverse than those following a loss of condenser vacuum, as described in paragraph 15.2.1.3.

15.2.1.2.4 Barrier Performance

The barrier performance parameters following a turbine trip would be less adverse than those following a loss of condenser vacuum (see paragraph 15.2.1.3), because the steam bypass control system would be available to remove steam to the condenser rather than using the atmospheric dump valves.

15.2.1.2.5 Radiological Consequences

The radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in paragraph 15.1.1.4.

15.2.1.3 Loss of Condenser Vacuum

15.2.1.3.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of condenser vacuum classifies it as a moderate frequency incident, as defined in reference 1 of section 15.0 A loss of condenser vacuum may occur due to failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove noncondensible gases, or excessive leakage of air through a turbine gland packing.

15.2.1.3.2 Sequence of Events and Systems Operation

The turbine generator trip that occurs due to a loss of condenser vacuum would normally generate an immediate reactor trip signal from the turbine stop valves (through unitized actuator pressure monitors). If credit is not taken for reactor trip on turbine trip, reactor trip would occur as a result of high-pressurizer pressure. The turbine bypass valves are unavailable following a loss of condenser vacuum due to the actuation of the condenser vacuum interlock on the turbine generator trip. The pressure increases in the primary and secondary systems following reactor trip are limited by the pressurizer and steam generator safety valves. The loss of condenser vacuum causes a turbine trip. Following turbine trip, offsite power is available to provide ac power to the auxiliaries. The case of loss of all normal ac power is presented in paragraph 15.2.1.4. The operator may cool the NSSS using manual operation of the auxiliary feedwater and the atmospheric dump valves any time after reactor trip occurs.

The analysis presented herein conservatively assumes operator action is delayed until 30 minutes after first indication of the event.

The consequences of a single malfunction of an active component or system following a loss of condenser vacuum are discussed in paragraph 15.2.2.3.

Table 15.2-1 gives a sequence of events that occur following a loss of condenser vacuum to the final stabilized condition.

15.2.1.3.3 Core and System Performance

15.2.1.3.3.1 Mathematical Model. The NSSS response to a loss of condenser vacuum was simulated using the CESEC computer program described in section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with the CE-1 CHF correlation described in chapter 4.

15.2.1.3.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used to analyze the NSSS response to a loss of condenser vacuum are discussed in section 15.0. In particular, those parameters that were unique to the analysis discussed below are listed in table 15.2-2. Selection of the automatic mode of operation for the pressurizer control systems has a negligible effect or the limiting parameters and merely influences the timing of the sequence of events.

The initial conditions for the principal process variables monitored by the COLSS were varied within the reactor operating region given in table 15.0-4 to determine the set of conditions that would produce the most adverse consequences following a loss of condenser vacuum. Various combinations of initial core inlet temperature, core inlet flowrate, and pressurizer pressure were considered in order to evaluate their effects on peak pressurizer and steam generator pressures. Decreasing the initial core inlet temperature delays the secondary heat removal, due to the opening of the steam generator safety valves, because of a lower initial secondary pressure. Any further decrease below 545F would have no effect, due to the rapidly decreasing core power after reactor trip. Decreasing the RCS pressure delays the high-pressurizer pressure reactor trip and the opening of the pressurizer safety valves. At an RCS pressure below 2100 $lb/in.^2a$, the steam generator safety valves have opened sufficiently to offset the delay in the energy removing capability of the pressurizer safety valves.

At an RCS pressure below 2100 $lb/in.^2a$, the steam generator safety valves have opened sufficiently to offset the delay in the energy removing capability of the pressurizer safety valves. Increasing the core inlet flowrate produces faster transport through the RCS of the primary energy increase, due to the loss of heat removal by the secondary. Above 110% of design flow, the high pressurizer pressure trip signal is generated soon enough to negate the effect of faster heat transport.

Table 15.2-1
SEQUENCE OF EVENTS FOR THE LOSS OF CONDENSER VACUUM

Time (seconds)	Event	Setpoint or Value
0.0	Closure of turbine stop valves on turbine trip due to loss of condenser vacuum	--
9.6	Steam generator safety valves begin opening, lb/in.^{2}a	1,100
10.8	High-pressurizer pressure trip condition, lb/in.^{2}a	2,422
11.9	Trip breakers open	--
12.2	CEAs begin to drop into core	--
12.6	Maximum core power	110.2% of full power
12.6	Pressurizer safety valves begin to open, lb/in.^{2}a	2,525
14.0	Maximum steam generator pressure, lb/in.^{2}a	1,137
14.3	Maximum RCS pressure, lb/in.^{2}a	2,582
17.0	Maximum pressurizer liquid volume, ft^3	891
17.0	Pressurizer safety valves closed, lb/in.^{2}a	2,525
330.0	Steam generator safety valves close, lb/in.^{2}a	1,056
1800.0	Operator opens atmospheric steam dump valves to begin plant cooldown to shutdown cooling	--
11600.0	Shutdown cooling initiated	--

Table 15.2-2
ASSUMPTIONS FOR THE LOSS OF CONDENSER VACUUM ANALYSIS

Parameter	Assumption
Initial core power level, MWt	3478
Core inlet coolant temperature, ^{o}F	545
Core mass flowrate, 10^6 lbm/h	161.9
Reactor coolant system pressure, $lb/in.^2a$	2100
One pin radial peaking factor, with uncertainty	1.70
Initial core minimum DNBR	2.06
Steam generator pressure, $lb.in.^2a$	840
Moderator temperature coefficient, 10^{-4} $\Delta\rho$/F	+0.5
Doppler coefficient multiplier	0.85
CEA worth for trip, 10^{-2} percent $\Delta\rho$	-7.95
Steam bypass control system	Inoperative
Reactor trip on turbine trip	Inoperative
Pressurizer level control system	Automatic
Pressurizer pressure control system	Automatic

15.2.1.3.3.3 Results. The dynamic behavior of important NSSS parameters following a loss of condenser vacuum are presented in figures 15.2-1 through 15.2-11.

The loss of steam flow due to closure of the turbine stop valves produces a rapid increase in the secondary pressure. This produces a rapid decrease in the primary-to-secondary heat transfer, which causes a rapid heatup of the primary coolant. The insurge to the pressurizer increases the pressurizer pressure producing a high pressurizer pressure alarm signal at 9.6 seconds and a high-pressurizer pressure reactor trip condition at 10.8 seconds. The CEAs begin dropping into the core at 12.2 seconds, which terminates the core power increase at 110.2% of full power.

The opening of the steam generator safety valves at 9.6 seconds and the pressurizer safety valves at 12.6 seconds combine with the decreasing core power due to reactor trip to rapidly reduce the primary and secondary pressures after reaching a maximum pressurizer pressure of 2582 $lb/in.^2a$. The pressurizer safety valves close at 17.0 seconds. The steam generator safety valves close at 330 seconds.

The steam generator safety valves continue to relieve steam to the atmosphere until the atmospheric steam dump valves are opened by operator action at 30 minutes. The plant is then cooled to 350F, at which time shutdown cooling is initiated.

The maximum RCS and secondary pressure do not exceed 110% of design pressure following a loss of condenser vacuum, thus assuring the integrity of the RCS and main steam system is maintained. The minimum DNBR of 1.95 indicates no violation of the fuel thermal limits.

15.2.1.3.4 Barrier Performance

15.2.1.3.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.2.1.3.3.

15.2.1.3.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.2.1.3.3.

15.2.1.3.4.3 Results. Figures 15.2-12 and 15.2-13 give the pressurizer and steam generator safety valves flowrates versus time for the loss of condenser vacuum transient. The steam discharged from the pressurizer is completely condensed in the quench tank and hence not released to the atmosphere. At 30 minutes, when the atmospheric steam dump valves are opened, the steam generator safety valves will have discharged no more than 88,100 pounds of steam. Approximately 506,000 pounds of steam would be discharged through the atmospheric steam dump valves during the 3-hour cooldown, giving a total steam release to the atmosphere of 594,100 pounds.

15.2.1.3.5 Radiological Consequences

The radiological consequences due to steam releases from the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in paragraph 15.1.1.4.

15.2.1.4 Loss of Normal AC Power

15.2.1.4.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of normal ac power classifies it as a moderate frequency incident, as defined in reference 1 of section 15.0. The loss of normal ac power is assumed to result in the loss of all power to the station auxiliaries and a concurrent turbine generator trip. This situation could result either from a complete loss of external grid (offsite) or a loss of the onsite ac distribution system. As a result, electrical power would be unavailable for the reactor coolant pumps, main feedwater pumps, main circulating water pumps, and pressurizer pressure and level control systems. Under such circumstances, the plant would experience a simultaneous loss of load, feedwater flow, and forced reactor coolant flow.

15.2.1.4.2 Sequence of Events and System Operation

At time zero, when all normal ac power is assumed to be lost to the plant, the turbine stop valves close, and it is assumed that the area of the turbine control valves is instantaneously reduced to zero. Also, the steam generator feedwater flow to both steam generators is instanteously assumed to go to zero. The reactor coolant pumps coast down and the reactor coolant flow begins to decrease. A turbine generator trip which occurs would normally generate an immediate reactor trip signal from the turbine master trip relay. Since credit is not taken for a reactor trip due to a turbine trip, a reactor trip will occur as a result of a low DNBR condition as soon as the flow coastdown begins. The low DNBR trip will ensure that the minimum DNBR will not be less than 1.2. In addition, the pressure increases in the RCS and steam generator, following the reactor trip, are limited by the pressurizer and steam generator safety valves.

The loss of all normal ac power is followed by automatic startup of the standby diesel generators, the power output of which is sufficient to supply electrical power to all necessary engineered safety features systems and to provide the capability of maintaining the plant in a safe shutdown condition. Subsequent to the reactor trip, stored and fission product decay energy must be dissipated by the reactor coolant system and main steam system. In the absence of forced reactor coolant flow, convective heat transfer into and out of the reactor core is supported by natural circulation reactor coolant flow. Initially, the residual water inventory in the steam generators is used as a heat sink, and the resultant steam is released to atmosphere by the spring-loaded steam generator safety valves. With the availability of standby diesel power, emergency feedwater is automatically initiated on a low steam generator water level signal. Additional equipment required to operate to maintain safe shutdown conditions is provided in table 8.3-1. Plant cooldown is operator controlled by the atmospheric steam dump valves if normal ac power cannot be restored within 30 minutes (based on emergency procedures). The analysis presented herein conservatively assumes operator action is delayed until 30 minutes after first indication of the event.

The consequences of a single malfunction of a component or system following a loss of normal ac power are discussed in paragraph 15.2.2.4.

Table 15.2-3 gives a sequence of events that occur following a loss of normal ac power to the final stabilized condition.

15.2.1.4.3 Core and System Performance

15.2.1.4.3.1 Mathematical Model. The NSSS response to a loss of normal ac ac power was simulated using the CESEC computer program described in section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with the CE-1 CHF correlation described in chapter 4. During the first 10.0 seconds, the reactor coolant pump coastdown is calculated by the digital computer code COAST described in section 15.0. After this time, the reactor coolant flowrate is extrapolated to an estimated natural circulation flow of 5.0% of nominal full power flow.

15.2.1.4.3.2 Input Parameters and Initial Conditions. In general, the input parameters and initial conditions used to analyze the NSSS response to a loss of normal ac power are discussed in section 15.0. In particular, those parameters that were unique to the analysis discussed below are listed in table 15.2-4. These parameters were chosen the same as the initial conditions for loss of forced reactor coolant flow, as discussed in paragraph 15.3.1.1.3.

15.2.1.4.3.3 Results. The dynamic behavior of important parameters following a loss of all normal ac power is presented in figures 15.2-14 through 15.2-24. The DNBR versus time is bounded by that presented in subsection 15.3.1 and is not presented.

The loss of all normal ac power from an operating limit results in an immediate DNBR trip condition. The CEAs begin to drop at 1.05 seconds. The negative reactivity provided by the CEAs rapidly reduces the reactor core power. The steam generator pressure increases rapidly due to the closure of the turbine control valve and the nonavailability of the steam bypass control system. The steam generator safety valves open at

Table 15.2-3
SEQUENCE OF EVENTS FOR THE LOSS OF NONEMERGENCY AC POWER

Time (seconds)	Event	Setpoint or Value
0.0	Loss of all normal ac power	--
0.6	Low DNBR trip condition	1.19 projected
0.75	Trip breakers open	--
1.05	CEAs begin to drop into core	--
1.45	Minimum DNBR occurs	1.19
4.0	Steam generator safety valves open, $lb/in.^2a$	1,100
4.2	Maximum RCS pressure, $lb/in.^2a$	2,441
8.6	Maximum steam generator pressure, $lb/in.^2a$	1,150
36.4	Low steam generator water level signal	27.0 ft above the tube sheet
89.4	Emergency feedwater reaches the steam generators	--
118.0	Steam generator safety valves close, $lb/in.^2a$	1,056
300.0	Steam generator safety valves open, $lb/in.^2a$	1,100
1800.0	Operator activates the remotely operated atmospheric dump valves	--
12180.0	Shutdown cooling initiated	--

Table 15.2-4
ASSUMPTIONS FOR THE LOSS OF ALL NONEMERGENCY AC POWER

Parameter	Assumption
Initial core power level, MWt	3,478
Core inlet coolant temperature, ^{o}F	560
Core mass flowrate, 10^6 lbm/h	141.9
Reactor coolant system pressure, lb/in.^{2}a	2,400
Steam generator pressure, lb/in.^{2}a	949
One pin radial peaking factor with uncertainty	1.67
Maximum axial peaking factor	1.94
Initial core minimum DNBR	1.31
Moderator temperature coefficient (10^{-4} $\Delta\rho/^{o}F$)	+0.5
Doppler coefficient multiplier	0.85
CEA worth on trip (10^{-2} $\Delta\rho$)	-7.95
Reactor regulating system	Manual
Steam bypass control system	Inoperative
Feedwater regulating system	Manual
Pressurizer level control system	Inoperative

4.0 seconds with the pressure reaching a maximum of 1150 $lb/in.^2a$ at 8.6 seconds after initiation of the event. The RCS pressure increases to 2441 $lb/in.^2a$ at 4.2 seconds, due to the decreased heat removal of the steam generators. Afterwards, the reduced reactor power following the reactor trip causes the RCS pressure and temperatures to decrease. Due to the loss of feedwater flow at the initiation of the event, the steam generator water level decreases. At 36.4 seconds, a low steam generator water level signal is generated. At 50 seconds, the reactor outlet temperature increases in a manner consistent with the core heat flux decay and coolant flow coastdown characteristics. Due to the increase in core average temperature, the RCS pressure also begins to increase. The emergency feedwater, which reaches the steam generators at 89.4 seconds, will lower the steam generator pressure and will provide a heat sink for the decay heat from the RCS. The steam generator safety valves will close at 118 seconds because of the lowered pressure. The steam generator continues to act as a heat sink for the decay heat of the RCS, until at 300 seconds, the steam generator safety valves again open. The emergency feedwater flow plus the steam generator safety valves continue to remove decay heat until standby ac power is again available or until operator action is taken. There is sufficient emergency feedwater available to give adequate time to cooldown the plant and initiate shutdown cooling. It is conservatively assumed that normal ac power is not available, and that at 30 minutes, the atmospheric steam dump valves are opened by the operator to cool down the plant. The primary system is then cooled at a maximum rate of 75F/h to 350F, at which point, shutdown cooling is initiated.

Therefore, for the loss of all normal ac power, the low DNBR trip assures that the DNBR will not decrease below 1.19.

15.2.1.4.4 Barrier Performance

15.2.1.4.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.2.1.4.3.

15.2.1.4.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.2.1.4.3.

15.2.1.4.4.3 Results. Figure 15.2-25 gives the steam generator safety valve flowrate versus time for the loss of all normal ac power. At 30 minutes when the atmospheric steam dump valves are conservatively assumed to be opened, the secondary safety valves will have discharged no more than 77,000 pounds of steam. Approximately 861,000 pounds of steam will be released through the atmospheric steam dump valves during the 2 hours and 53 minutes cooldown to 350F. Therefore, the total steam released to the atmosphere prior to initiation of shutdown cooling is 938,000 pounds.

15.2.1.4.5 Radiological Consequences

The radiological consequences due to steam releases for the secondary system are less severe than the consequences of the inadvertent opening of the atmospheric dump valve discussed in paragraph 15.1.1.4.

15.2.2 INFREQUENT INCIDENTS

15.2.2.1 Loss of External Load with a Concurrent Single Failure of an Active Component

15.2.2.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of external load with a concurrent single failure of an active component classifies it as an infrequent incident as defined in reference 1 of section 15.0. A loss of external load is caused by abnormal events in the electrical distribution network.

15.2.2.1.2 Sequence of Events and Systems Operation

The systems operations following a loss of external load with a concurrent single failure of an active component are the same as those described in paragraph 15.2.1.1.2. The single malfunction of a component or system is discussed in paragraph 15.2.2.3.1 for the loss of condenser vacuum with a concurrent single failure of an active component. The resultant sequence of events would produce consequences no more adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component, which is described in paragraph 15.2.2.3.

15.2.2.1.3 Core and System Performance

The core and system performance parameters, following a loss of external load with a concurrent single failure of an active component, would be no more adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component which is described in paragraph 15.2.2.3.

15.2.2.1.4 Barrier Performance

The barrier performance parameters following a loss of external load with a concurrent single failure of an active component would be less adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component (see paragraph 15.2.2.3), because the steam bypass control system would be available to remove steam to the condenser rather than using the atmospheric dump valves.

15.2.2.1.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the inadvertent opening of an atmospheric dump valve discussed in paragraph 15.1.2.4.

15.2.2.2 Turbine Trip with A Concurrent Single Failure of an Active Component

15.2.2.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a turbine trip with a concurrent single failure of an active component classifies it as an infrequent incident defined in reference 1 of section 15.0. The conditions that can produce a turbine trip are listed in paragraph 15.2.1.2.1.

15.2.2.2.2 Sequence of Events and Systems Operation

The systems operations following a turbine trip with a concurrent single failure of an active component are the same as those described in paragraph 15.2.1.2.2. The single malfunction of a component or system is discussed in paragraph 15.2.2.3.1 for the loss of condenser vacuum with a concurrent single failure of an active component. The resultant sequence of events would produce consequences no more adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component, as described in paragraph 15.2.2.3.

15.2.2.2.3 Core and System Performance

The core and system performance parameters following a turbine trip with a concurrent single failure of an active component would be no more adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component as described in paragraph 15.2.2.3.

15.2.2.2.4 Barrier Performance

The barrier performance parameters following a turbine trip with a concurrent single failure of an active component would be less adverse than those following a loss of condenser vacuum with a concurrent single failure of an active component (see paragraph 15.2.2.3), because the steam bypass control system would be available to remove steam to the condenser rather than using the atmospheric dump valves.

15.2.2.2.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the inadvertent opening of an atmospheric dump valve discussed in paragraph 15.1.2.4.

15.2.2.3 Loss of Condenser Vacuum with Failure of a Primary Safety Valve to Open

15.2.2.3.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of condenser vacuum with a concurrent single failure of an active component classifies this incident as an infrequent incident as defined in reference 1 of section 15.0. The cause of the loss of condenser vacuum is discussed in paragraph 15.2.1.3.1. Various active component single failures were considered to determine which failure had the most adverse effect following a loss of condenser vacuum. The single failures considered were (1) a loss of all ac power on reactor trip, (2) failure of one primary safety valve to open, and (3) failure of one steam generator safety valve to open. The failure of one primary safety valve to open produces the most adverse effect following a loss of condenser vacuum. For such a failure, it must be postulated that a malfunction occurs in the spring mechanism that operates the valve or in the valve itself.

15.2.2.3.2 Sequence of Events and System Operation

The systems and reactor trip which operate following a loss of condenser vacuum with failure of one primary safety valve to open are the same as those described in paragraph 15.2.1.3.2 following a loss of condenser vacuum.

Table 15.2-5 gives a sequence of events that occur following a loss of condenser vacuum with concurrent failure of one primary safety valve to open.

15.2.2.3.3 Core and System Performance

15.2.2.3.3.1 Mathematical Model. The mathematical model used for evaluation of core and system performance is identical to that described in paragraph 15.2.1.3.3.

15.2.2.3.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of core and systems performance are identical to those described in paragraph 15.2.1.3.3.

Table 15.2-5
SEQUENCE OF EVENTS FOR THE LOSS OF CONDENSER VACUUM
WITH FAILURE OF A PRIMARY SAFETY VALVE

Time (seconds)	Event	Setpoint or Value
0.0	Closure of turbine stop valves on turbine trip due to loss of condenser vacuum	--
9.75	Steam generator safety valves begin opening, $lb/in.^2a$	1,100
10.8	High pressurizer pressure trip condition, $lb/in.^2a$	2,422
11.9	Trip breakers open	--
12.2	CEAs begin to drop into core	--
12.6	Maximum core power	110.2% of full power
12.6	Available pressurizer safety valve begins to open, $lb/in.^2a$	2,525
13.9	Maximum steam generator pressure, $lb/in.^2a$	1,138
14.6	Maximum RCS pressure, $lb/in.^2a$	2,612
17.0	Maximum pressurizer liquid volume, ft^3	888
18.0	Pressurizer safety valve closed, $lb/in.^2a$	2,525
330.0	Steam generator safety valves close, $lb/in.^2a$	1,056
1800.0	Operator opens atmospheric dump valves to begin plant cooldown	--
11600.0	Shutdown cooling initiated	--

15.2.2.3.3.3 Results. The dynamic behavior of the NSSS following a loss of condenser vacuum with concurrent failure of one primary safety valve to open is similar to that following a loss of condenser vacuum which is described in paragraph 15.2.1.3.3. Therefore, only the pressurizer pressure transient is presented here in figure 15.2-26. The maximum core power reached, following a loss of condenser vacuum with concurrent failure of a primary safety valve to open, is 110.3% of full power. The peak RCS and main steam system pressures were 2612 lb/in.^{2}a and 1138 lb/in.^{2}a, respectively. These pressures are within 110% of design assuring the integrity of the RCS and MSS is maintained following a loss of condenser vacuum with concurrent failure of a primary safety valve to open. The minimum DNBR of 1.95 indicates no violation of the fuel thermal limits.

15.2.2.3.4 Barrier Performance

15.2.2.3.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.2.1.3.3.

15.2.2.3.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.2.1.3.3.

15.2.2.3.4.3 Results. Figures 15.2-27 and 15.2-28 give the pressurizer and steam generator safety valve flowrate versus time following a loss of condenser vacuum with concurrent failure of a pressurizer safety valve to open. Until operator action is taken at 30 minutes, the total steam release to atmosphere discharged through the steam generator safety valves has been no more than 88,300 pounds. The operator would then begin a controlled NSSS cooldown at 75F/h by opening the atmospheric steam dump valves to discharge steam at a rate of 52 lbm/s. After 2.71 hours, the primary system will have reached an average temperature of 350F at which point the shutdown cooling system may be placed in operation. The total steam release to atmosphere during the course of this transient is 595,600 pounds.

15.2.2.3.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the inadvertent opening of an atmospheric dump valve discussed in paragraph 15.1.2.4.

15.2.2.4 Loss of all Normal AC Power with a Concurrent Single Failure of an Active Component

Any credible single failure of an active component concurrent with a loss of all normal ac power produces consequences less severe than those following a single reactor coolant pump shaft seizure, which is described in section 15.3.3.1.

15.2.2.5 Loss of Normal Feedwater Flow

15.2.2.5.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of normal feedwater flow classifies it as an infrequent incident as defined in ANSI N18.2.[1]

A loss of normal feedwater flow is defined as a reduction in feedwater flow to the steam generators when operating at power without a corresponding reduction in steam flow from the steam generators. The result of this flow mismatch is a reduction in the steam generator water inventory and a subsequent heatup of the primary coolant. The complete loss of normal feedwater case is analyzed since this condition requires the most rapid response from the plant protection system (PPS). Due to several failures, a complete loss of normal feedwater flow can result from the loss of both main feedwater pumps or the loss of four condensate pumps. In manual feedwater control, closing the feedwater control or isolation valves can also result in a complete loss of normal feedwater flow.

15.2.2.5.2 Sequence of Events and Systems Operation

The complete loss of normal feedwater flow case is analyzed by assuming an instantaneous stoppage of feedwater flow to both steam generators. The PPS provides protection against the loss of the secondary heat sink by the steam generator low water level trip and the automatic initiation of the emergency feedwater system. The emergency feedwater consists of one motor-driven and one turbine-driven emergency feedwater pump. The high-pressurizer pressure trip provides protection in the event the RCS pressure limit is approached. The steam bypass control system is assumed to be in the automatic mode, which maximizes the decrease in steam generator water inventory. Table 15.2-6 presents the sequence of events for the complete loss of normal feedwater from initiation of the event unit termination at a cold shutdown condition.

The consequences of a single malfunction of a component or system following a loss of normal feedwater flow are discussed in paragraph 15.2.3.2.

15.2.2.5.3 Core and System Performance

15.2.2.5.3.1 Mathematical Model. The NSSS response to a loss of normal feedwater flow was simulated using the CESEC computer program described in section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with CE-1 CHF correlation described in chapter 4.

Table 15.2-6
SEQUENCE OF EVENTS FOR THE LOSS OF NORMAL FEEDWATER

Time (seconds)	Event	Setpoint or Value
0.0	Termination of all feedwater flow	--
9.8	Main steam bypass valve opens	950 lb/in.^{2}a header pressure
42.0	Low steam generator water level alarm	28.5 ft above the tube sheet
46.8	Low steam generator water level trip signal	27.0 ft above the tube sheet
47.0	Maximum core power	103.4% of full power
47.2	Reactor trip breakers open	--
47.5	CEAs begin to drop into core	--
49.6	Maximum RCS pressure, lb/in.^{2}a	2,160
51.2	Steam generator safety valves begin to open, lb/in.^{2}a	1,100
54.4	Maximum steam generator pressure, lb/in.^{2}a	1,154
71.4	Steam generator safety valves close, lb/in.^{2}a	1,056
89.5	Emergency feedwater reaches steam generator	--
130.0	Minimum steam generator inventory	12.0 of nominal inventory
160.0	Minimum RCS pressure, lb/in.^{2}a	1,519
1800	Operator opens atmospheric steam dump valves	--
11400	Shutdown cooling initiated	--

15.2.2.5.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used to analyze the NSSS response to a complete loss of normal feedwater are discussed in section 15.0. In particular, those parameters which were unique to the analysis discussed below are listed in table 15.2-7.

The initial conditions for the principal process variables monitored by the COLSS system were varied within the reactor operating space given in table 15.0-4 to determine the set of conditions that would produce the maximum decrease in steam generator water inventory following a complete loss of normal feedwater flow. No set of initial conditions could be found such that for a complete loss of normal feedwater flow the RCS pressure would approach 110% of the design pressure. Various combinations of initial core inlet temperature, initial pressurizer pressure, and initial core flowrate were considered. Increasing the initial core inlet temperature increases the secondary side pressure. The increased initial steam generator pressure causes the turbine steam bypass system to open sooner after the cessation of the feedwater flow. Therefore, an inlet temperature of 560F was used in this analysis. Lowering the initial pressurizer pressure to 2000 lb/in.^{2}a insures that the reactor trip signal will not be generated from a high-pressurizer pressure signal. A reactor trip on low steam generator water level will minimize the steam generator water inventory during this transient. The initial core flowrate has little effect on the transient minimum steam generator water inventory. Above 100% flow, the minimum steam generator water level increases slightly. At 90% flow, there is only a 1% change in the minimum steam generator water inventory. Therefore, 100% of nominal flow was used for this analysis.

Another important parameter varied to minimize the steam generator water inventory during a loss of normal feedwater flow was the initial steam generator water level. This parameter was set at the high-level alarm setting. At this setting, the reactor trip on low steam generator water level is delayed, so that the primary coolant temperatures will be increased to the maximum possible value. Increasing the primary coolant temperatures will increase the secondary pressure and minimize the steam generator water inventory.

Finally a large bottom peaked axial shape was utilized to ensure conservative power reduction as the CEAs are inserted on reactor trip.

15.2.2.5.3.3 Results. The dynamic behavior of important parameters following a loss of normal feedwater are presented in figures 15.2-29 through 15.2-39.

The complete loss of normal feedwater results in an increase in the secondary pressure and temperature. Due to this increase, the RCS temperatures begin to increase. The turbine continues to operate with a subsequent decrease in secondary side steam generator inventory. The RCS pressure increases as the temperature and power increases. The reactor is tripped

Table 15.2-7
ASSUMPTIONS FOR THE LOSS OF NORMAL FEEDWATER ANALYSIS

Parameter	Assumption
Initial core power level, MWt	3478
Core inlet coolant temperature, ^{o}F	560
Core mass flowrate, 10^{6} lbm/h	141.0
Reactor coolant system pressure, $lb/in.^{2}a$	2000
Steam generator pressure, $lb/in.^{2}a$	950
One pin radial peaking factor, with uncertainty	1.70
Maximum axial peaking factor	1.99
Initial core minimum DNBR	1.27
Moderator temperature coefficient (10^{-4} $\Delta\rho/^{o}F$)	+0.5
Doppler coefficient multiplier	0.85
CEA worth on trip (10^{-2} $\Delta\rho$)	-7.95
Main steam bypass control system	Automatic
Feedwater regulating system	Malfunction
Reactor regulating system	Manual

by a low steam generator water level signal 46.8 seconds after initiation of the transient. The CEAs begin to drop at 47.5 seconds. The reactor core power level has increased to a maximum of 103.4% of full power at this time. However, the DNBR has not decreased below the initial value due to the increase in reactor coolant system pressure. The negative reactivity provided by the CEAs rapidly reduces the reactor core power. The steam bypass control system in combination with the steam generator safety valves rapidly cool down the RCS following the reactor trip. The maximum pressures in the RCS and main steam systems are 2160 and 1154 $lb/in.^2a$, respectively. Emergency feedwater reaches the steam generators 42 seconds after actuation of low steam generator water level signal trip. The total steam generator inventory reaches its minimum value (12.0% of the nominal inventory) at 130 seconds. The steam bypass control system operates to remove decay heat until operator action is taken. This analysis conservatively assumes that operator action is delayed until 30 minutes after initiation of the event. The primary system is then cooled to 350F by use of the atmospheric steam dump valves at which point shutdown cooling is initiated.

Therefore, for the complete loss of normal feedwater flow, the DNB ratio is not less than the initial value and the PPS assures that the steam generator heat removal capacity is maintained and that the RCS pressure does not exceed 110% of design.

15.2.2.5.4 Barrier Performance

15.2.2.5.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.2.2.5.3.

15.2.2.5.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.2.2.5.3.

15.2.2.5.4.3 Results. Figure 15.2-39 gives the steam generator safety valve flowrate versus time for the loss of normal feedwater. At 30 minutes, when the atmospheric steam dump valves are opened, the steam generator safety valves will have discharged no more than 29,000 pounds of steam. Approximately 800,000 pounds of steam will be released through the atmospheric steam dump valves during the 2-hour and 40-minute cooldown. The total steam released to the atmosphere prior to initiation of shutdown cooling is 829,000 pounds, which is less than that released during the loss of normal ac power incident.

15.2.2.5.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the inadvertent opening of an atmospheric dump valve discussed in paragraph 15.1.1.4.

15.2.3 LIMITING FAULTS

15.2.3.1 Feedwater System Pipe Breaks

15.2.3.1.1 Identification of Causes and Frequency Classification

The estimated frequency of a feedwater system pipe break classifies it as a limiting fault incident as defined in reference 1 of section 15.0. A feedwater system pipe break may occur due to a pipe failure in the main feedwater system.

15.2.3.1.2 Sequence of Events and Systems Operation

A feedwater system pipe break may produce a total loss of normal feedwater and a very rapid blowdown of one steam generator. If normal plant electrical power is lost, this superimposes a loss of primary coolant flow, turbine load, pressurizer pressure and level control, and steam bypass control. The culmination of these events is a rapid decrease in the heat transfer capability of both steam generators and eventual elimination of one steam generator's heat transfer capability. The result is an RCS heatup and pressurization. The NSSS is protected during this transient by the pressurizer safety valves and the following reactor trips: (1) steam generator low water level, (2) steam generator low pressure, (3) high pressurizer pressure, and (4) low DNBR. Depending on the particular initial conditions, any one of these trips may terminate this transient. The NSSS is also protected by the steam generator safety valves and the auxiliary feedwater system which serve to maintain the integrity of the secondary heat sink following reactor trip. In this analysis however, one of the two auxiliary feedwater pumps is assumed to fail as the most adverse single active failure. The operator can initiate a controlled plant cooldown using the atmospheric steam dump valves any time after reactor trip occurs. The analysis presented herein conservatively assumes operator action is delayed until 30 minutes after the first initiating event. Table 15.2-8 gives the sequence of events that occurs following a feedwater system pipe break to the final stabilized condition.

15.2.3.1.3 Core and System Performance

15.2.3.1.3.1 Mathematical Model. The NSSS response to a feedwater system pipe break was simulated using the CESEC-ATWS computer program described in section 15.0 along with the blowdown model described below. Using the

Table 15.2-8
SEQUENCE OF EVENTS FOR THE FEEDWATER SYSTEM PIPE BREAK (Sheet 1 of 2)

Time (seconds)	Event	Setpoint or Value
0.0	Double-ended rupture of the main feedwater line	--
14.0	Low water level trip condition in the steam generators	5% of instrument range
14.0	Auxiliary feedwater actuation signal generated by low water level condition	--
14.0	Normal onsite and offsite power lost	--
14.0	Low DNBR trip condition	1.19 projected DNBR
14.8	Trip breakers open	--
15.0	High-pressurizer pressure trip condition, $lb/in.^2a$	2,422
15.1	CEAs begin to drop into core	--
15.2	Maximum core power	103.7% of full power
15.8	Pressurizer safety valves open, $lb/in.^2a$	2,525
16.5	Minimum DNBR	1.19
17.2	Maximum RCS pressure, $lb/in.^2a$	2,740
17.2	Maximum pressurizer surge line flow, lbm/s	2,033
17.4	Steam generator safety valves open, $lb/in.^2a$	1,100
17.8	Maximum pressurizer pressure, $lb/in.^2a$	2,633
18.0	Low pressure trip condition in the steam generator connected to the ruptured feed line, $lb/in.^2a$	675

Table 15.2-8
SEQUENCE OF EVENTS FOR THE FEEDWATER SYSTEM PIPE BREAK (Sheet 2 of 2)

Time (seconds)	Event	Setpoint or Value
18.0	Main steam isolation signal generated, $lb/in.^2a$	675
18.8	Steam generator connected to the ruptured feed line empties	--
21.5	Maximum steam generator pressure, $lb/in.^2a$	1155
24.0	Minimum pressurizer steam volume, ft^3	434
24.5	Pressurizer safety valves close, $lb/in.^2a$	2,525
68.2	Auxiliary feedwater flow initiated to the steam generator connected to the intact feed line, lbm/s	97
90.0	Minimum liquid mass in the steam generator connected to the intact feed line, lbm	21,300
1800	Operator opens the atmospheric steam dump valves to begin plant cooldown to shutdown cooling	--
13300	Shutdown cooling initiated	--

core heat flux and core inlet conditions calculated by CESEC-ATWS, the thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in section 15.0 with the CE-1 CHF correlation described in chapter 4.

Blowdown of the steam generator nearest the feedwater line break was modeled assuming frictionless critical flow calculated by the Henry-Fauske correlation (reference 1). The enthalpy of the blowdown is assumed to be that of saturated liquid initially. As the steam generator liquid mass decreases, the quality of the blowdown is allowed to increase according to that quality which is calculated by assuming that all of the liquid mass would be contained in the downcomer region, and that it forms a homogenous two-phase mixture with a two-phase level which remains at the height of the break location (bottom of the feedwater ring). This model conservatively underestimates the blowdown quality and energy and overestimates the discharge rate, thereby leading to a more rapid blowdown and subsequent loss of steam generator heat removal capability.

Assuming the two-phase mixture level remains at the feedwater ring as the quality increases, also provides a very conservative prediction of the minimum steam generator liquid mass existing in the steam generator connected to the ruptured feedwater line at a low water level trip condition. Since this model underestimates the quality in the downcomer, the two-phase density and static head between the level sensors are overestimated. This method will, therefore, determine a higher level for a given liquid mass than can actually occur, conservatively delaying the low level trip condition.

15.2.3.1.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used to analyze the NSSS response to a feedwater pipe break are discussed in section 15.0. In particular, those parameters which were unique to the analysis discussed below are listed in table 15.2-9.

The initial conditions for the principal process variables monitored by the COLSS were varied within the reactor operating space given in table 15.0-4 to determine the set of conditons that would produce the most adverse consequences following a feedwater system pipe break. The full spectrum of break areas was considered up to a break size of the combined area of the flow distributing nozzles in the bottom of the feedwater ring. The time for the loss of normal plant electrical power, the initial intact steam generator inventory and the initial RCS pressure were adjusted within the plant operating space in order to produce as nearly as possible simultaneous trip conditions for: (1) the intact steam generator low water level, (2) the ruptured steam generator low water level, (3) the high pressurizer pressure, and (4) low DNBR. Selection of these conditions maximizes the RCS pressure and the mismatch between core power and steam generator heat removal capacity just prior to the CEAs dropping into the core. Due to the more rapid loss of steam generator heat transfer capability as the break size increases for the steam generator connected to the ruptured

Table 15.2-9
ASSUMPTIONS FOR THE FEEDWATER SYSTEM PIPE BREAK

Parameter	Assumption
Initial core power, MWt	3478
Core inlet coolant temperature, ^{o}F	560
Core mass flowrate, 10^6 lbm/h	132.2
Reactor coolant system pressure, lb/in.^{2}a	2300
One pin radial peaking factor, with uncertainty	1.45
Initial core minimum DNBR	1.24
Steam generator pressure, lb/in.^{2}a	949
Moderator temperature coefficient, 10^{-4} $\Delta\rho$/F	+0.5
Doppler coefficient multiplier	0.85
CEA worth for trip, 10^{-2} $\Delta\rho$	-8.55
Steam bypass control system	Inoperative
Pressurizer pressure control system	Inoperative
Pressurizer level control system	Inoperative
Feedwater line break area, ft^2	1.076
Initial intact steam generatory inventory, lbm	123000
Auxiliary feedwater capacity assuming one failed pump, gal/min	700

feed line, the largest break area becomes the most adverse case. Core inlet temperature and flow had negligible effects on the peak RCS pressure for a given blowdown rate. However, maximizing the core inlet temperature also maximizes the steam generator pressure, which increases the maximum blowdown rate. The maximum inlet temperature of 560F also maximizes the RCS energy content and thereby increases the radiological releases associated with steam generator safety valve and atmospheric steam dump valve flows.

Of those systems and components called upon to mitigate the consequences of a feedwater system pipe break (i.e., pressurizer and steam generator safety valves, feed line check valves, auxiliary feedwater system, and reactor protective system), failure of the pressurizer or steam generator safety valves, or the feed line check valves is not considered credible. With respect to the reactor protective system, the most reactive CEA is conservatively assumed to be stuck in the fully withdrawn position. Therefore, the worst active single failure, in addition to the stuck CEA is the failure of one out of the two auxiliary feedwater pumps. This failure leads to larger radiological releases through the steam generator safety valves due to the relatively higher steam generator pressure which results with only one-half the auxiliary feedwater flow available.

15.2.3.1.3.3 Results. The dynamic behavior of important parameters following a feedwater system pipe break is presented in figures 15.2-40 through 15.2-56.

The double-ended rupture of the main feedwater line is assumed to instantaneously terminate feedwater flow to one steam generator due to closure of the check valve between the steam generator and the break. Critical flow is assumed to be instantaneously established from the other steam generator due to the break location between the steam generator and the check valve. The first 12 seconds are characterized by a very gradual heatup of the primary and secondary systems due to the absence of subcooled feedwater flow to the steam generators. Over the next 3 seconds, the steam generator connected to the ruptured feedline loses its heat transfer capability due to the depleted inventory. This initiates a strong RCS-to-steam generator power mismatch, which is further aggravated when the steam generator connected to the intact feed line is isolated as a result of the loss of normal ac power at 14 seconds (i.e., the turbine stop valves are assumed to close instantaneously). The loss of normal plant electrical power occurs simultaneously with low water level trips from both steam generators. The rapidly increasing RCS coolant temperatures produce a large insurge to the pressurizer, causing its pressure to exceed the high pressure trip setpoint at 15 seconds. By 15.1 seconds the CEAs begin to drop into the core; however, the RCS pressure continues to increase passing the pressurizer safety valve setpoint of 2525 lb/in.^{2}a at 15.8 seconds, until the pressure turns around after reaching a maximum of 2740 lb/in.^{2}a in the RCS at 17.2 seconds and 2633 lb/in.^{2}a in the pressurizer at 17.8 seconds. The core heat flux has decayed sufficiently by this time to reduce the

RCS-to-steam generator power imbalance. By 17.4 seconds, the steam generator safety valves open limiting the steam generator pressure to a maximum of 1155 $lb/in.^2a$. With the steam generator pressure and temperature stabilized, the RCS-to-steam generator heat transfer remains relatively constant as the core heat flux continues to decrease. By 24 seconds, the power imbalance reverses with the steam generator removing more energy than the core produces. The pressurizer safety valves close at 24.5 seconds as the primary coolant temperatures decrease. The auxiliary feedwater flow reaches the intact steam generator by 68.2 seconds and matches the safety valve steam flow by 90.0 seconds, thus preventing further depletion of the steam generator liquid inventory below 21,300 lbm. The RCS pressure again increases from 80 to 300 seconds as the relatively cold coolant, which exits from the core between 40 and 150 seconds, finally reaches the steam generator under the low flow conditions that exist following loss of ac power. The decrease in differential temperature (RCS-to-steam generator) reduces the heat transfer rate. The steam generator safety valves continue to relieve to the atmosphere until the atmospheric dump valves are opened by the operator at 30 minutes. The plant is then cooled to 350F at which time shutdown cooling is initiated.

Although this transient should only be required to meet faulted stress limits, the maximum RCS and steam generator pressures do not exceed 110% of design pressure (i.e., the upset stress limit) following a feedwater system pipe break, thus assuring the integrity of the RCS and the steam generator connected to the intact feed line. The minimum DNBR of 1.19 indicates no violation of the fuel thermal limits.

15.2.3.1.4 Barrier Performance

15.2.3.1.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.2.3.1.3.

15.2.3.1.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.2.3.1.3.

15.2.3.1.4.3 Results. Figures 15.2-52 and 15.2-53 give the pressurizer and steam generator safety valves flowrates versus time for the feedwater system pipe break transient. At 30 minutes when the atmospheric dump valves are opened, the steam generator safety valves will have discharged no more than 140,000 pounds of steam. Approximately 934,000 pounds of steam would be discharged through the atmospheric dump valves during the 3.2 hours of cooldown, giving total steam release to the atmosphere of 1,074,000 pounds. The steam generator connected to the ruptured feedwater line discharges 149,000 pounds of fluid to containment. The pressurizer safety valves release 1585 pounds of steam to the quench tank.

15.2.3.1.5 Radiological Consequences

The radiological consequences of this event are less severe than the consequences of the main steam line break discussed in paragraph 15.1.3.1.

15.2.3.2 Loss of Normal Feedwater Flow with an Active Failure in the Turbine Steam Bypass System

15.2.3.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a loss of normal feedwater flow with a concurrent single failure of an active component classifies this incident as an infrequent incident as defined in reference 1 of section 15.0. The causes of a loss of normal feedwater flow are discussed in paragraph 15.2.2.5.1. Various active component single failure were considered to determine which failure had the most adverse effect following a loss of normal feedwater flow. The single active failures considered were: (1) a loss of all normal ac power on reactor trip, (2) failure of the steam bypass control system open, and (3) loss of 50% of emergency feedwater. The failure of the turbine steam bypass control system open produces the minimum steam generator inventory in the shortest period of time following a loss of normal feedwater flow. This failure could be caused by an electrical signal malfunction. This malfunction results in a quick opening signal to all the turbine bypass valves. It is assumed that the failure in the steam bypass control system (SBCS) results in these valves remaining open, even in the presence of closure signals generated by the SBCS due to adverse steam generator or condenser conditions (e.g., low pressure, and low level) until a main steam isolation signal (MSIS) is generated.

15.2.3.2.2 Sequence of Events and Systems Operation

The systems and reactor trip which operate following a loss of normal feedwater flow with failure of the steam bypass control system open are the same as those described in paragraph 15.2.2.5.2, except for the operation of the bypass system and the generation of an MSIS. The MSIS is generated due to low steam generator pressure and provides protection against emptying the steam generators.

Table 15.2-10 gives a sequence of events that occur following a loss of normal feedwater flow with the turbine steam bypass system open.

15.2.3.2.3 Core and System Performance

15.2.3.2.3.1 Mathematical Model. The mathematical model used for evaluation of core and system performance is identical to that described in paragraph 15.2.2.5.3.

Table 15.2-10
SEQUENCE OF EVENTS FOR THE LOSS OF FEEDWATER FLOW WITH AN ACTIVE FAILURE IN THE TURBINE STEAM BYPASS SYSTEM

Time (seconds)	Event	Setpoint or Value
0.0	Termination of all feedwater flow	--
9.8	Turbine steam bypass valves fully open, $lb/in.^2a$ header pressure	950
42.8	Low steam generator water level trip signal	27.0 ft above the tube sheet
43.2	Reactor trip breakers open	--
43.3	Maximum core power	124% of full power
43.5	CEAs begin to drop into core	--
75.5	Low pressurizer pressure safety injection actuation signal, $lb/in.^2a$	1,560
80.7	Safety injection flow commences, $lb/in.^2a$	1,485
83.9	Pressurizer empties, $lb/in.^2a$	1,443
85.5	Emergency feedwater reaches steam generators	--
95.9	Reactor coolant pumps cavitate, flow coastdown commences	--
97.8	Main steam isolation signal, $lb/in.^2a$	675
100.7	Minimum steam generator pressure, $lb/in.^2a$	643
101.2	Main steam isolation valves fully closed	--
101.2	Minimum steam generator water inventory	5.6% of nominal inventory
1800	Operator begins cooldown	--
10200	Shutdown cooling initiated	--

15.2.3.2.3.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of core and systems performance are identical to those described in paragraph 15.2.2.5.3 except that a moderator coefficient of -3.3×10^{-4} $\Delta\rho/^{o}F$ was utilized. The negative moderator coefficient insures a large power increase during the cooldown caused by the turbine bypass valves failing open. The radial peak and axial shape for this case were chosen such that a DNBR trip condition would not occur before the low steam generator water level trip signal. This procedure was utilized in order to allow the heat flux to increase to the maximum possible value before trip. This procedure insures a transient which will result in the minimum steam generator inventory.

15.2.3.2.3.3 Results. The dynamic behavior of important parameters following a loss of normal feedwater flow with failure of the turbine steam bypass system open are presented in figures 15.2-57 through 15.2-67.

The complete loss of normal feedwater flow results in an increase in the steam generator pressure and temperature. When the header pressure exceeds 950 lb/in.^{2}a, 9.8 seconds after cessation of feedwater flow, the SBCS signals and the bypass valves open. This results in an increased main steam flow incident concurrent with a loss of feedwater. As the RCS begins to cool down due to the increased steam flow, the negative moderator coefficient causes the reactor power to increase. The steam generator inventory is decreasing rapidly due to the full open turbine steam bypass valves and the operating turbine. The primary coolant temperature and pressure are decreasing rapidly when the reactor is tripped at 42.8 seconds due to a low steam generator water level signal. The control element assemblies begin to drop at 43.5 seconds. The reactor power has increased to 124% of full power at this time. However, the DNBR has not decreased below 1.19 during this transient. After the reactor trip, the turbine will trip, but the turbine bypass valves remain open. The RCS will continue to cool down and the RCS pressure and temperature will decrease. Emergency feedwater reaches the steam generators 42 seconds after actuation of the low steam generator water level trip. A safety injection actuation signal (SIAS) is initiated at 75.5 seconds due to the low pressurizer pressure. The pressurizer empties at 83.9 seconds with a pressurizer pressure of 1443 lb/in.^{2}a. A low steam generator pressure signal is generated at 97.8 seconds. The main steam isolation valves fully close at 101.2 seconds. At this time, the total steam generator water inventory reaches its minimum value (5.6% of the nominal inventory). The steam generator inventory will increase as the emergency feedwater continues to operate. The reactor coolant pumps cavitate due to the decreasing primary pressure and temperature at 95.9 seconds. The cavitation causes a flow coastdown which results in an increase in the RCS pressure and temperatures. The pressure in the steam generators begins to increase due to the isolation of the steam generators. It is conservatively assumed that no operator action will be taken until 30 minutes after the initiation of the event. At this time, the operator will take control of the atmospheric dump valves and begin cooldown of the plant in accordance with appropriate emergency procedures. The RCS is then cooled to an average temperature of 350F, at which point shutdown cooling is initiated.

15.2.3.2.4 Barrier Performance

15.2.3.2.4.1 Mathematical Model. The mathematical model used for evaluation of barrier performance is identical to that described in paragraph 15.2.2.5.4.

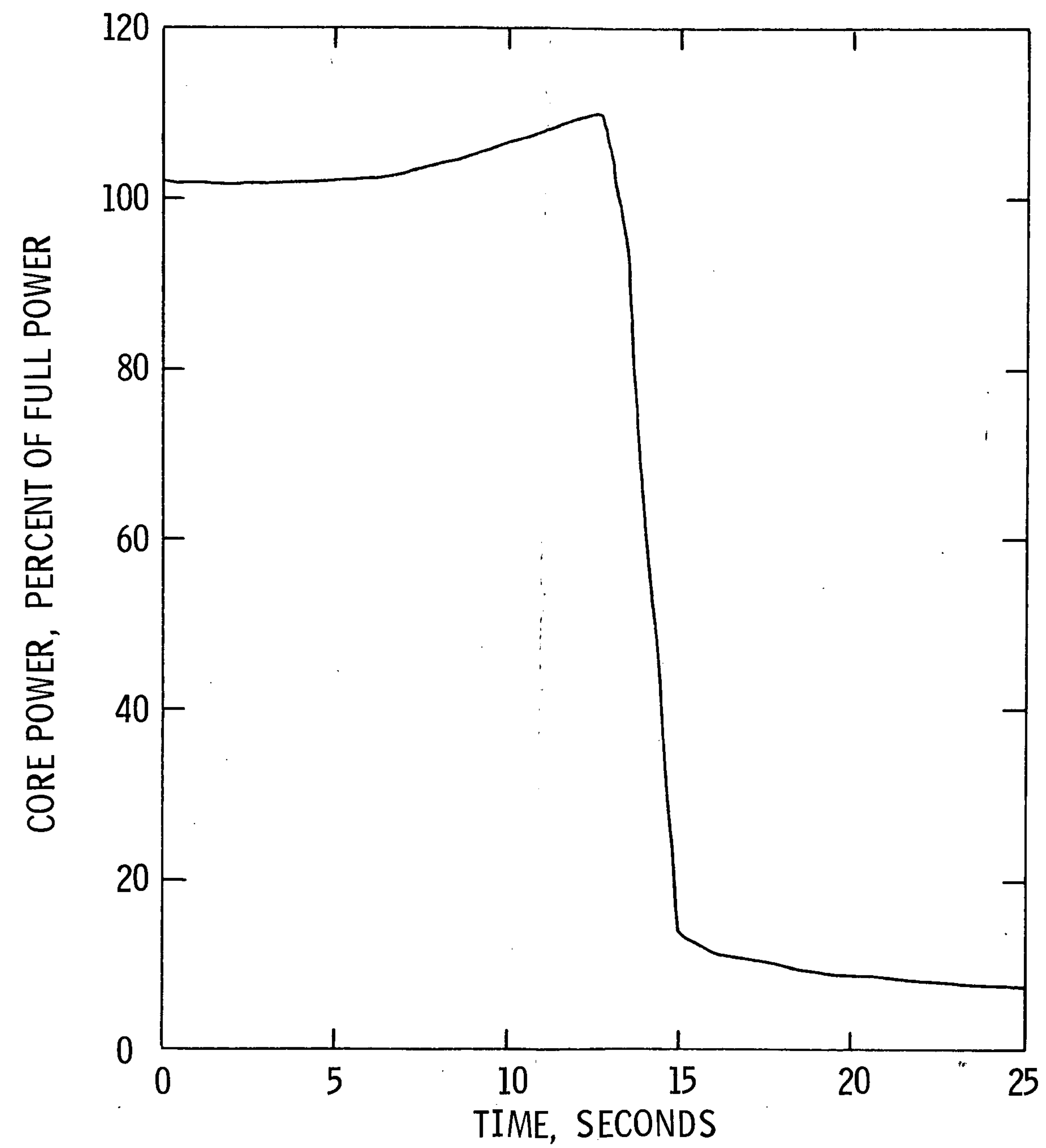
15.2.3.2.4.2 Input Parameters and Initial Conditions. The input parameters and initial conditions used for evaluation of barrier performance are identical to those described in paragraph 15.2.2.5.4.

15.2.3.2.4.3 Results. There are no releases to atmosphere until the operator begins cooldown of the plant 30 minutes after the cessation of feedwater flow. The cooldown at 75F/h is controlled by opening the atmospheric dump valves. After a 2-hour and 20-minute cooldown, the primary system will have reached an average temperature of 350F, at which time the shutdown cooling system will be placed in operation. The approximate total steam release to atmosphere during the course of this transient is 700,000 pounds.

REFERENCES

1. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1973.

2. Henry, R. E., and Fauske, H. K., "The Two-Phase Critical Flow of One-Component Mixture in Nozzles, Orifices, and Short Tubes," Journal of Heat Transfer, May 1971.

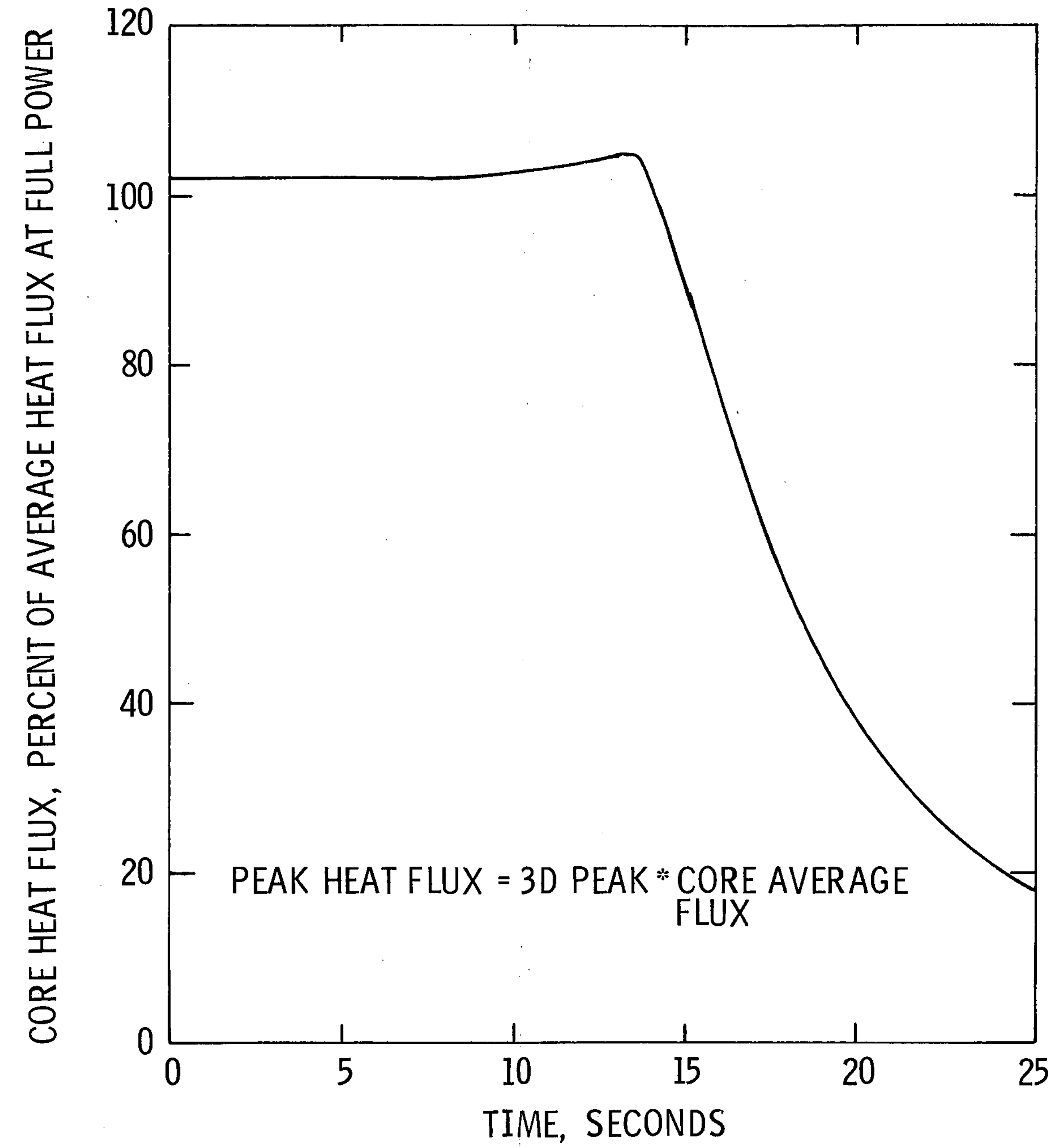


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM
NEUTRON POWER vs. TIME

Figure 15.2-1

10079-11D-983
100C6



10079-11D-984
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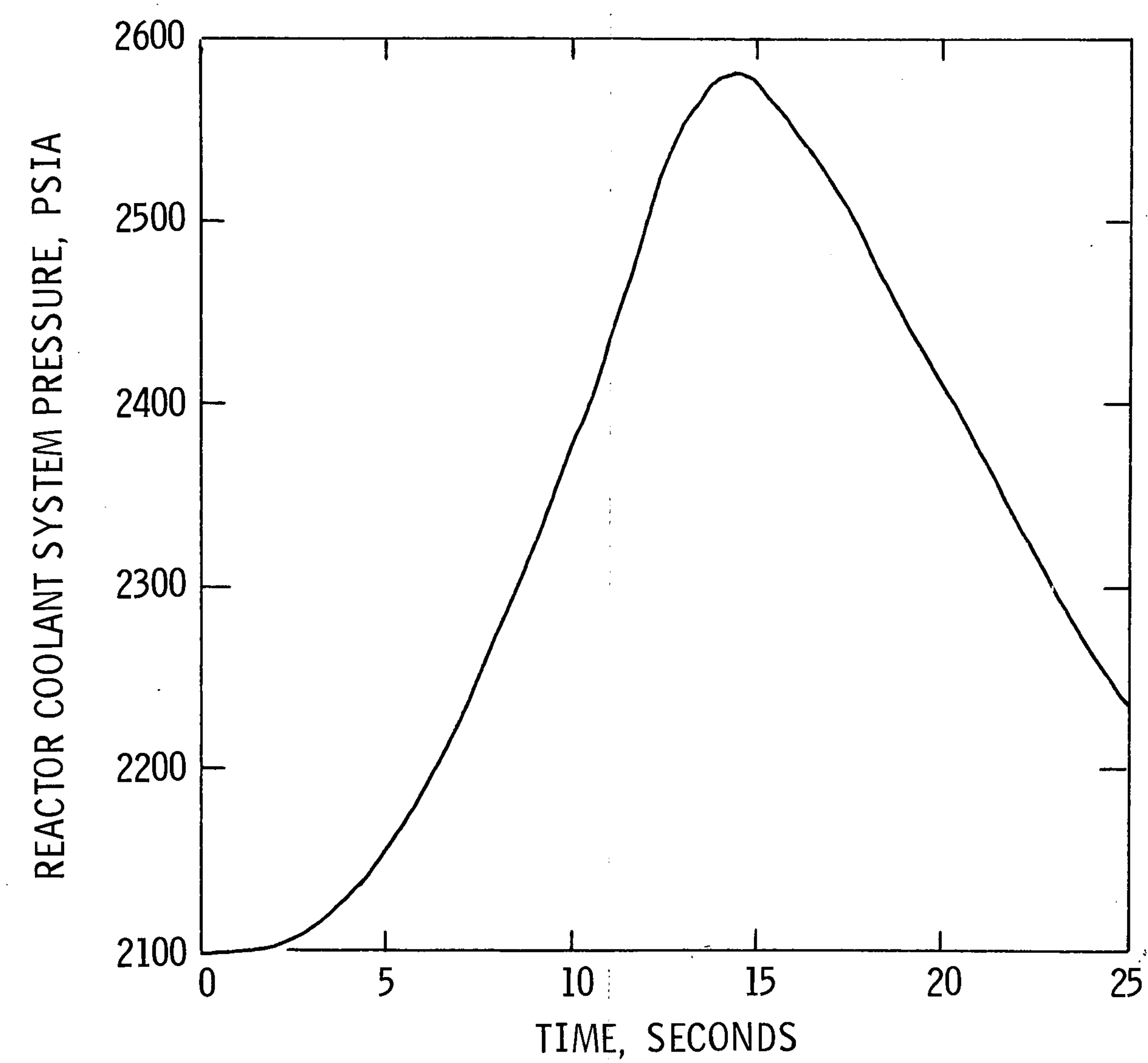
LOSS OF CONDENSER VACUUM
CORE AVERAGE HEAT FLUX vs. TIME

Figure 15.2-2

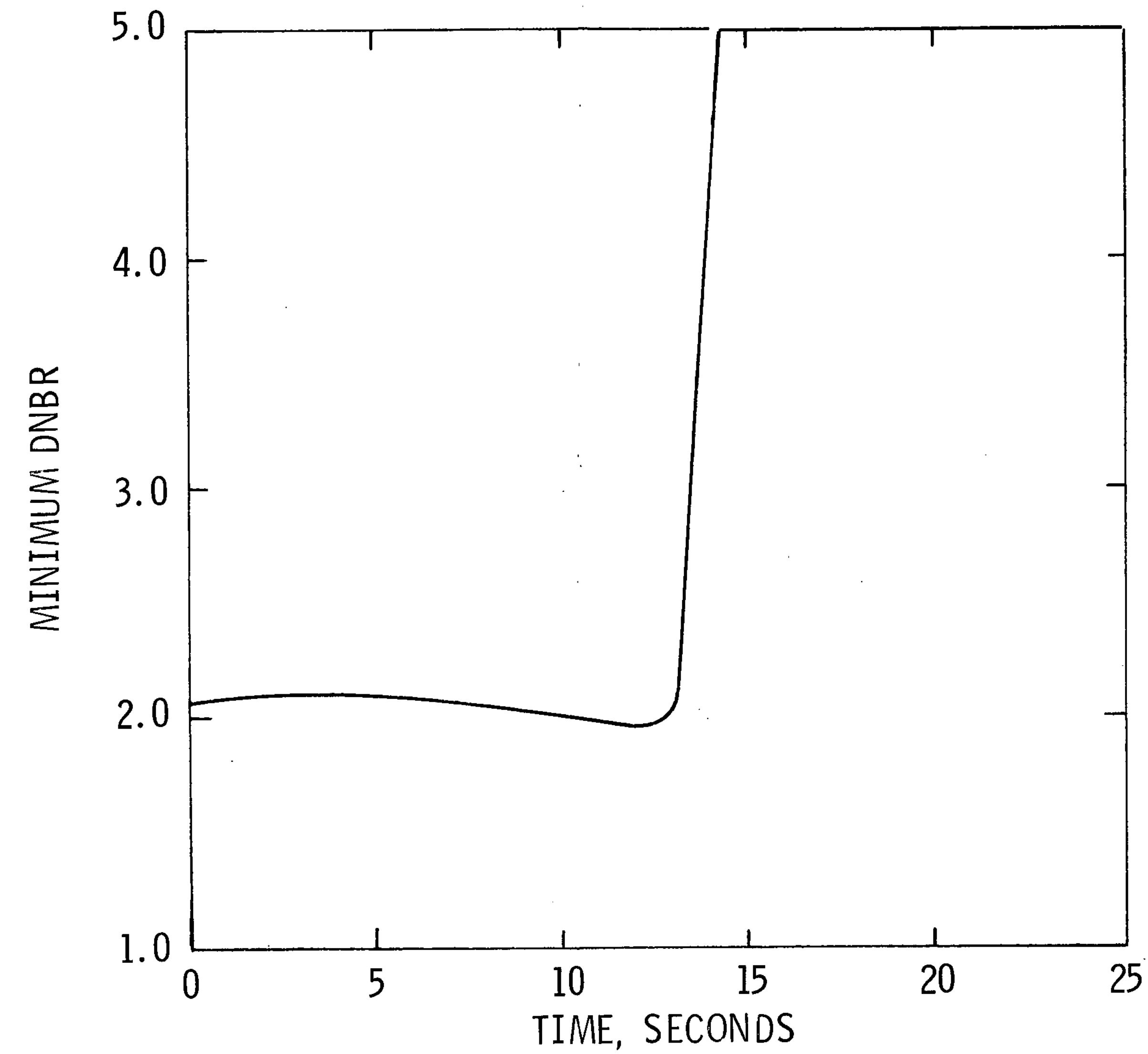


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM REACTOR COOLANT SYSTEM PRESSURE vs. TIME

Figure 15.2-3

10079-11D-985
10OC6



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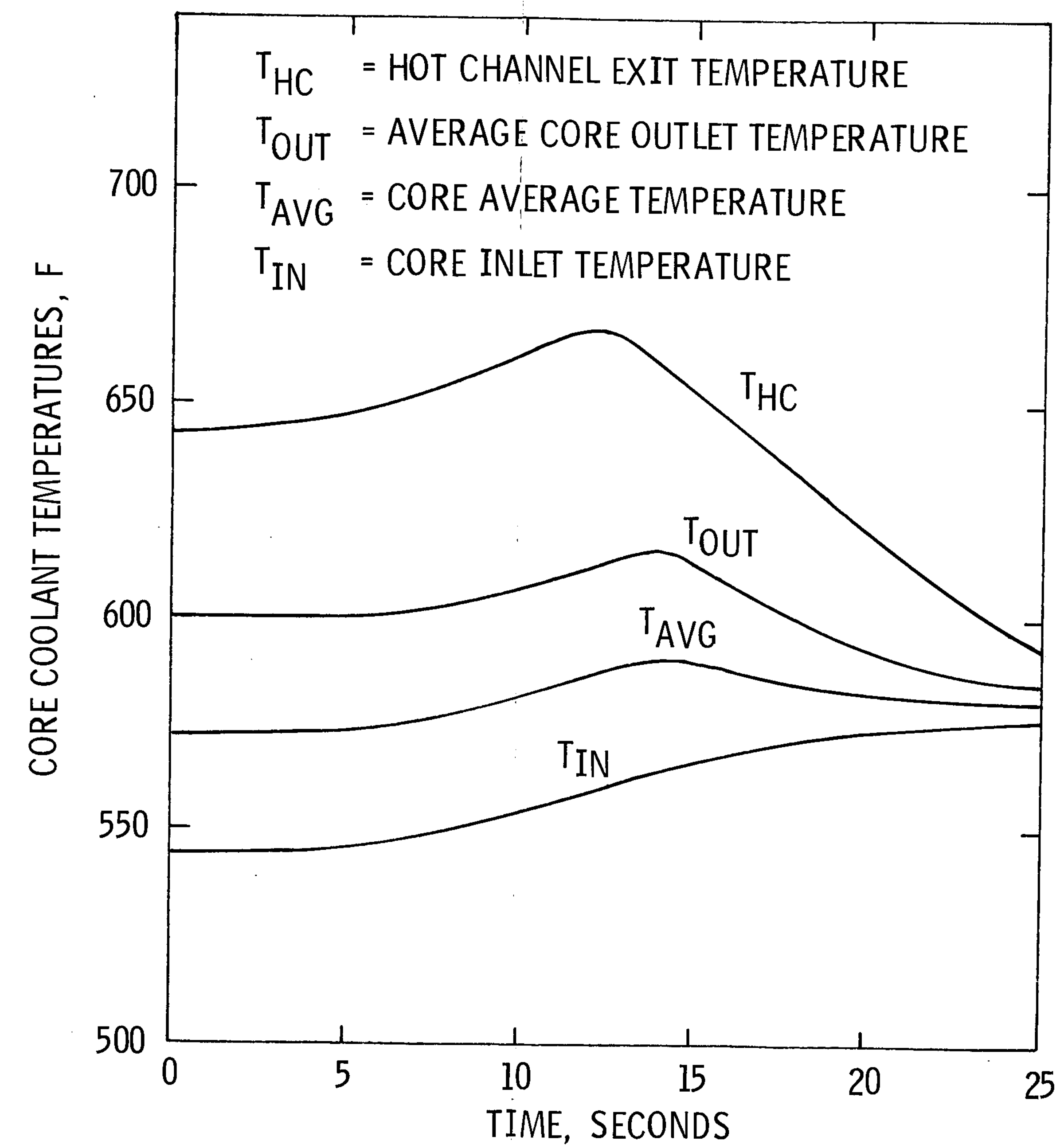
Units 2 & 3

LOSS OF CONDENSER VACUUM
MINIMUM DNBR vs. TIME

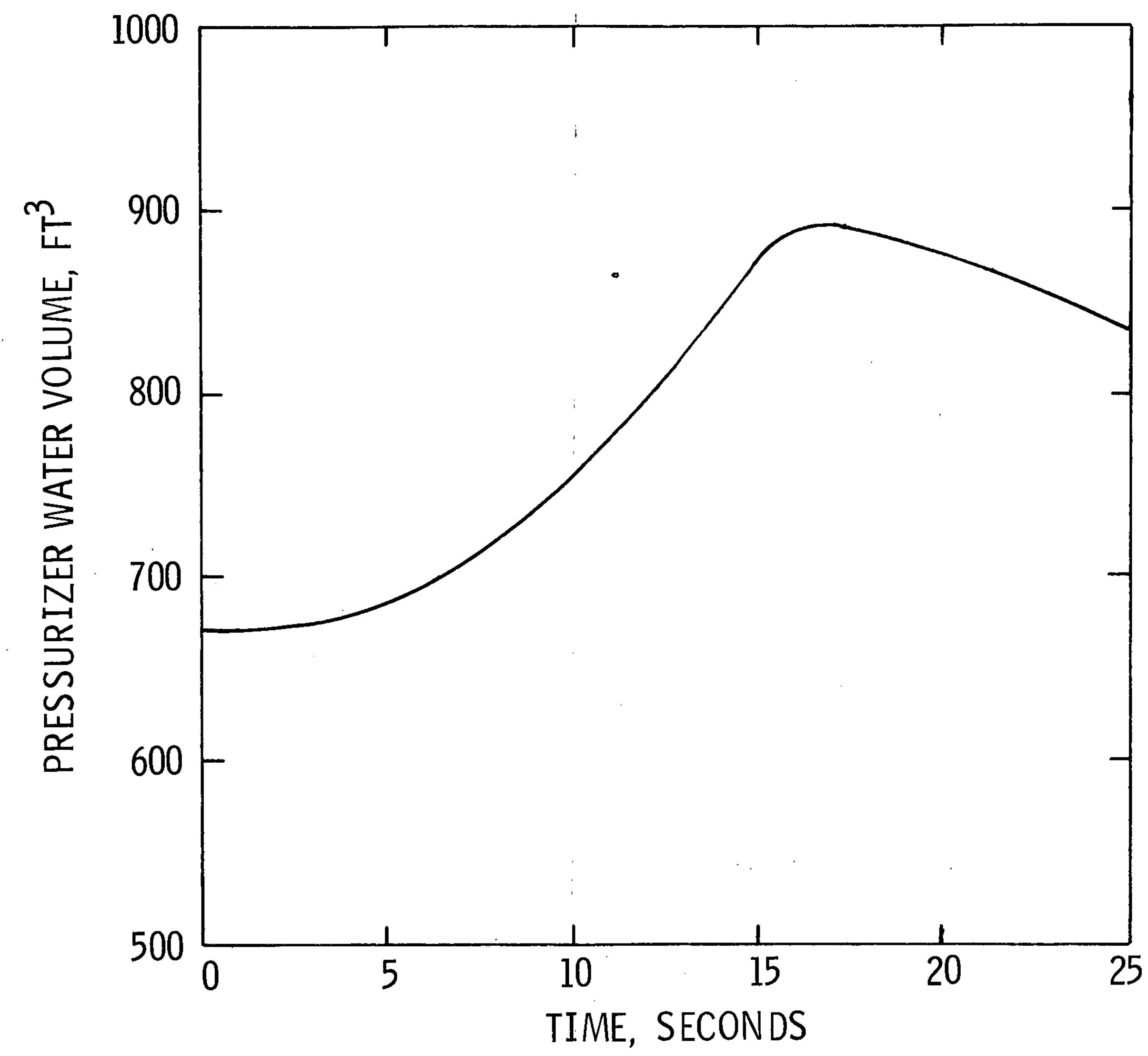
Figure 15.2-4

10079-11D-986
10OC6



10079-11D-987
10OC6

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
LOSS OF CONDENSER VACUUM PRIMARY SAFETY VALVE FLOWRATE vs. TIME
Figure 15.2-5

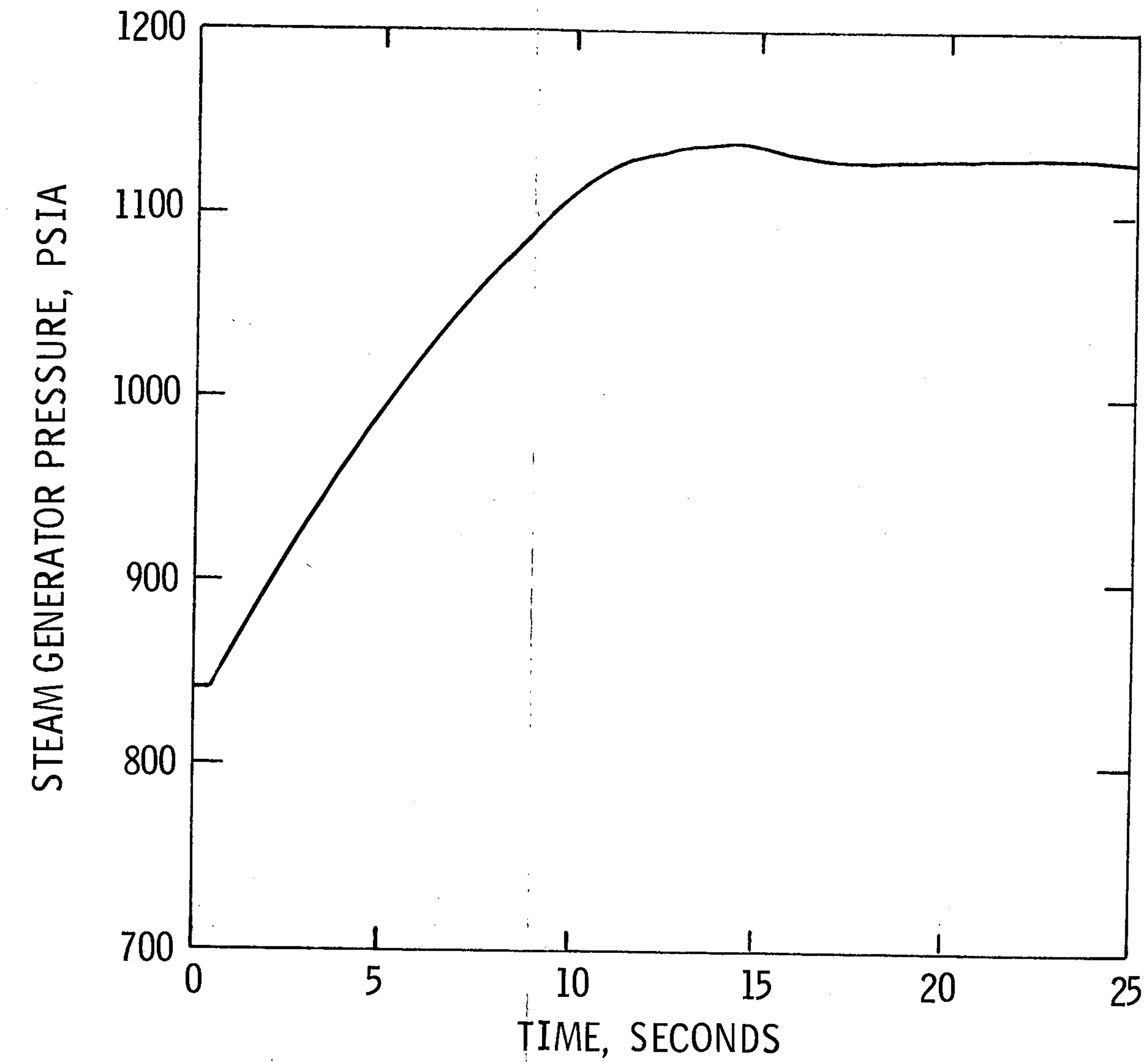


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM PRESSURIZER WATER VOLUME vs. TIME

Figure 15.2-6

10079-11D-988
10OC6



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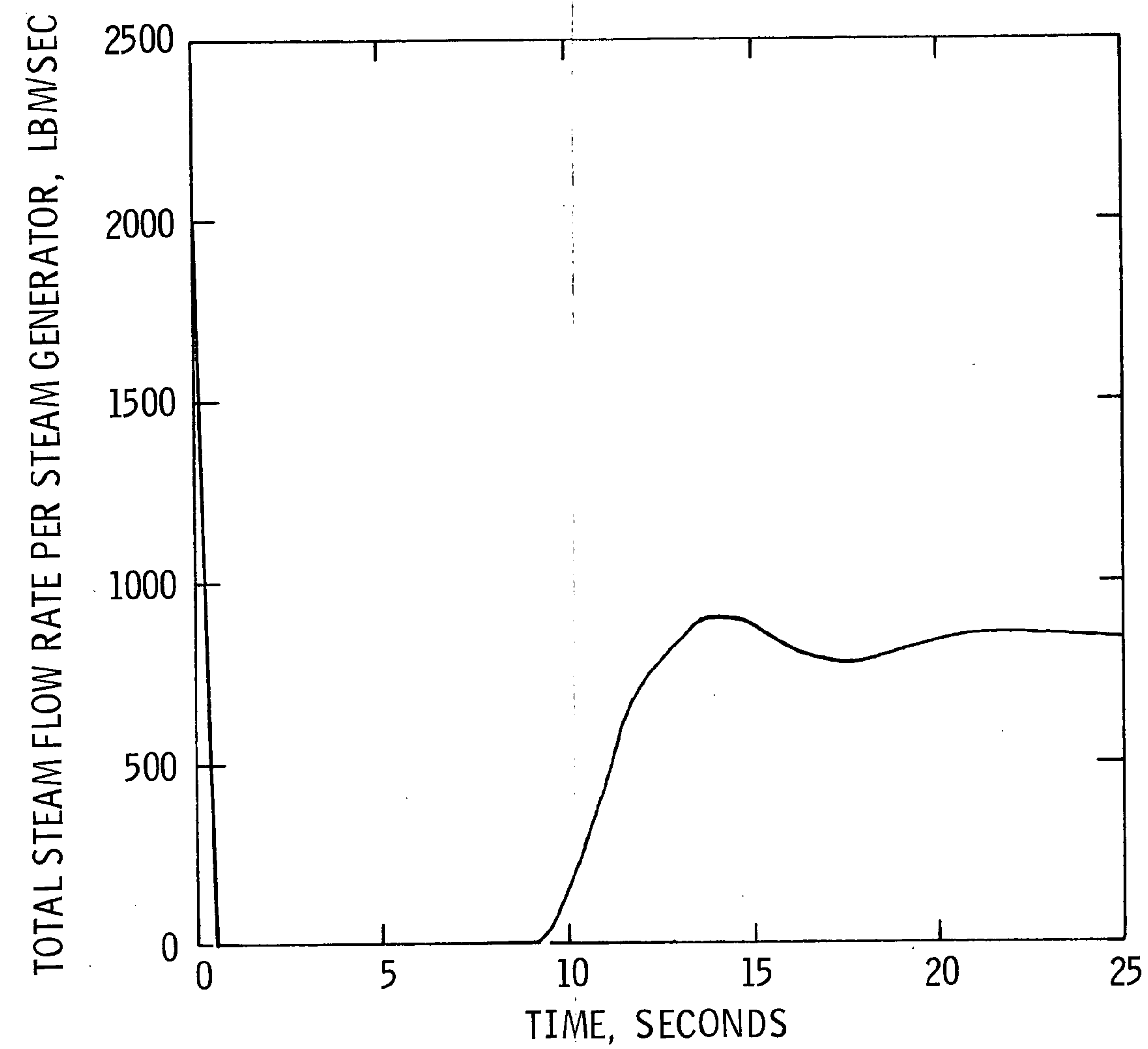
Units 2 & 3

LOSS OF CONDENSER
VACUUM STEAM GENERATOR
PRESSURE vs. TIME

Figure 15.2-7

10079-11D-989
10OC6



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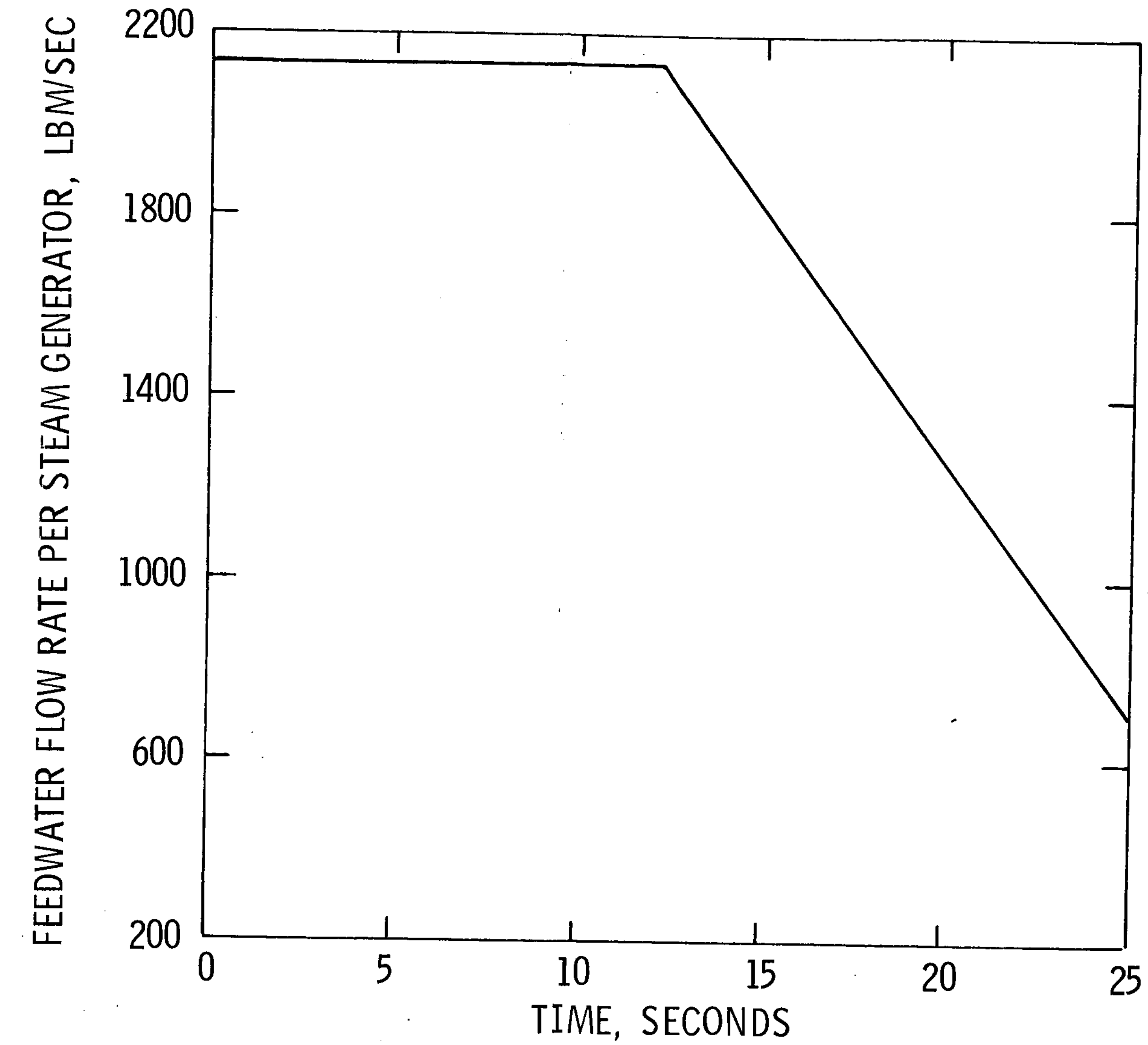
Units 2 & 3

LOSS OF CONDENSER VACUUM
TOTAL STEAM FLOWRATE PER
STEAM GENERATOR vs. TIME

Figure 15.2-8

10079-11D-990
10OC6



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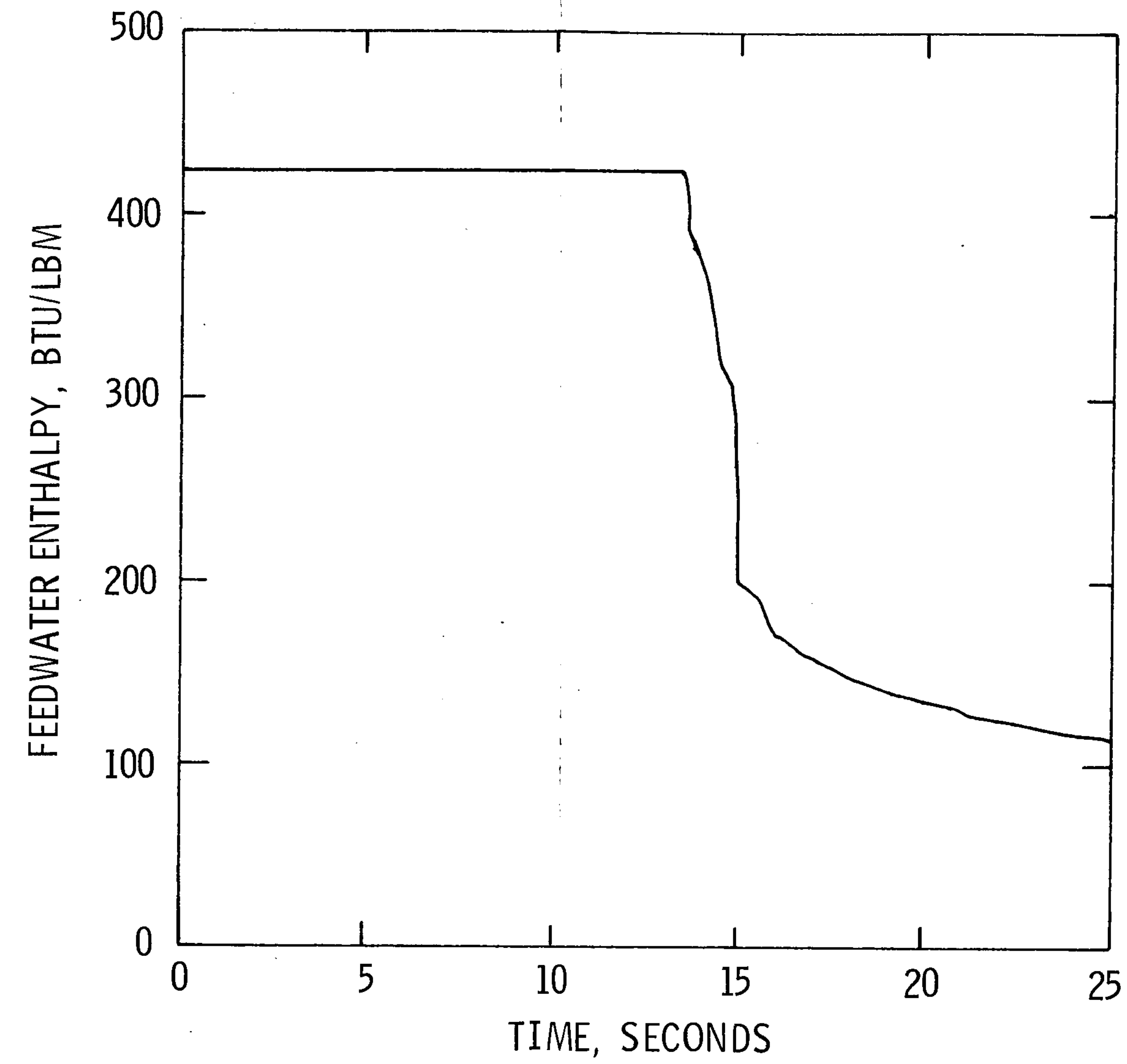
Units 2 & 3

LOSS OF CONDENSER VACUUM
FEEDWATER FLOWRATE PER
STEAM GENERATOR vs. TIME

Figure 15.2-9

10079-11D-991
10OC6



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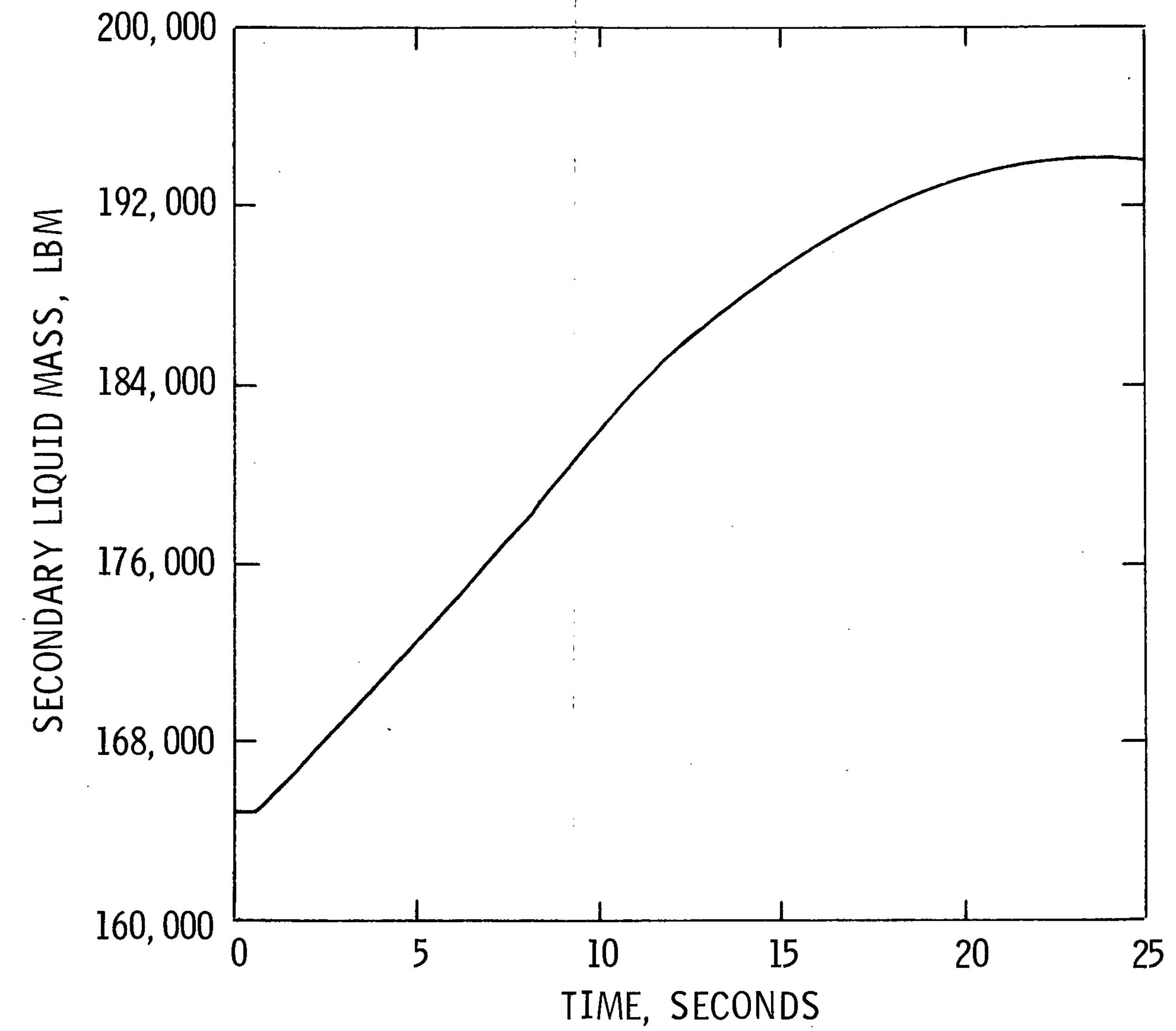
Units 2 & 3

LOSS OF CONDENSER VACUUM
FEEDWATER ENTHALPY vs. TIME

Figure 15.2-10

10079-11D-992
100C6



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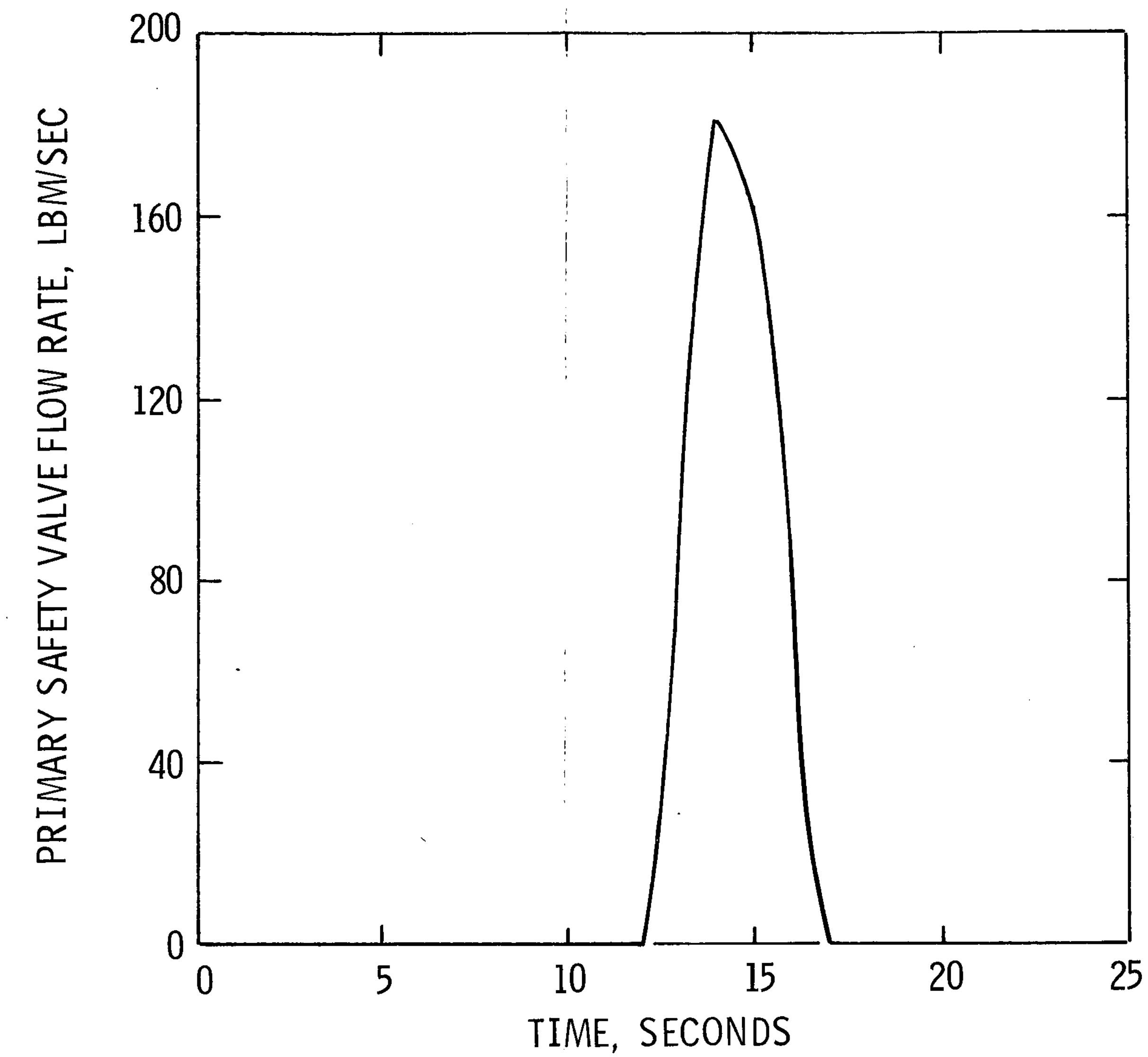
Units 2 & 3

LOSS OF CONDENSER VACUUM SECONDARY LIQUID MASS vs. TIME

Figure 15.2-11

10079-11D-993
10OC6

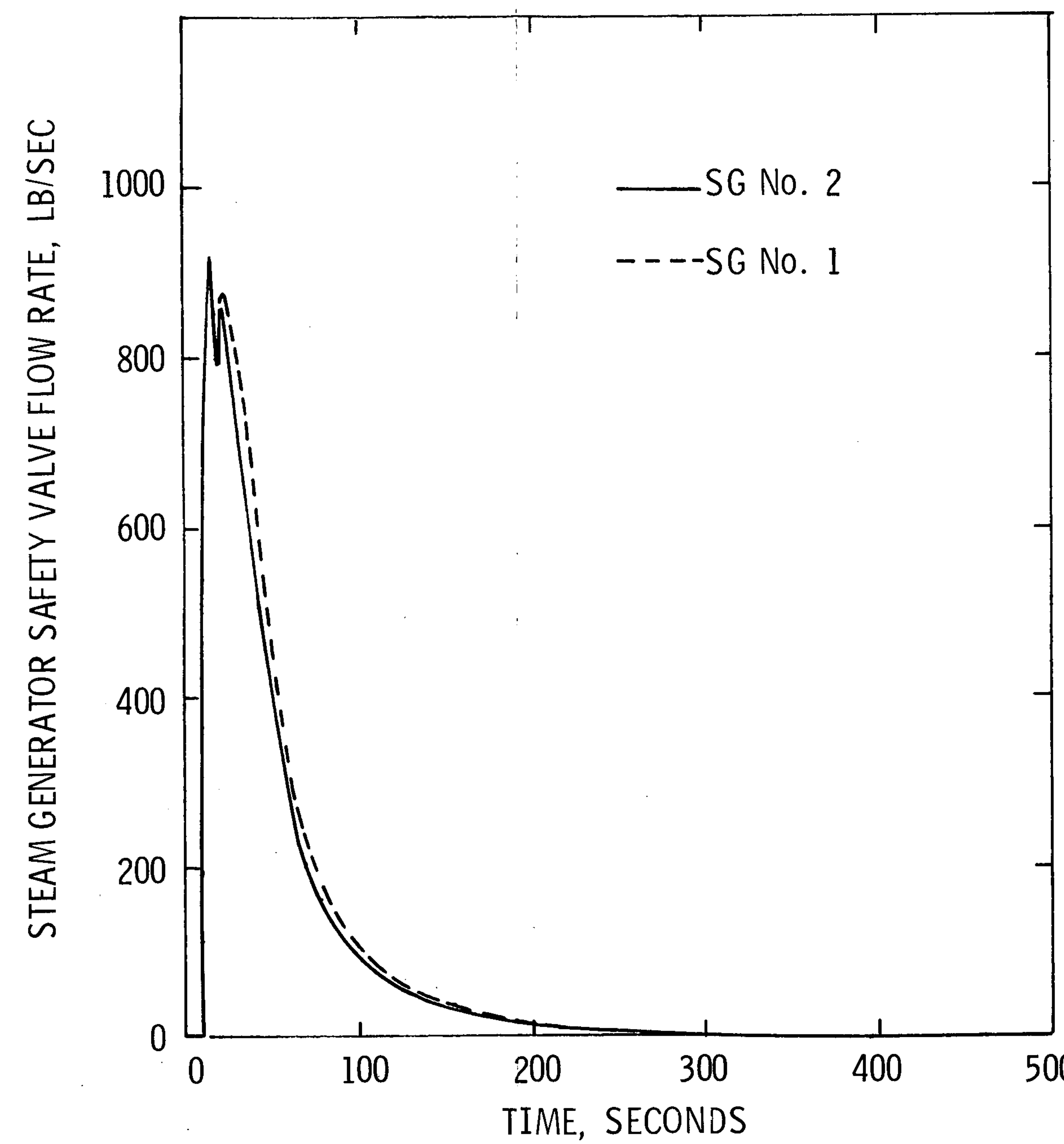


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM PRIMARY SAFETY VALVE FLOWRATE vs. TIME

Figure 15.2-12

10079-11D-994
10OC6



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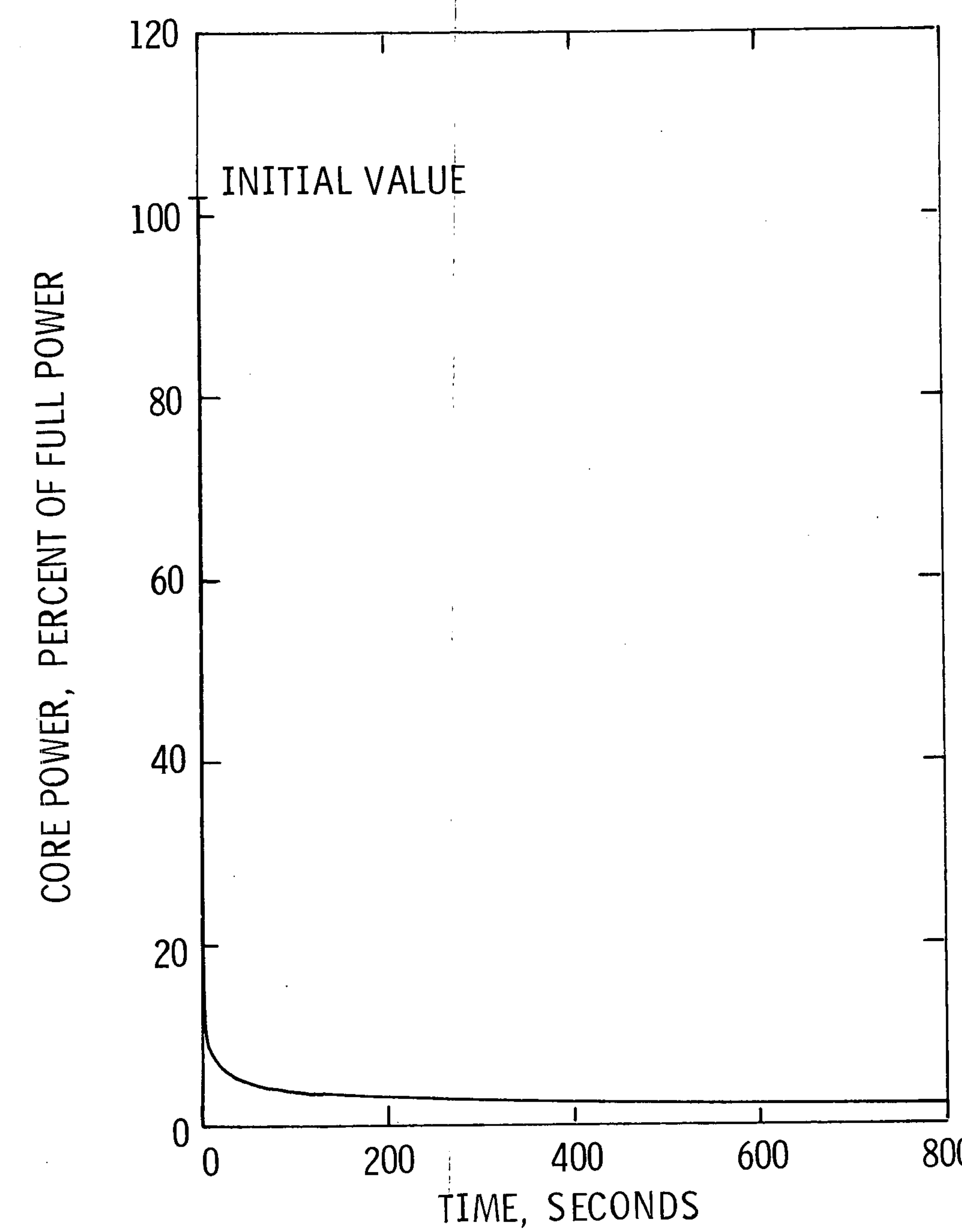
Units 2 & 3

LOSS OF CONDENSER VACUUM
STEAM GENERATOR SAFETY
VALVE FLOWRATE vs. TIME

Figure 15.2-13

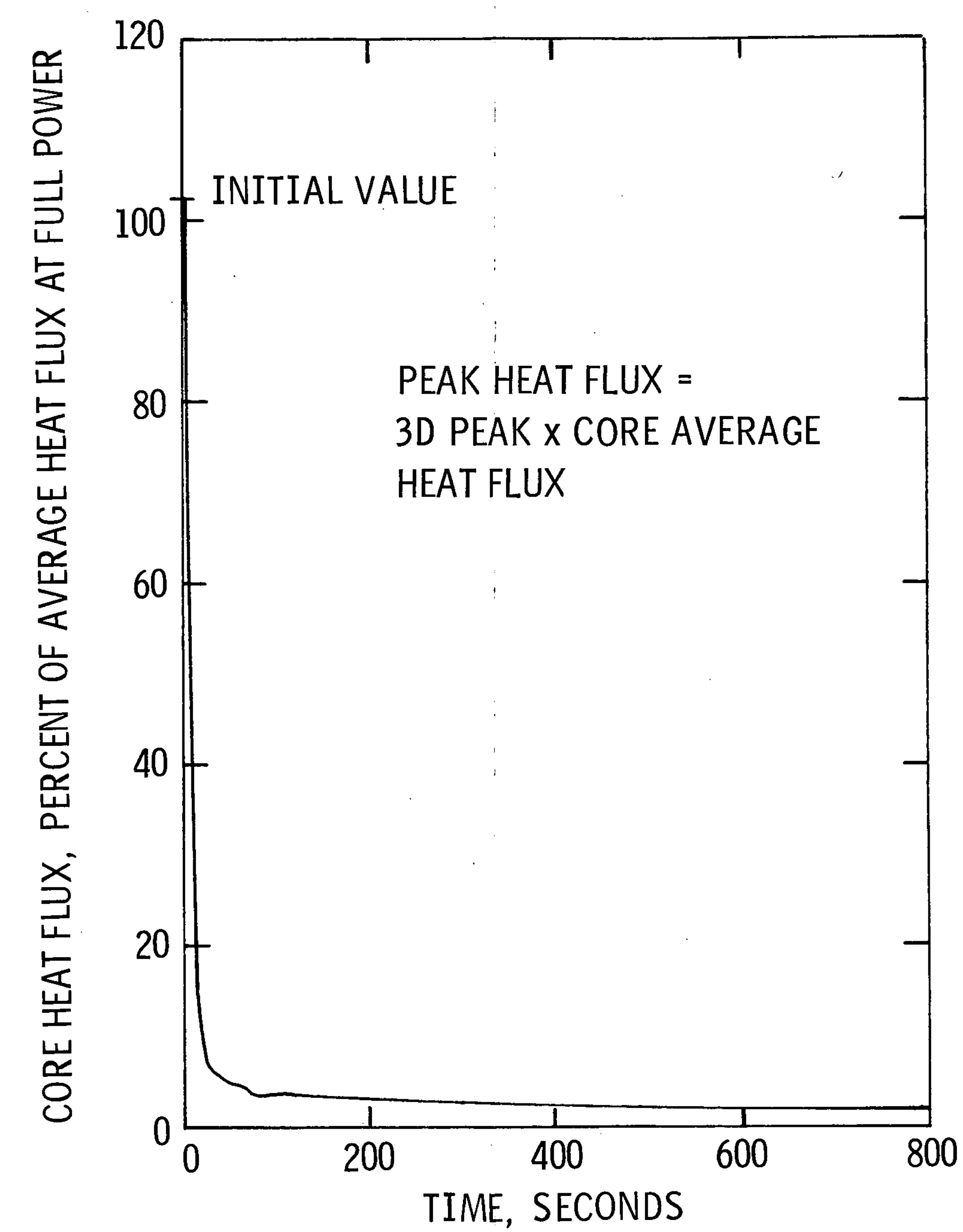
10079-11D-995
10OC6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC
POWER CORE POWER
vs. TIME

Figure 15.2-14

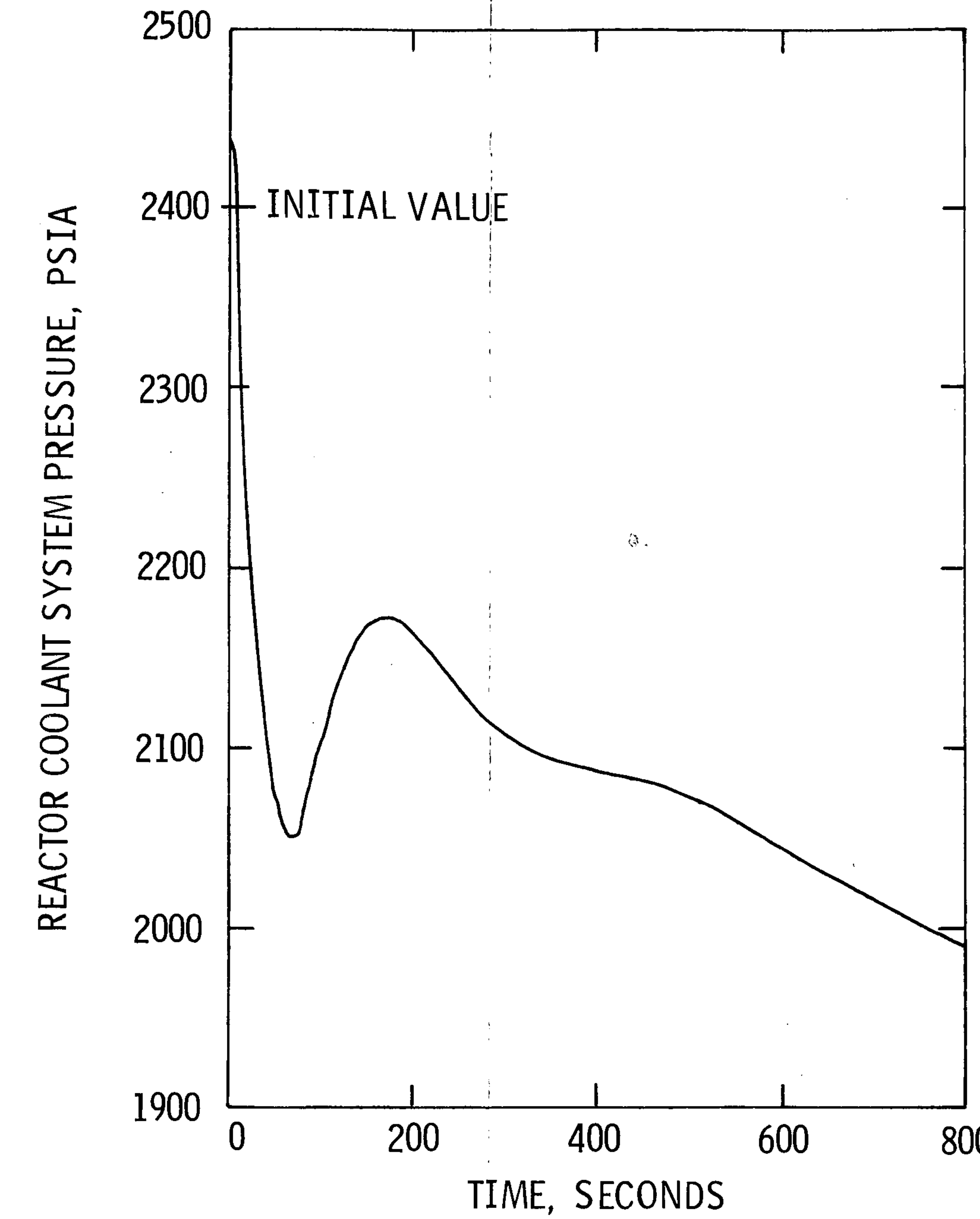


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
CORE AVERAGE HEAT FLUX
vs. TIME

Figure 15.2-15

10079-11D-997
10OC6

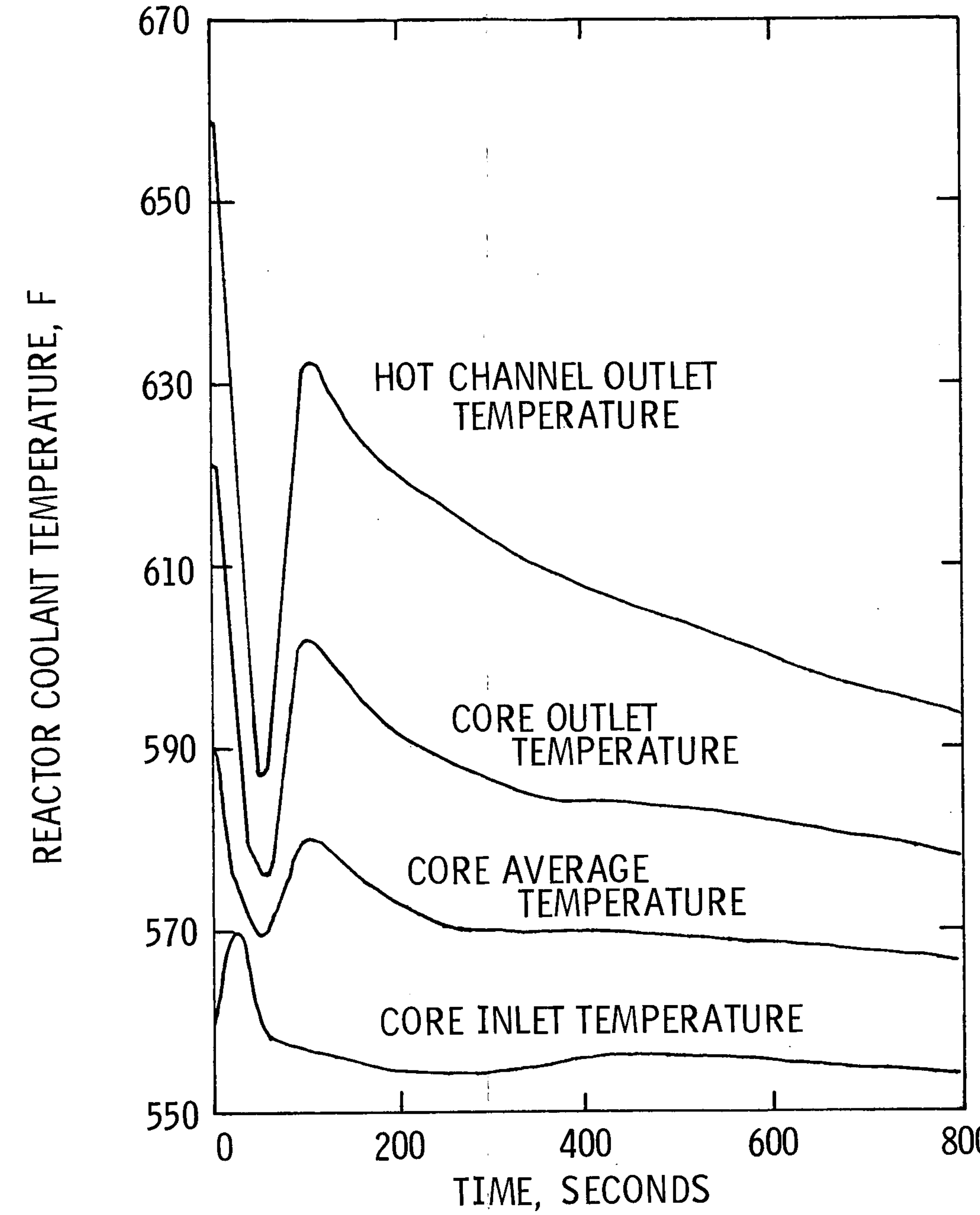


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
REACTOR COOLANT SYSTEM
PRESSURE vs. TIME

Figure 15.2-16

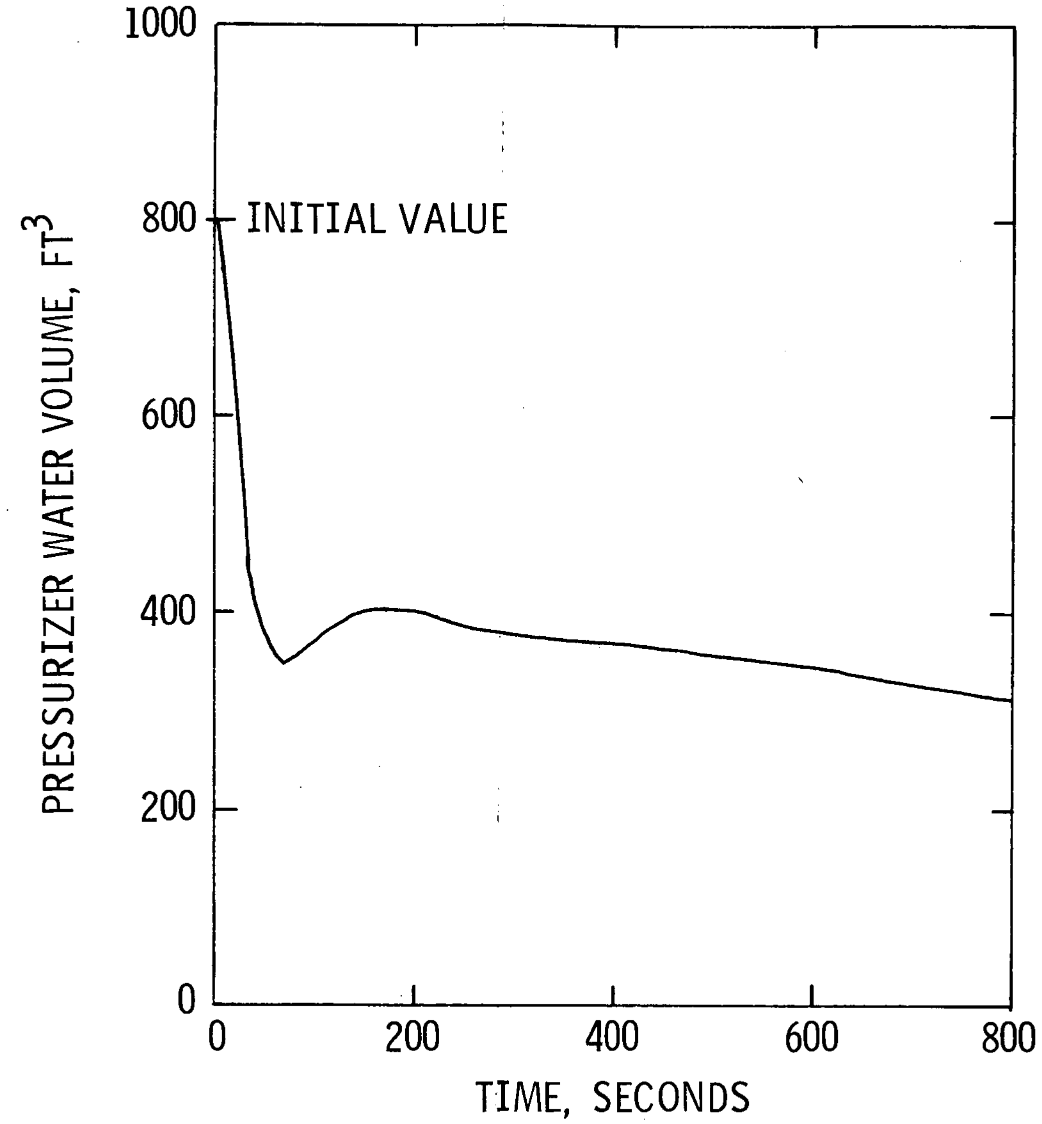
10079-11D-998
10OC6



SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
REACTOR COOLANT TEMPERATURES
vs. TIME

Figure 15.2-17

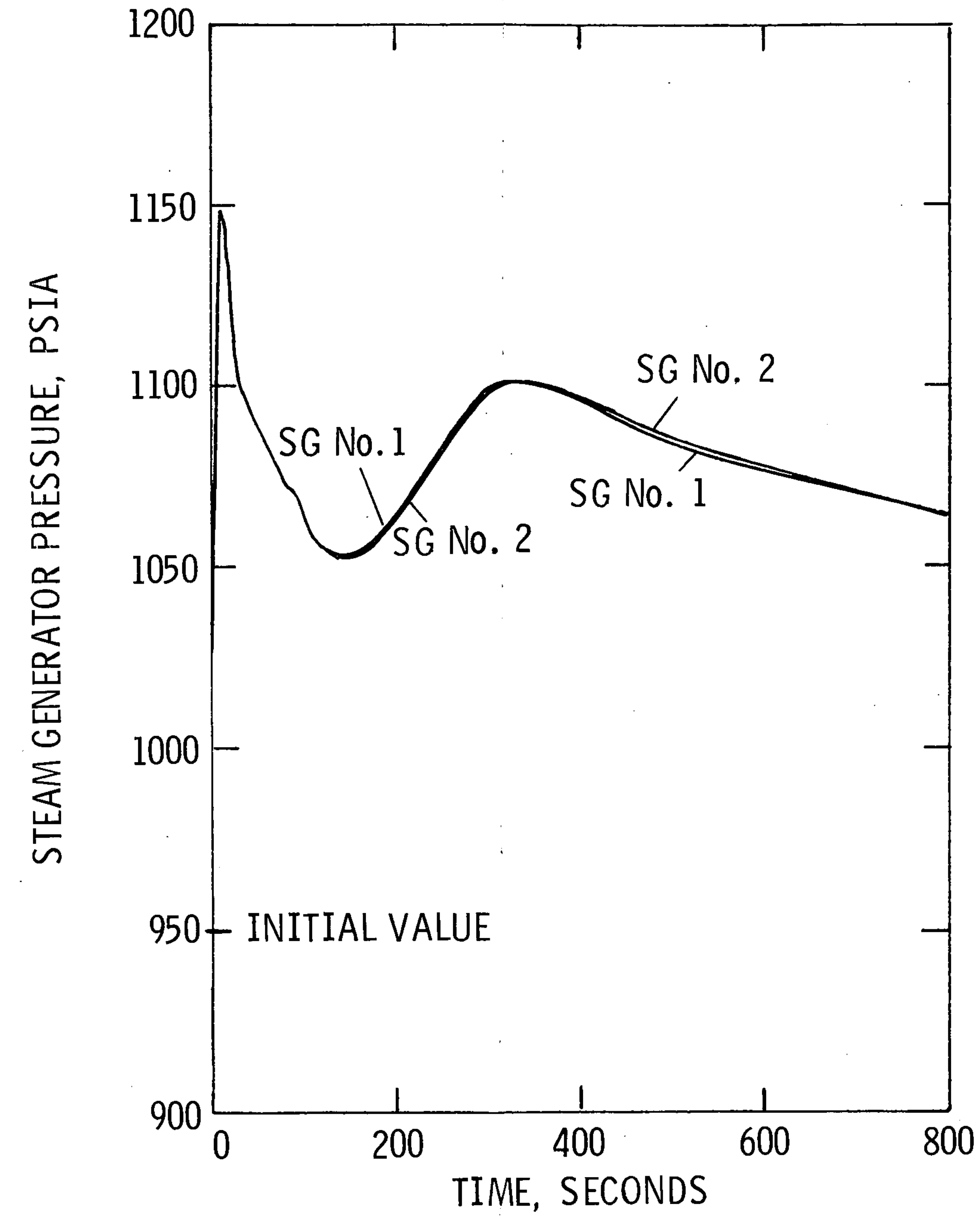


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
PRESSURIZER WATER
VOLUME vs. TIME

Figure 15.2-18

10079-11D-1000

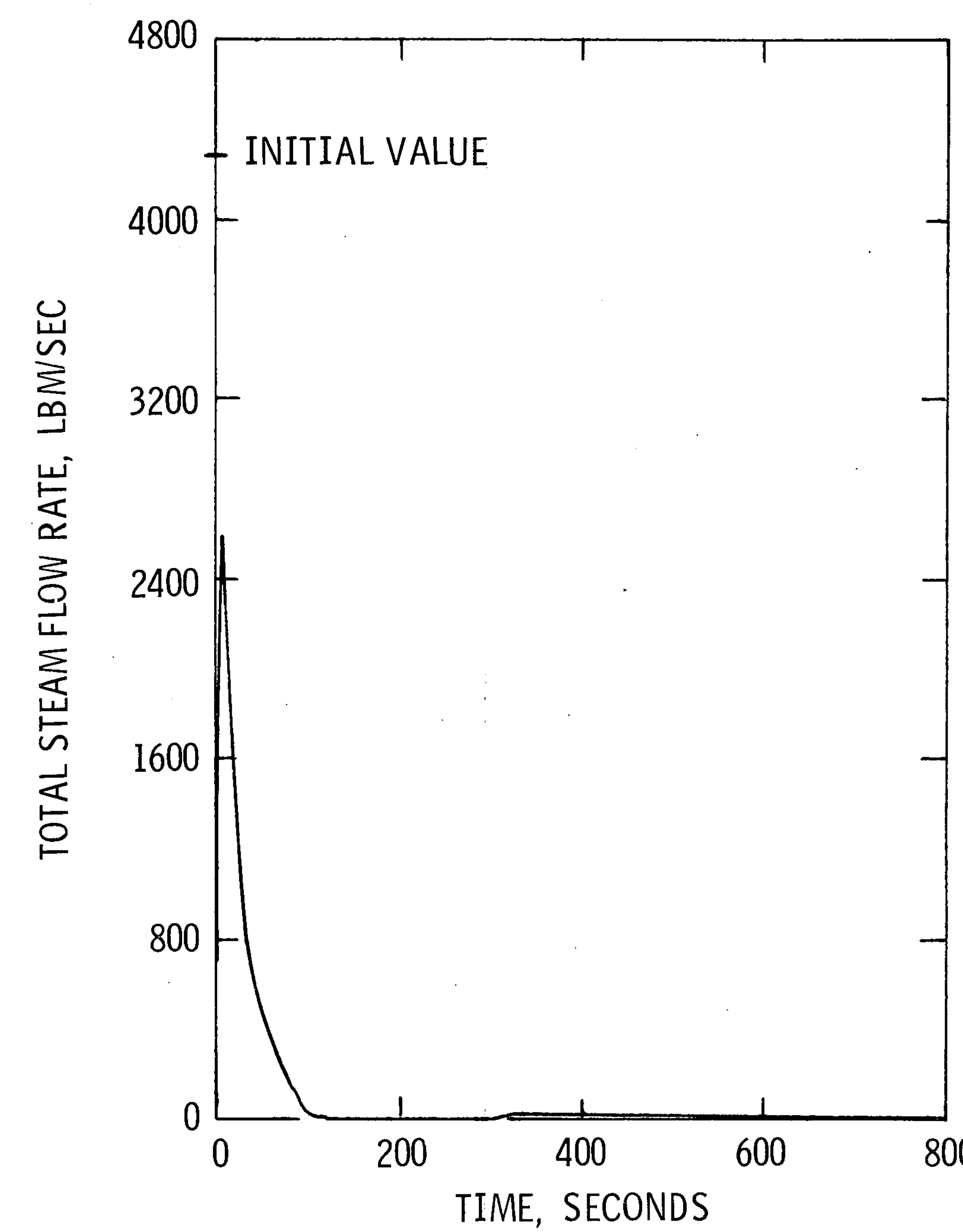


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
STEAM GENERATOR PRESSURE vs. TIME

Figure 15.2-19

10079-11D-1001
10OC6

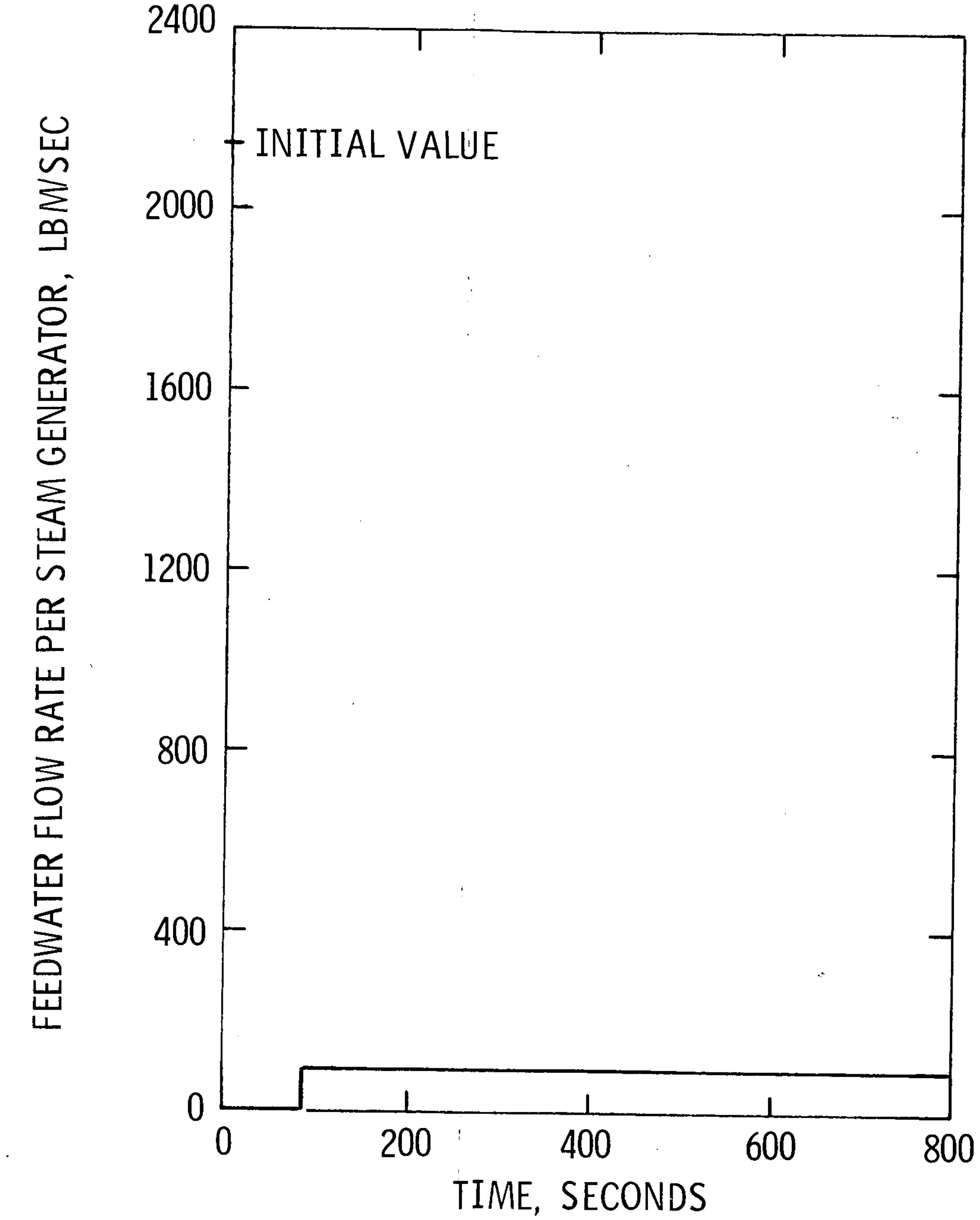


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
TOTAL STEAM FLOWRATE vs. TIME

Figure 15.2-20

10079-11D-1002
10OC6

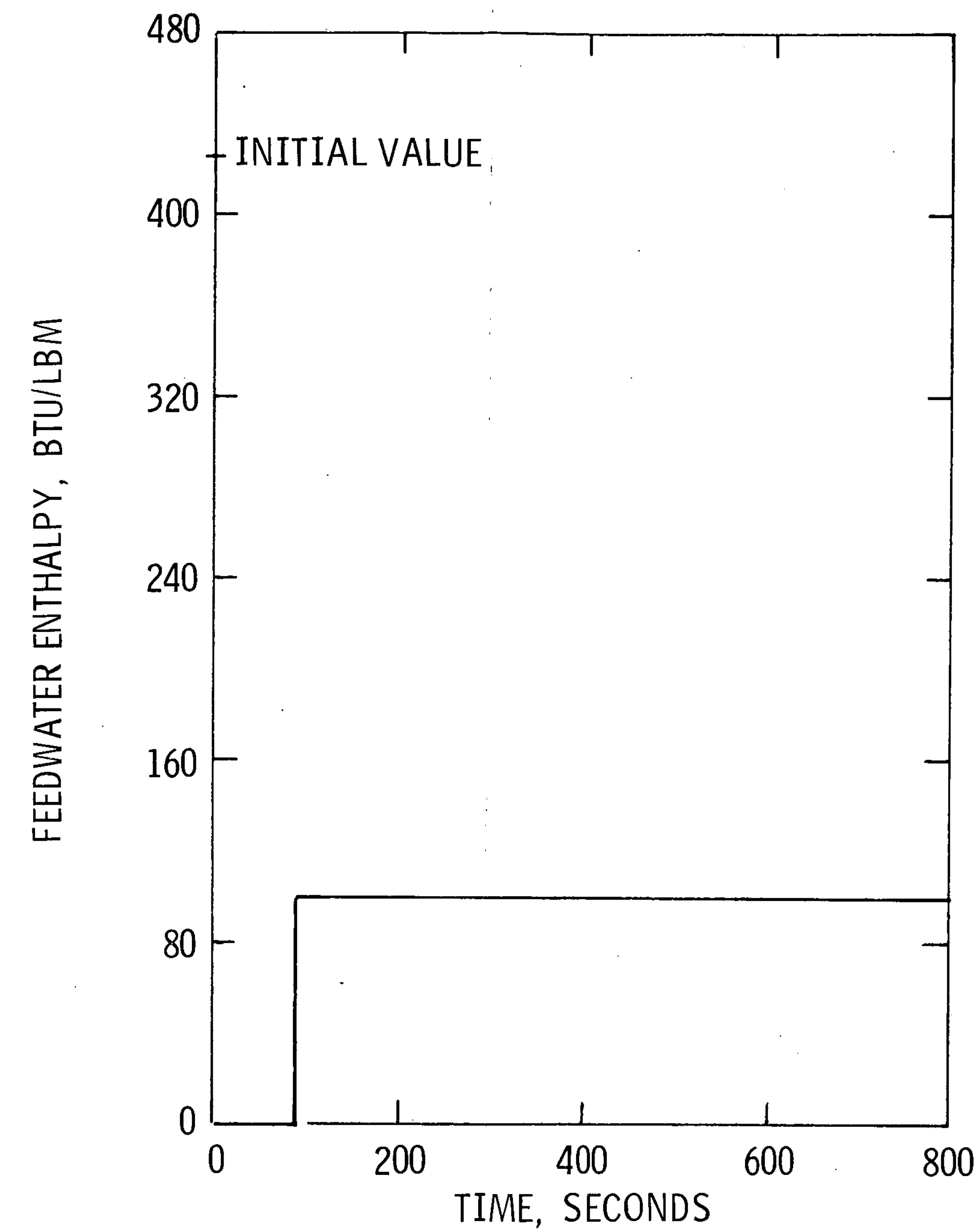


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
FEEDWATER FLOWRATE PER STEAM
GENERATOR vs. TIME

Figure 15.2-21

10079-11D-1003
10OC6

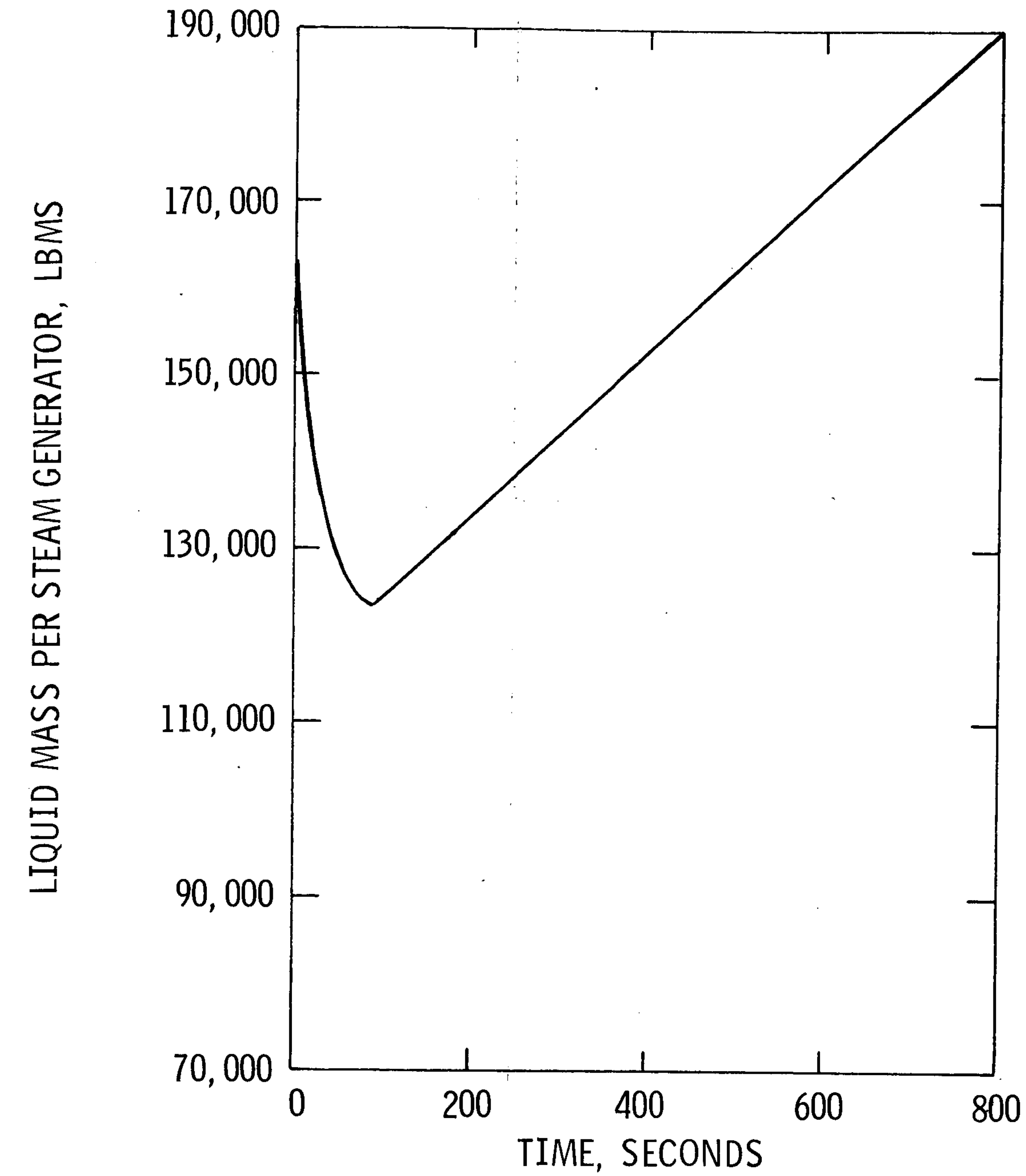


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
FEEDWATER ENTHALPY vs. TIME

Figure 15.2-22

10079-11D-1004
100C6

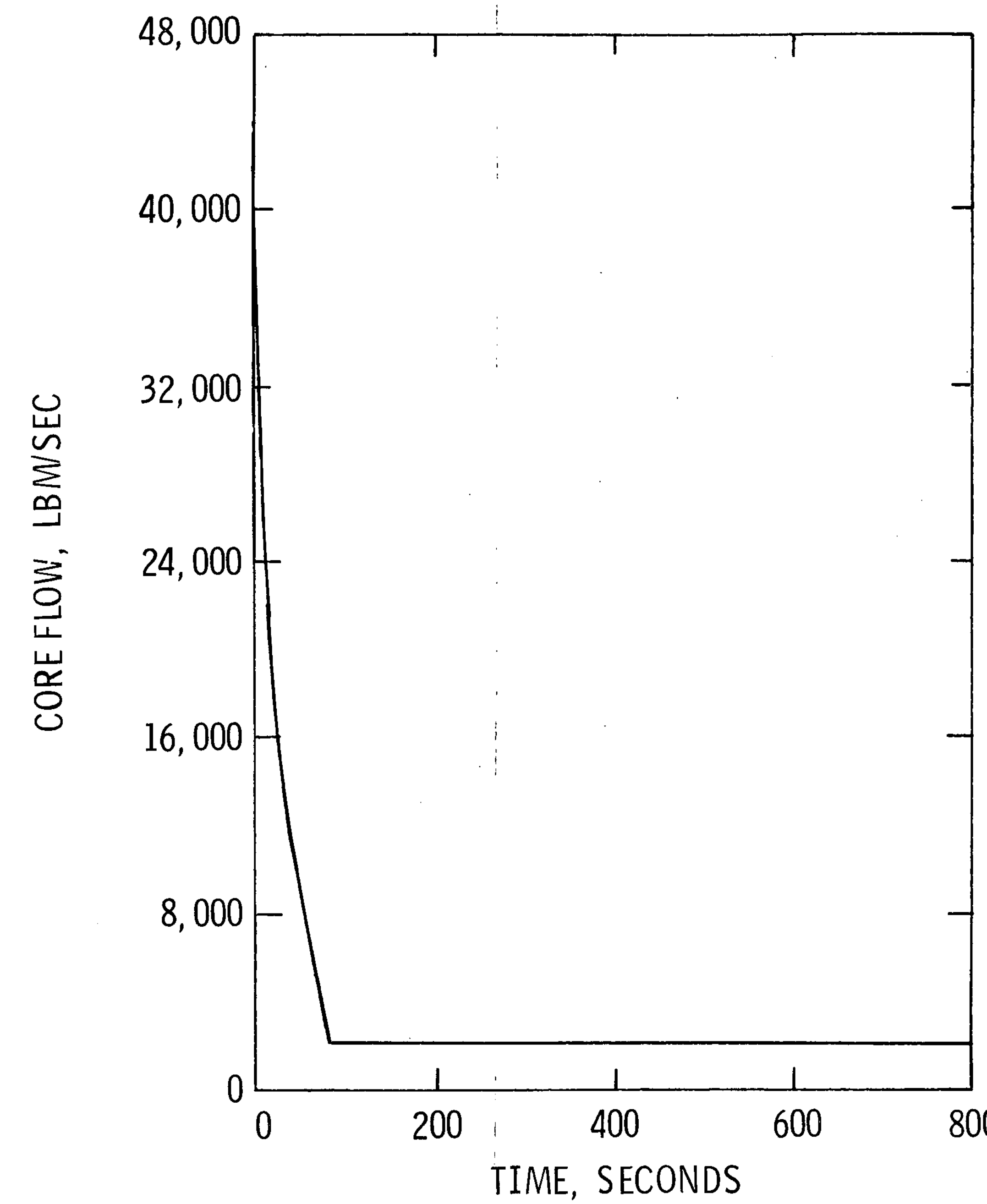


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
LIQUID MASS PER STEAM
GENERATOR vs. TIME

Figure 15.2-23

10079-11D-1005
10OC6

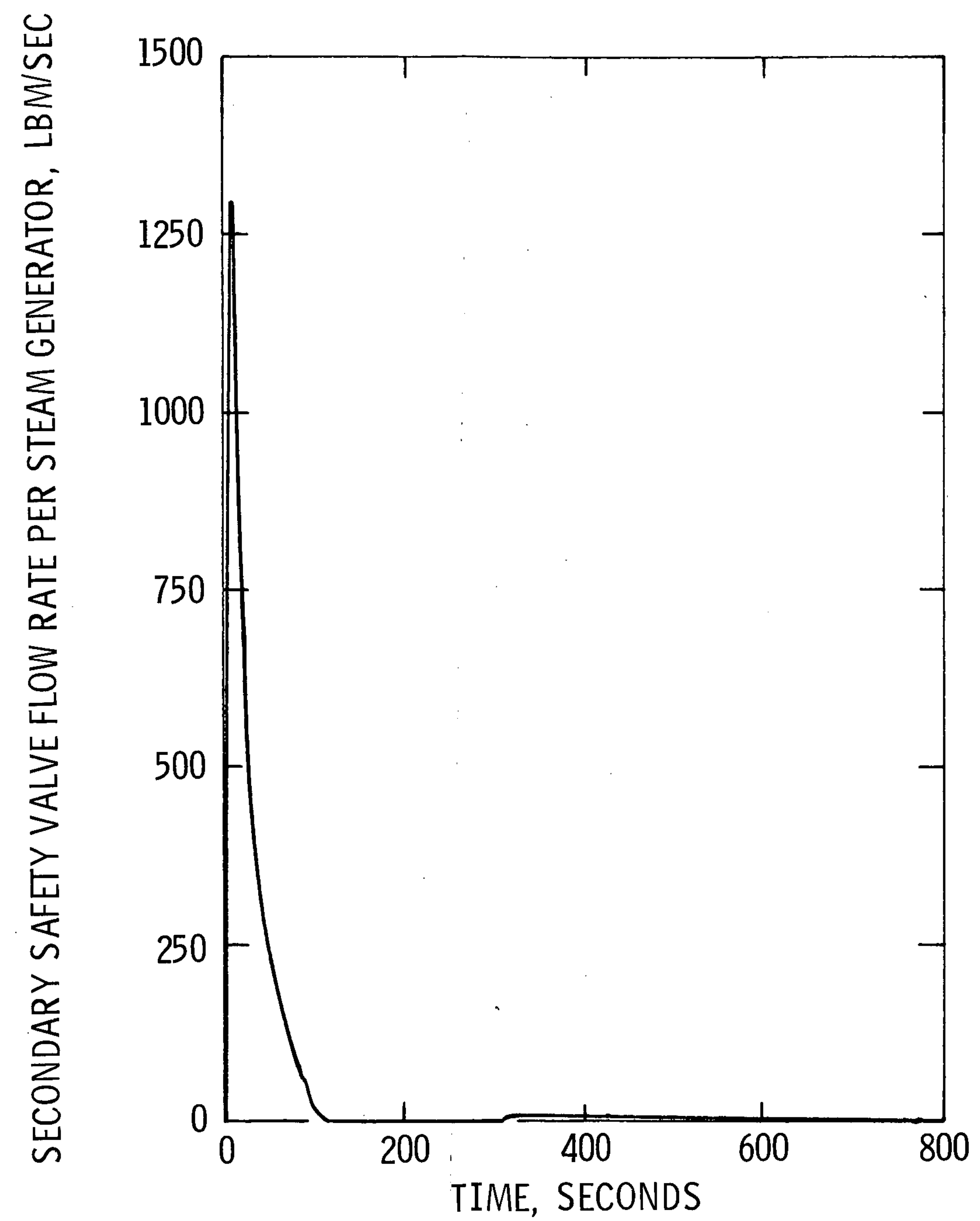


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
CORE FLOWRATE vs. TIME

Figure 15.2-24

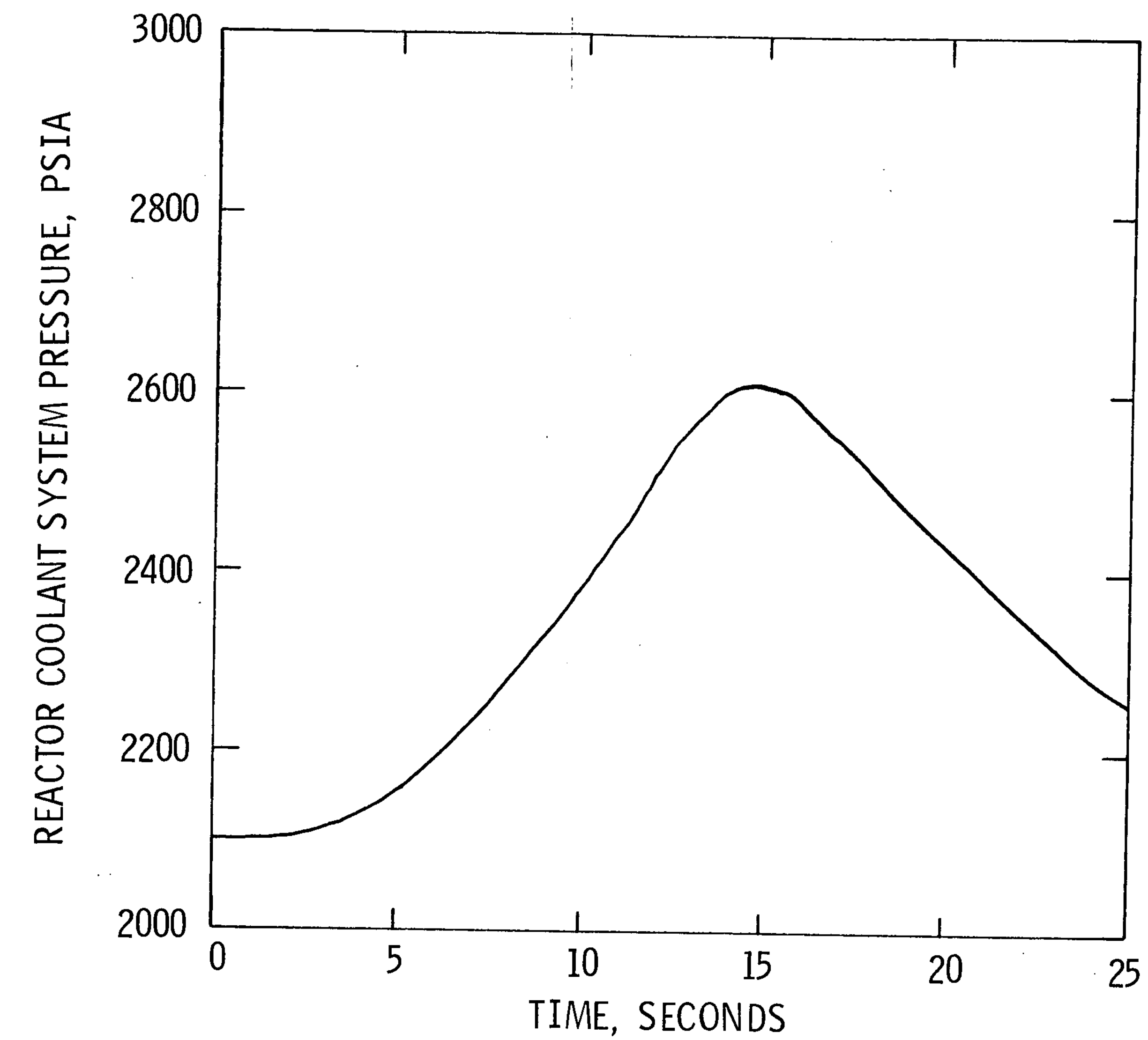
10079-11D-1006
10OC6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF ALL NORMAL AC POWER
SECONDARY SAFETY VALVE FLOWRATE
PER STEAM GENERATOR vs. TIME

Figure 15.2-25

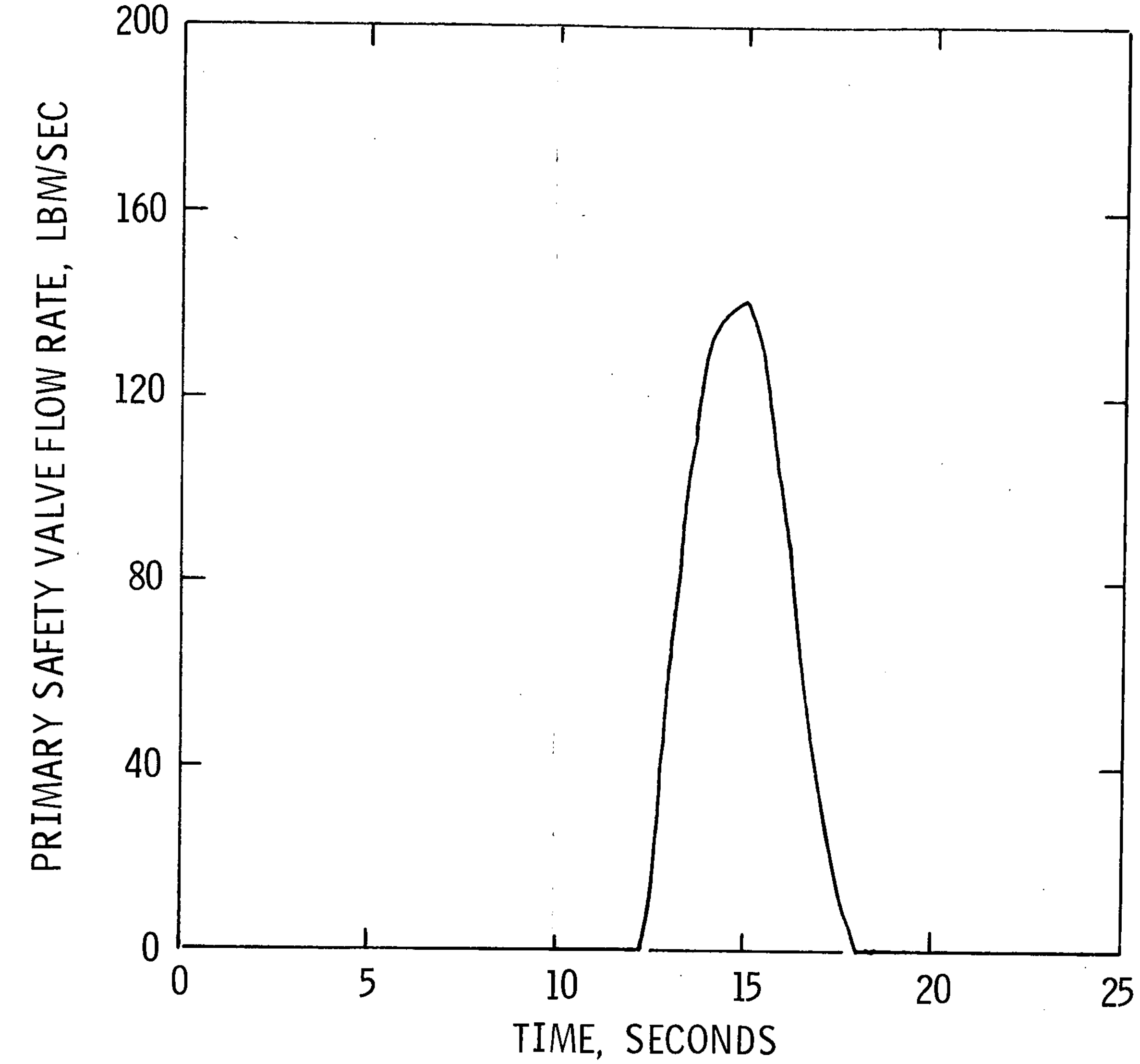


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM WITH A CONCURRENT SINGLE FAILURE OF AN ACTIVE COMPONENT REACTOR COOLANT SYSTEM PRESSURE vs. TIME

Figure 15.2-26

10079-11D-1008
10OC6

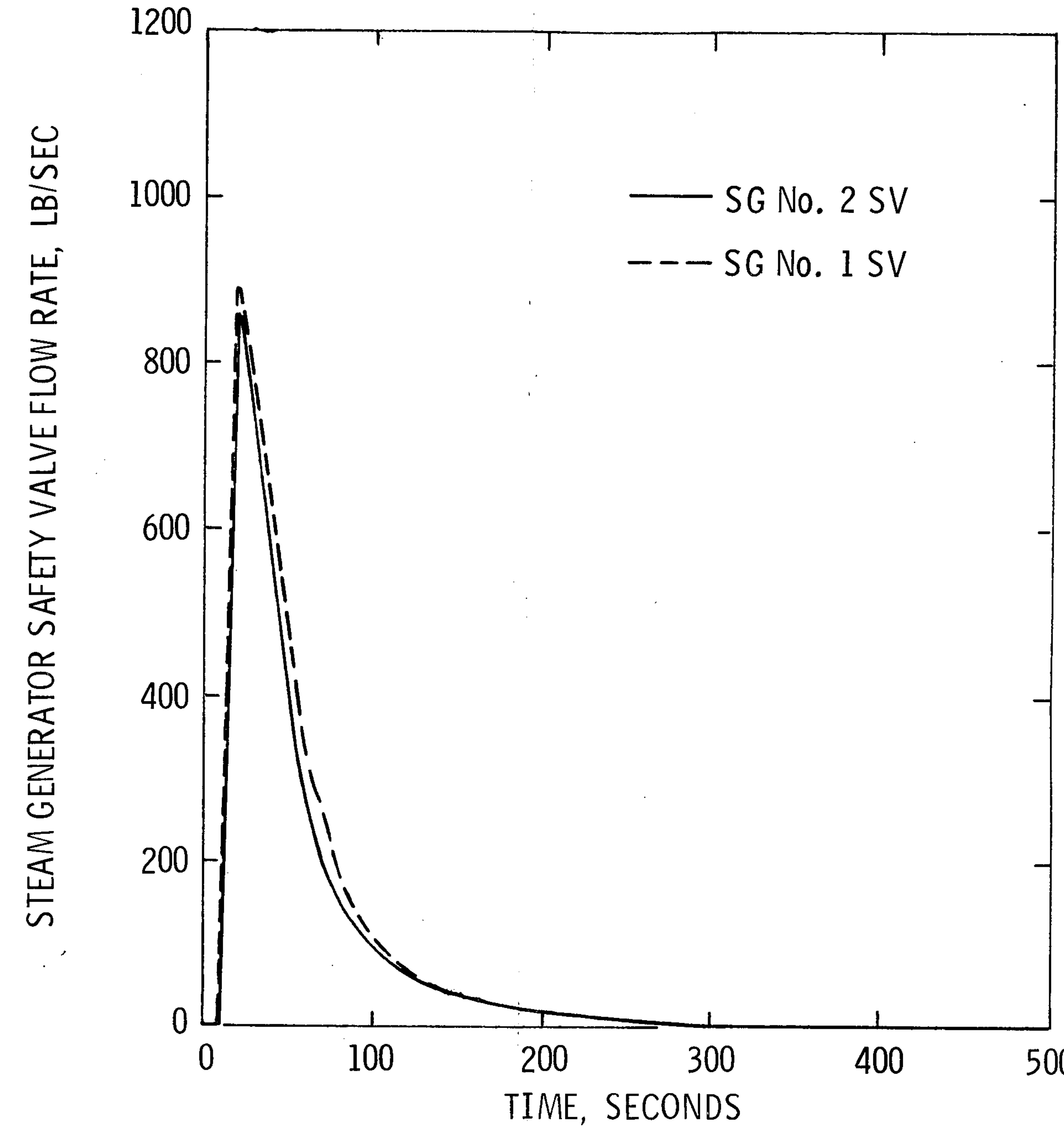


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM WITH A CONCURRENT FAILURE OF AN ACTIVE COMPONENT PRIMARY SAFETY VALVE FLOWRATE vs. TIME

Figure 15.2-27

10079-11D-1009
10OC6

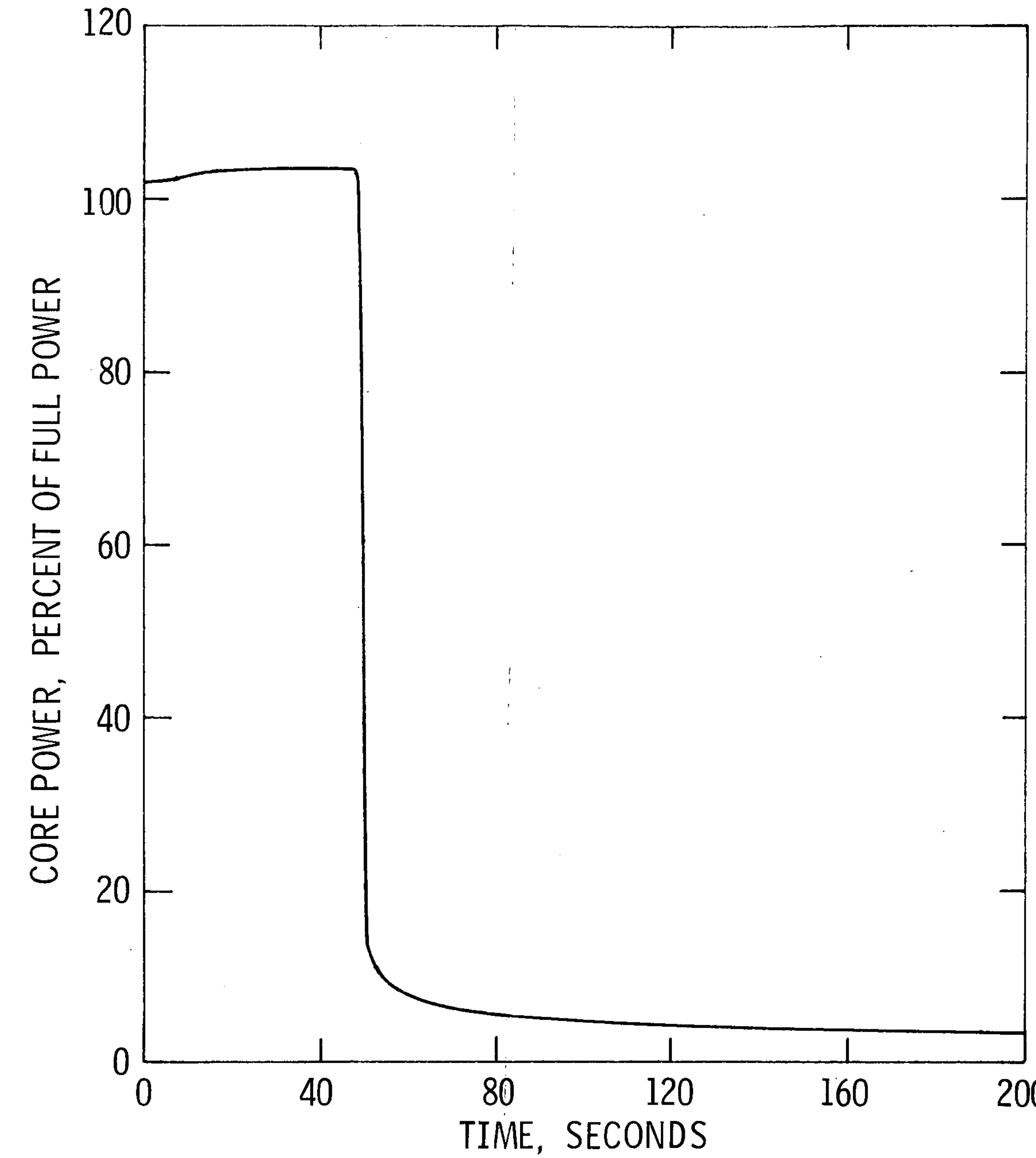


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF CONDENSER VACUUM
STEAM GENERATOR SAFETY VALVE
FLOWRATE vs. TIME

Figure 15.2-28

10079-11D-1010
10OC6

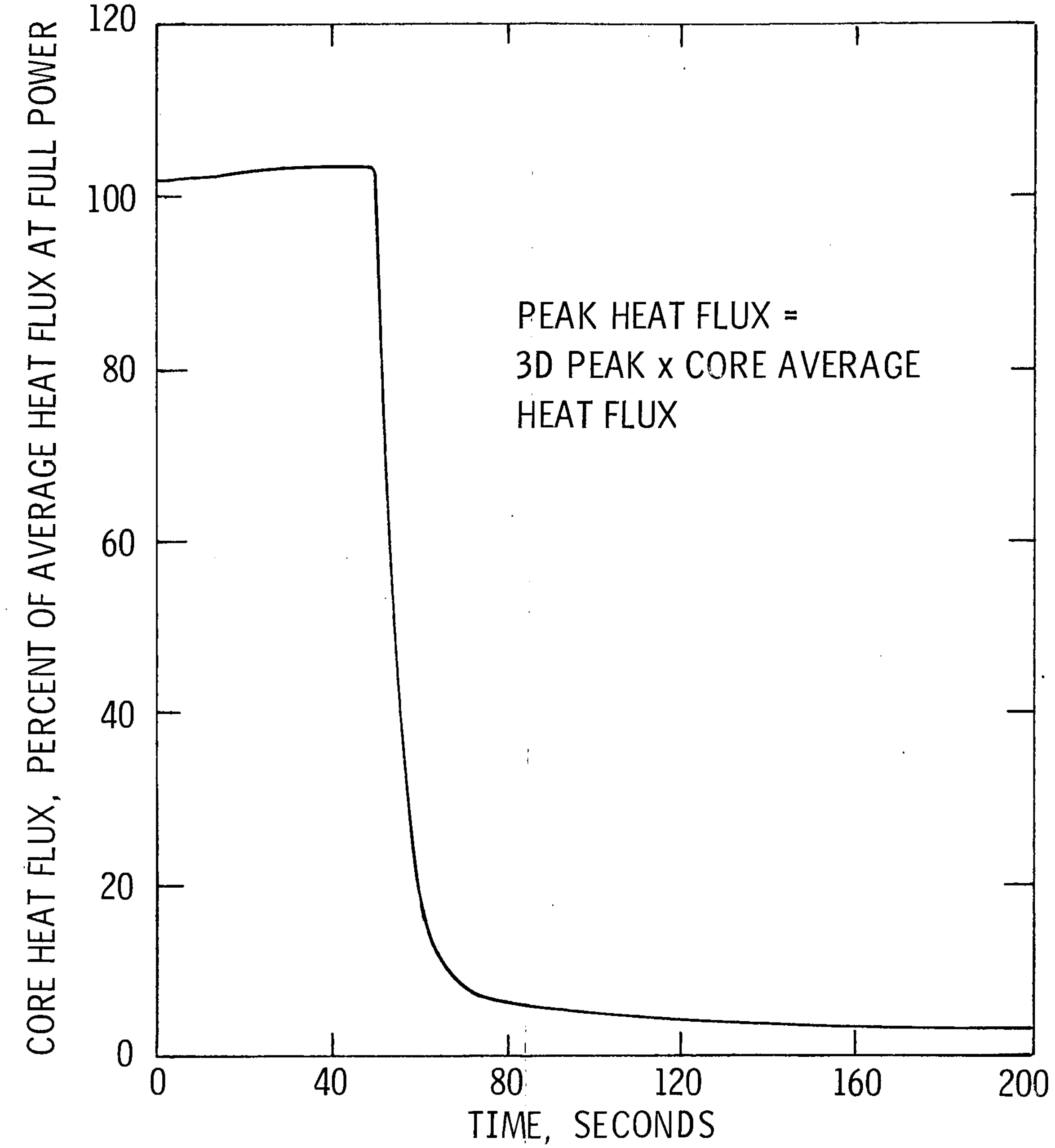


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
CORE POWER vs. TIME

Figure 15.2-29

10079-11D-1011
10OC6

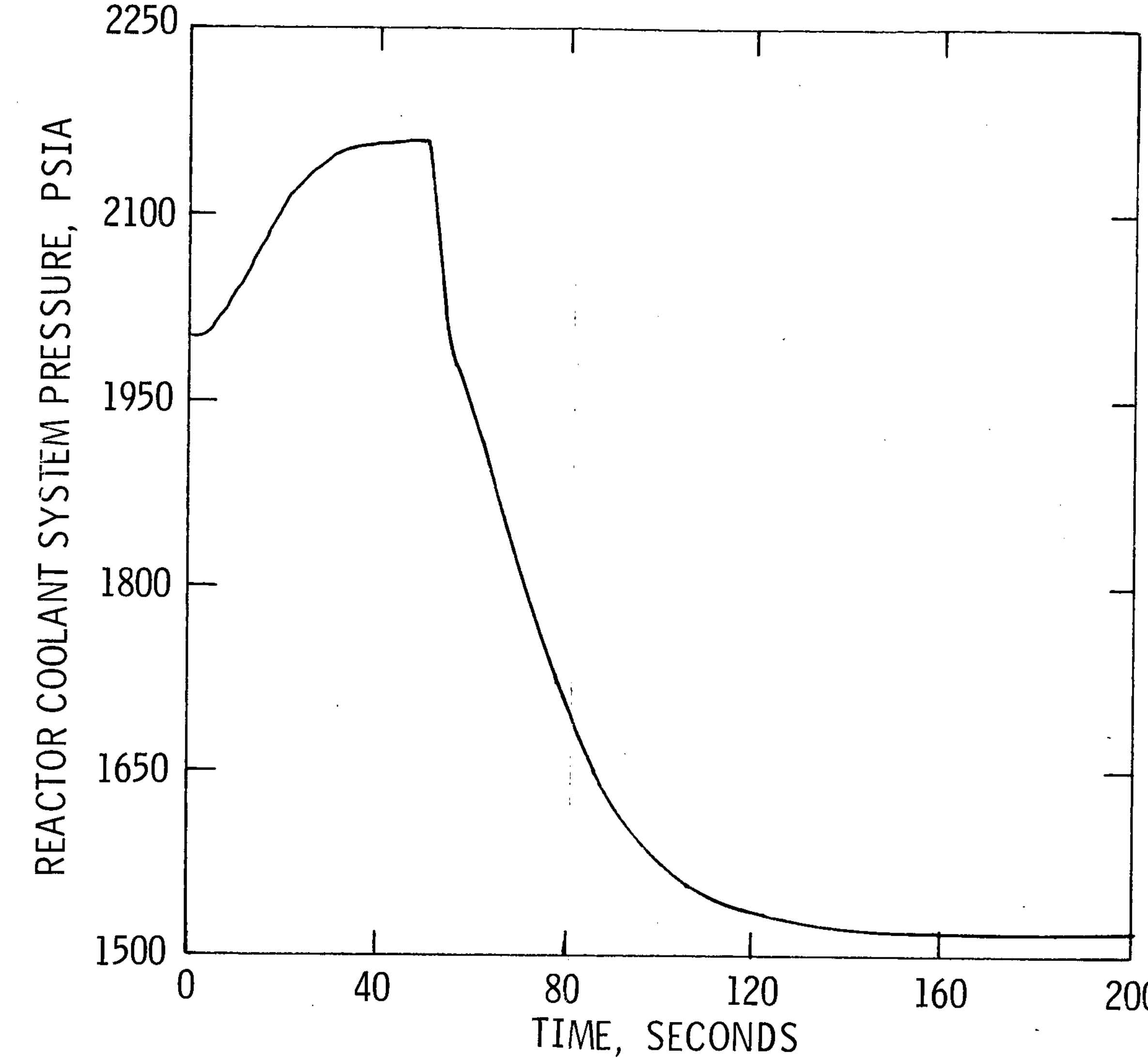


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
CORE AVERAGE HEAT FLUX vs. TIME

Figure 15.2-30

10079-11D-1012
10OC6

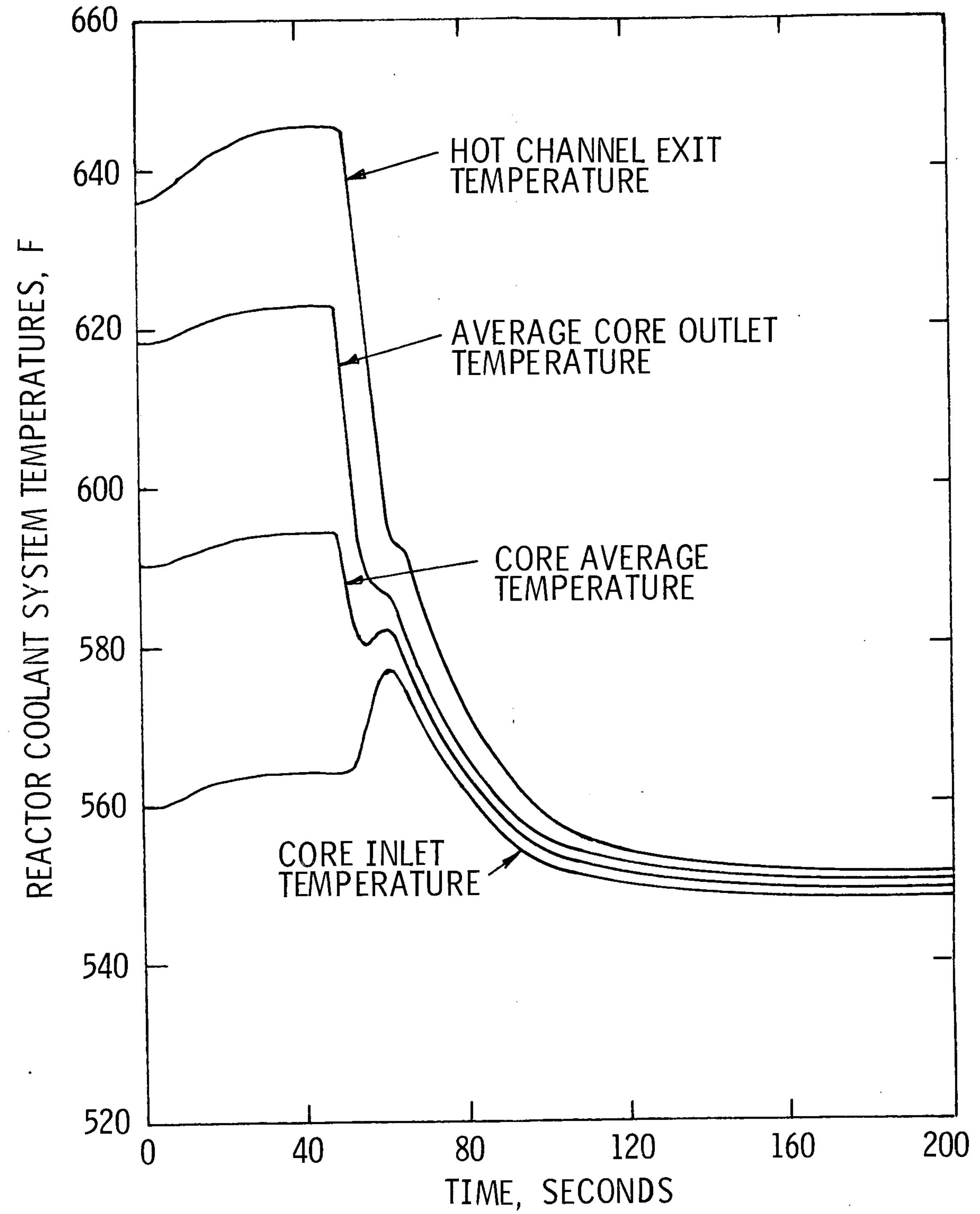


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
REACTOR SYSTEM COOLANT
PRESSURE vs. TIME

Figure 15.2-31

10079-11D-1013
10OC6

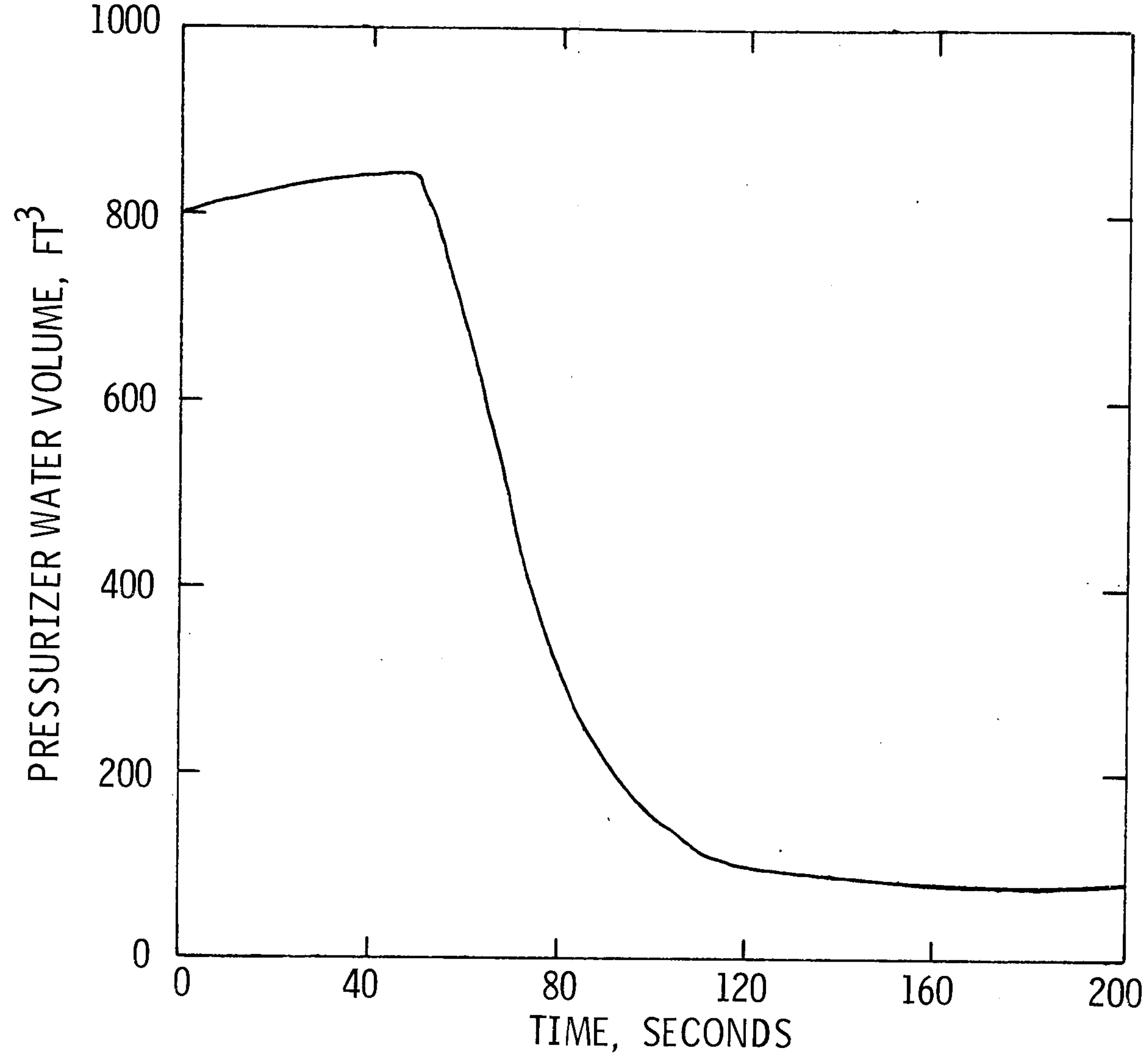


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
REACTOR SYSTEM COOLANT
TEMPERATURES vs. TIME

Figure 15.2-32

10079-11D-1014
10OC6

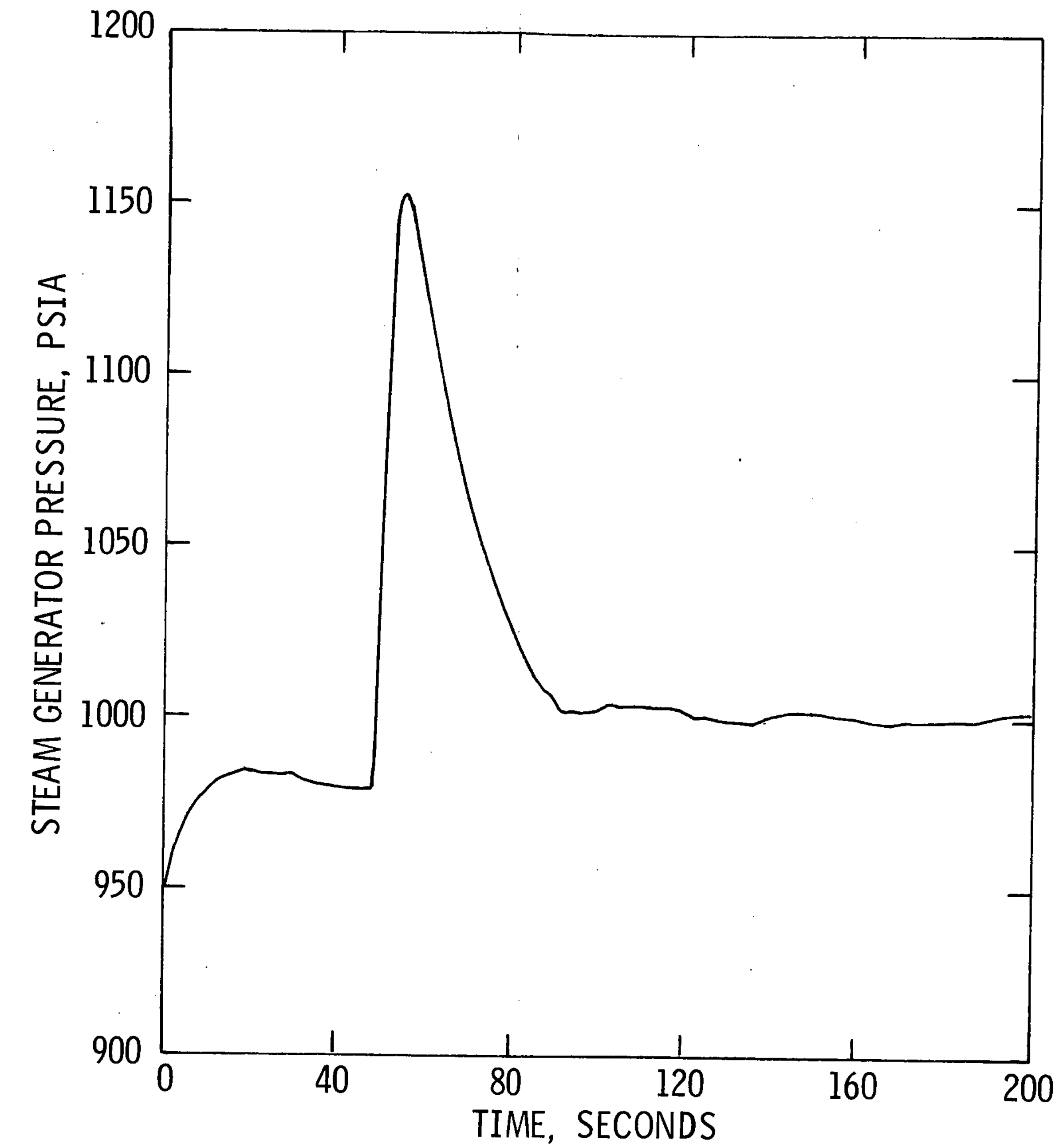


SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
PRESSURIZER WATER
VOLUME vs. TIME

Figure 15.2-33

10079-11D-1015
10OC6

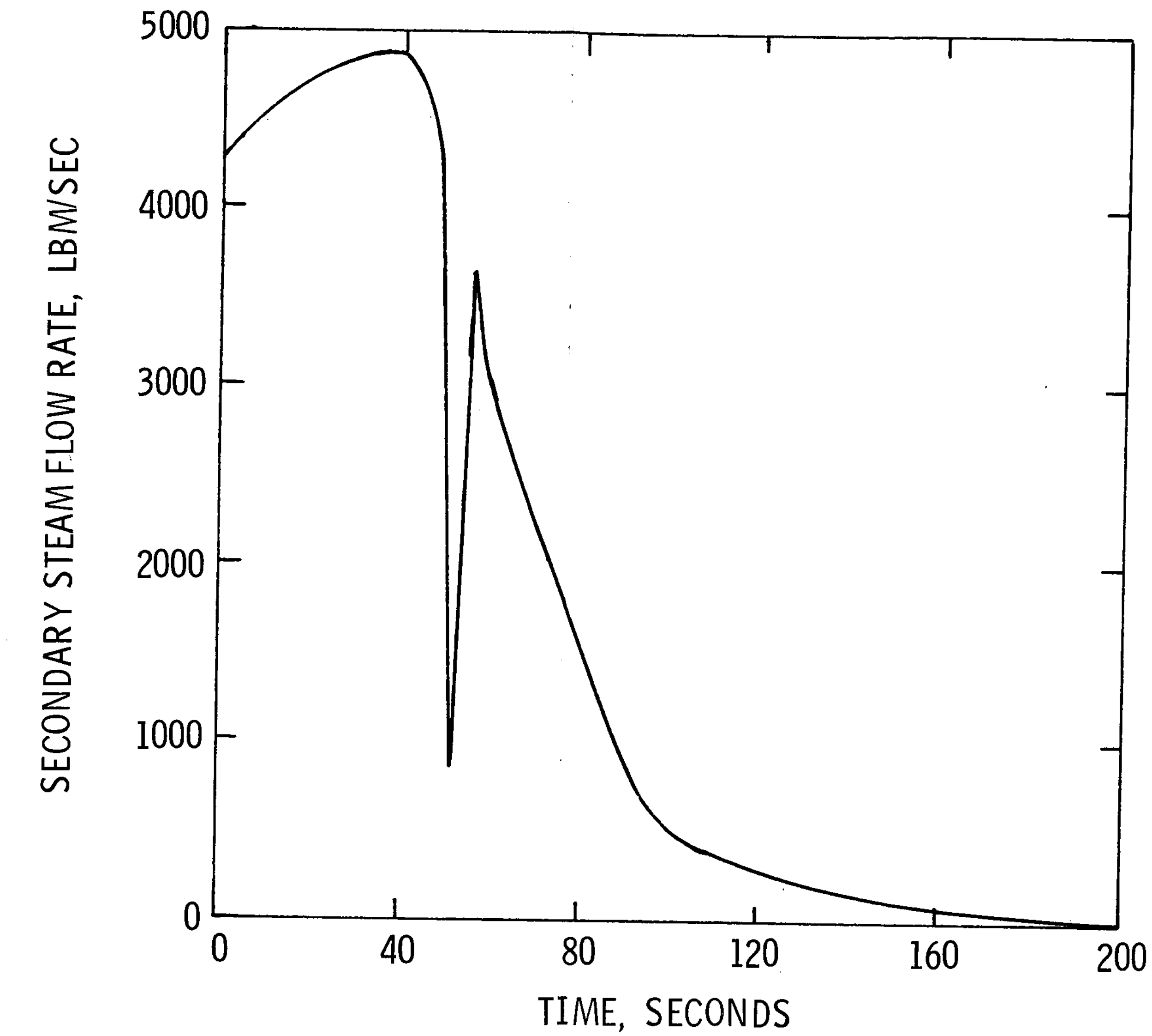


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
STEAM GENERATOR PRESSURE vs. TIME

Figure 15.2-34

10079-11D-1016
10OC6

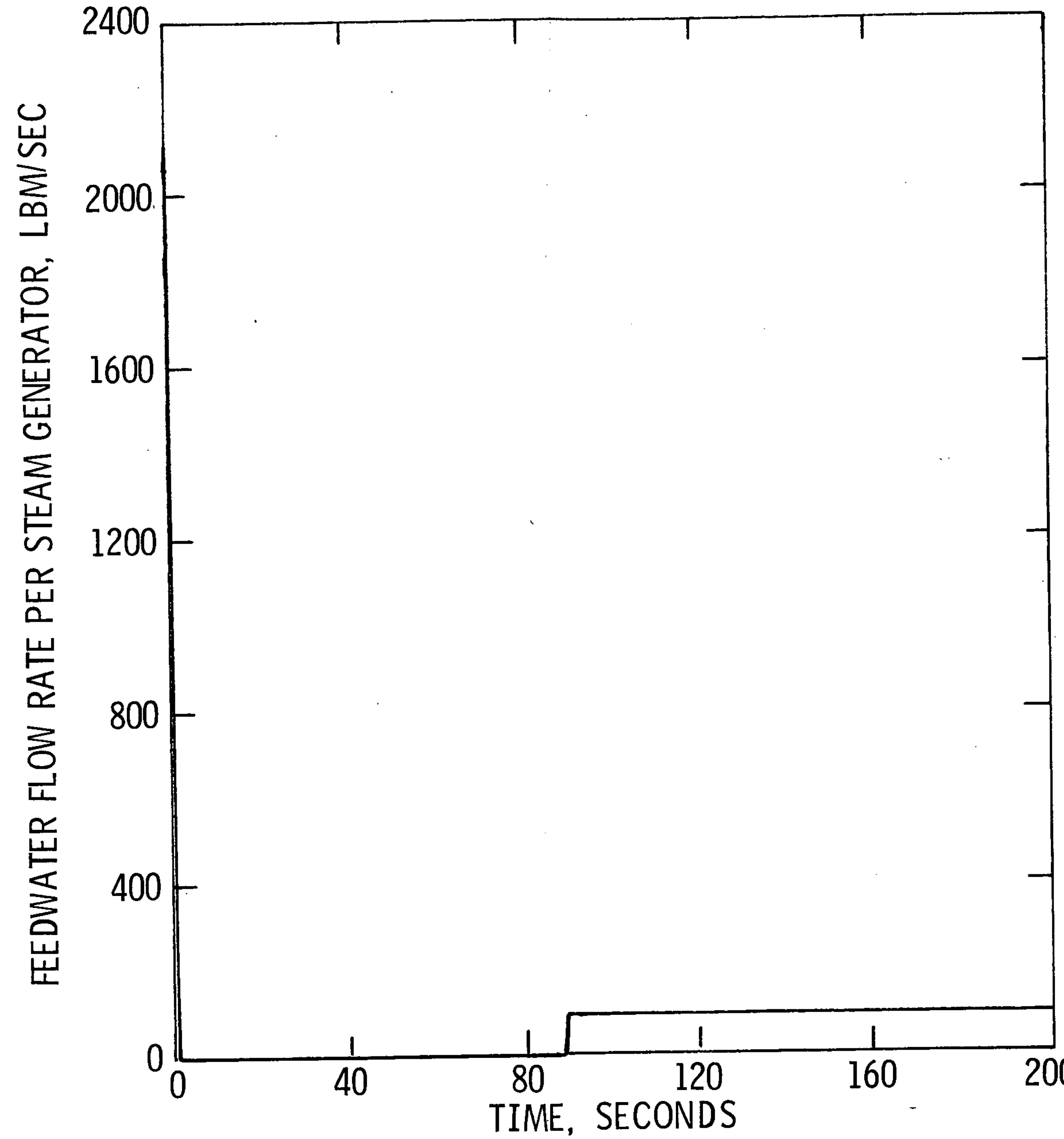


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
SECONDARY STEAM FLOWRATE vs. TIME

Figure 15.2-35

10079-11D-1017
10OC6

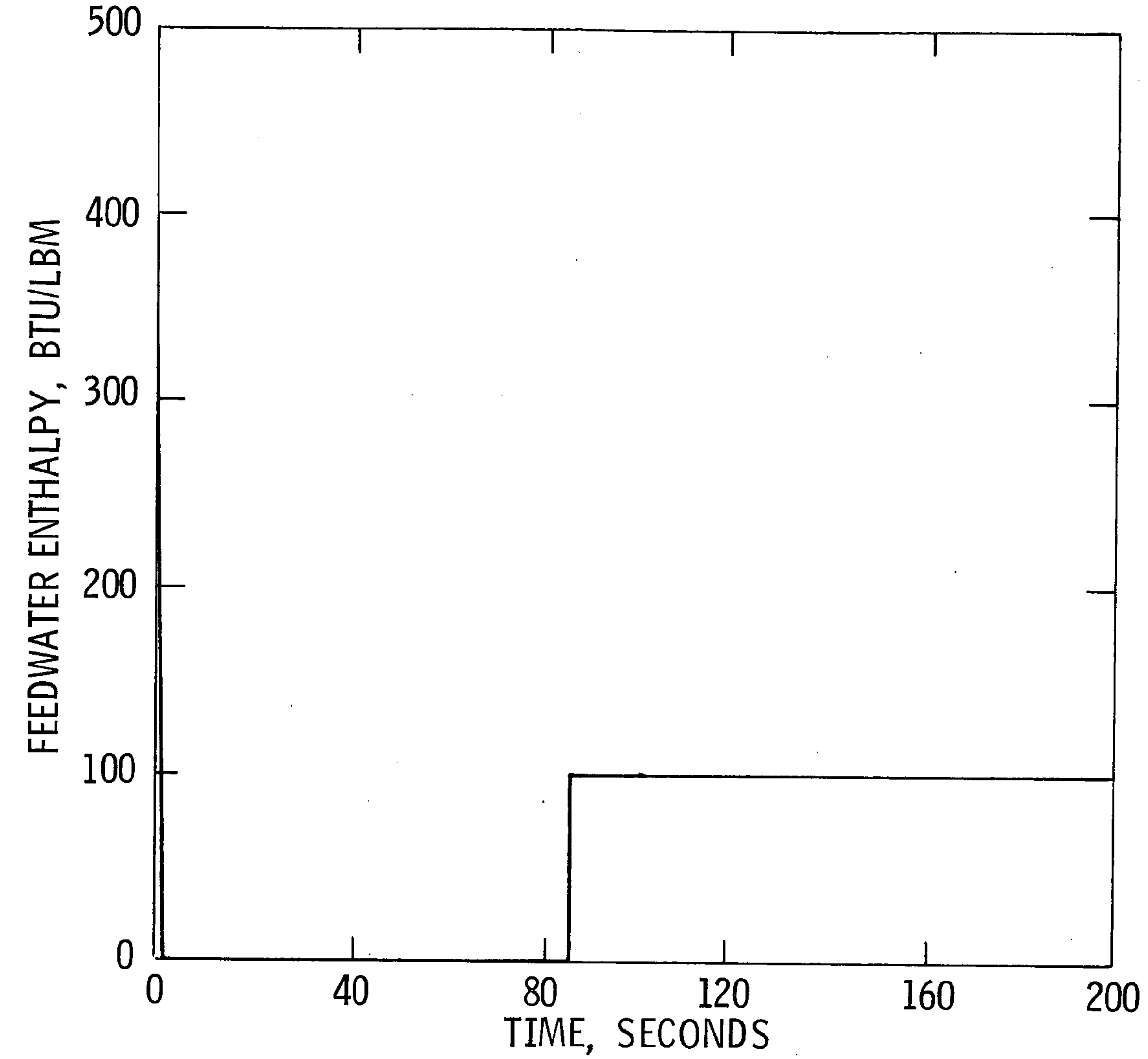


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
FEEDWATER FLOWRATE PER
STEAM GENERATOR vs. TIME

Figure 15.2-36

10079-11D-1018
10OC6

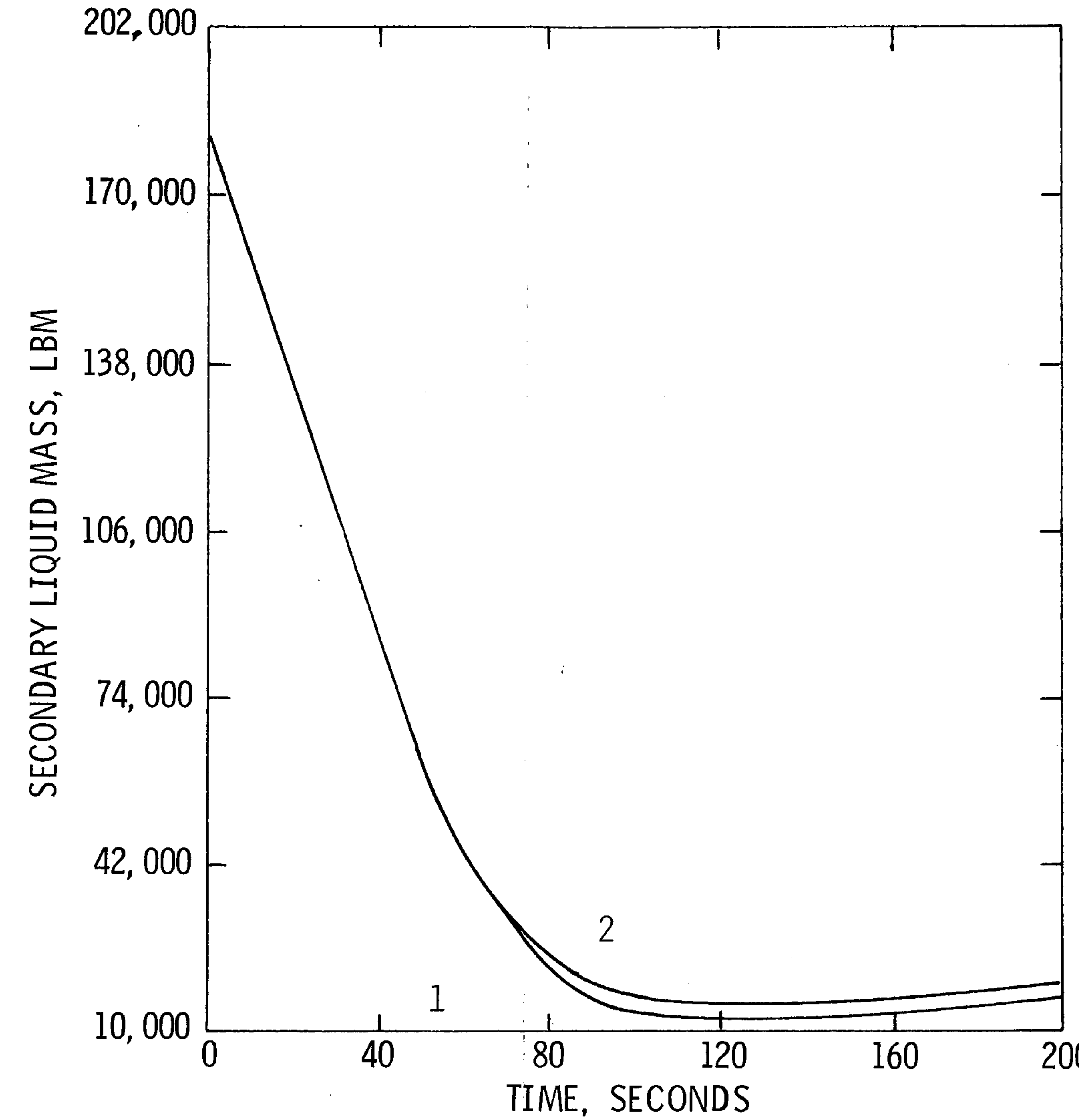


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
FEEDWATER ENTHALPY vs. TIME

Figure 15.2-37

10079-11D-1019
10OC6

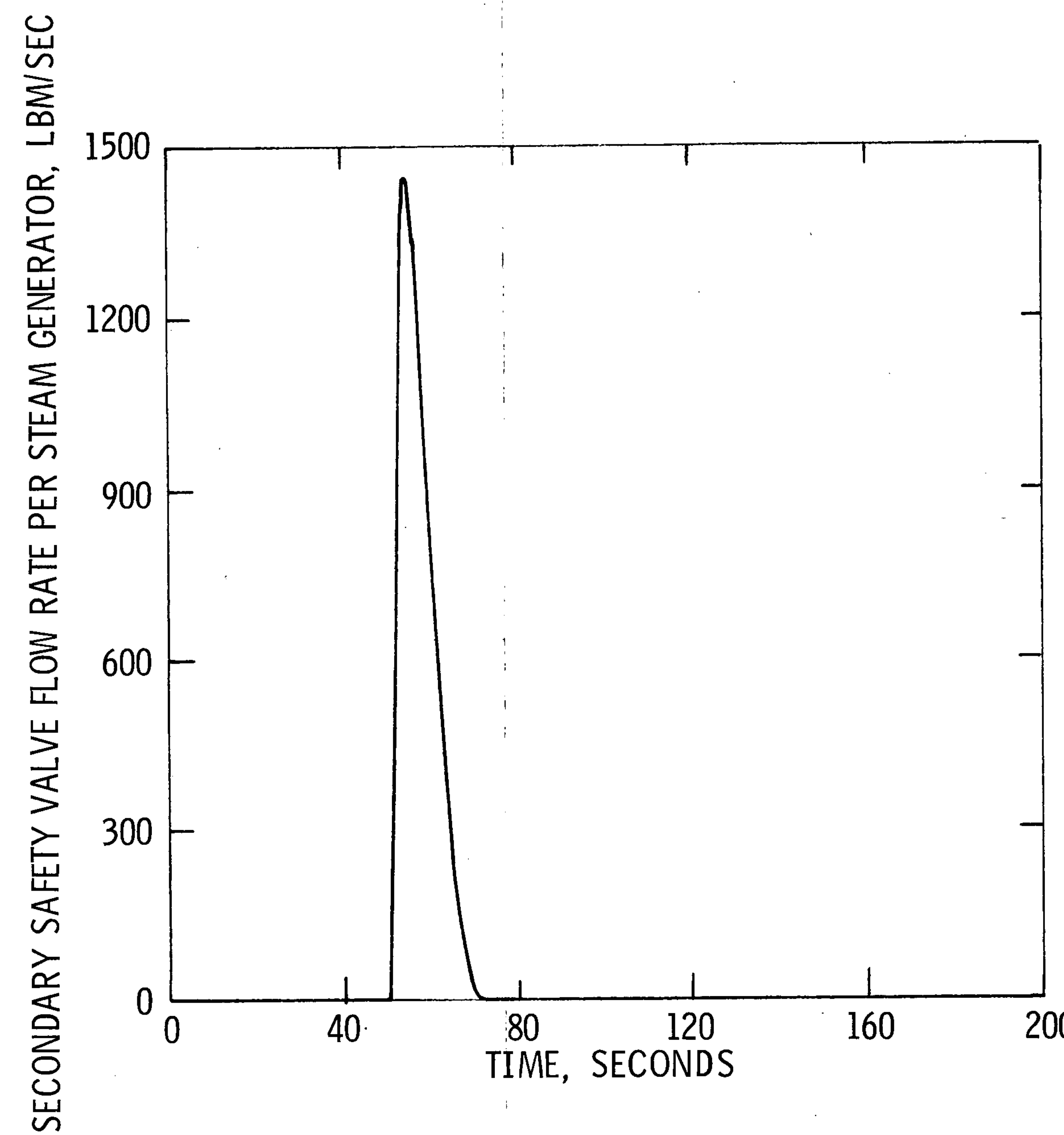


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
SECONDARY LIQUID MASS vs. TIME

Figure 15.2-38

10079-11D-1020
10OC6

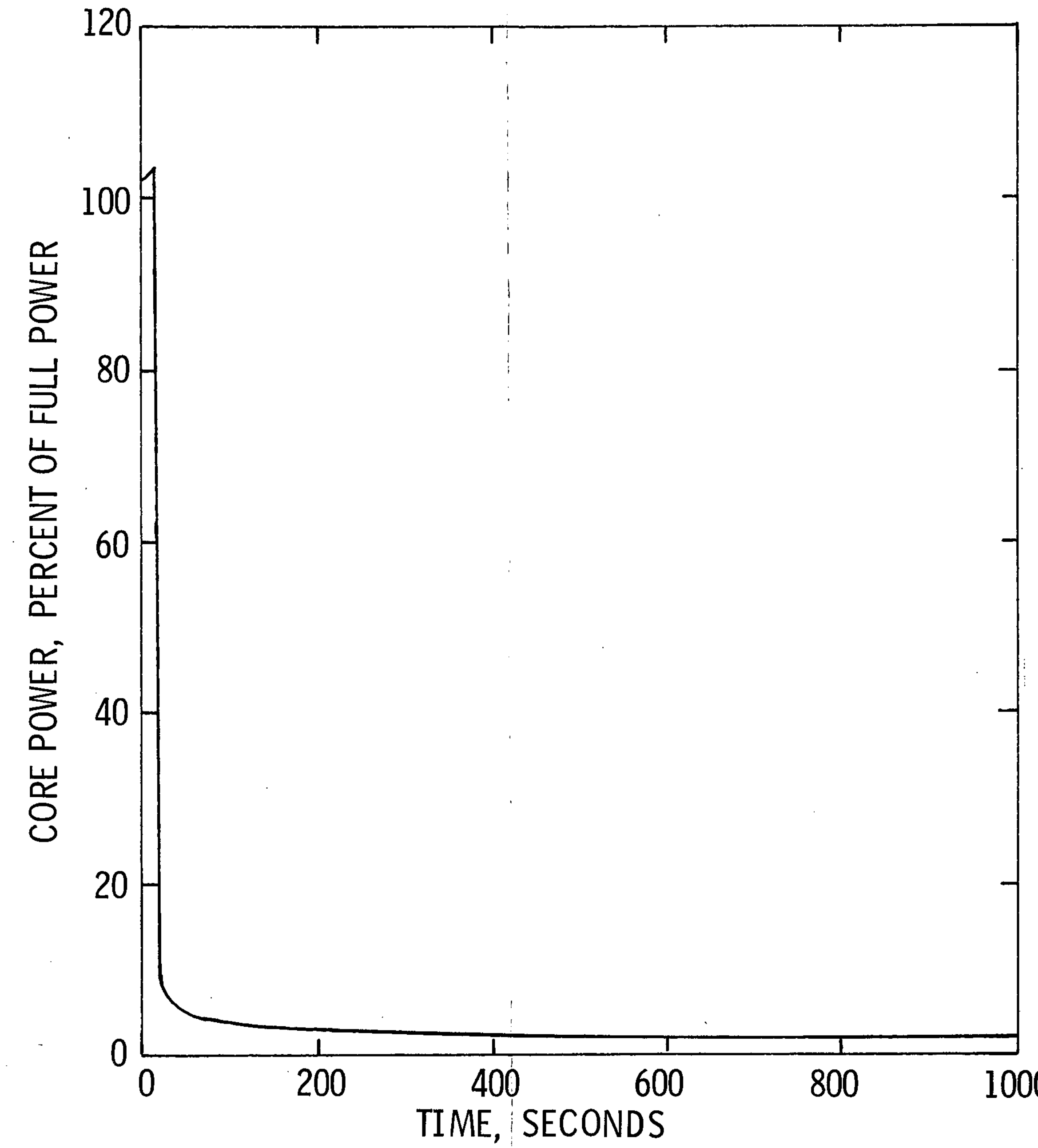


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
SECONDARY SAFETY VALVE FLOWRATE
PER STEAM GENERATOR
vs. TIME

Figure 15.2-39

10079-11D-1021
10OC6

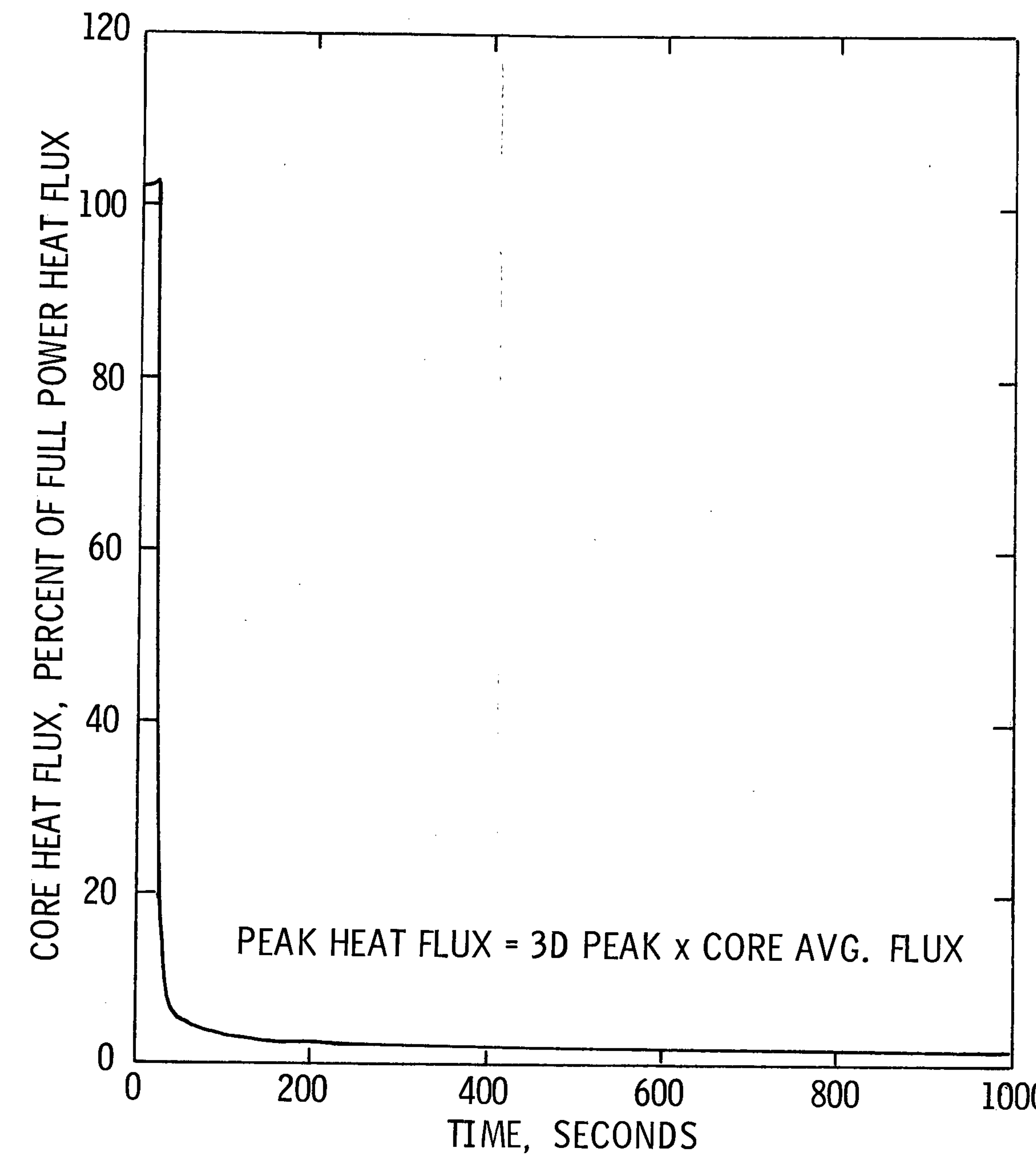


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
CORE POWER vs. TIME

Figure 15.2-40

10079-11D-1022
10OC6

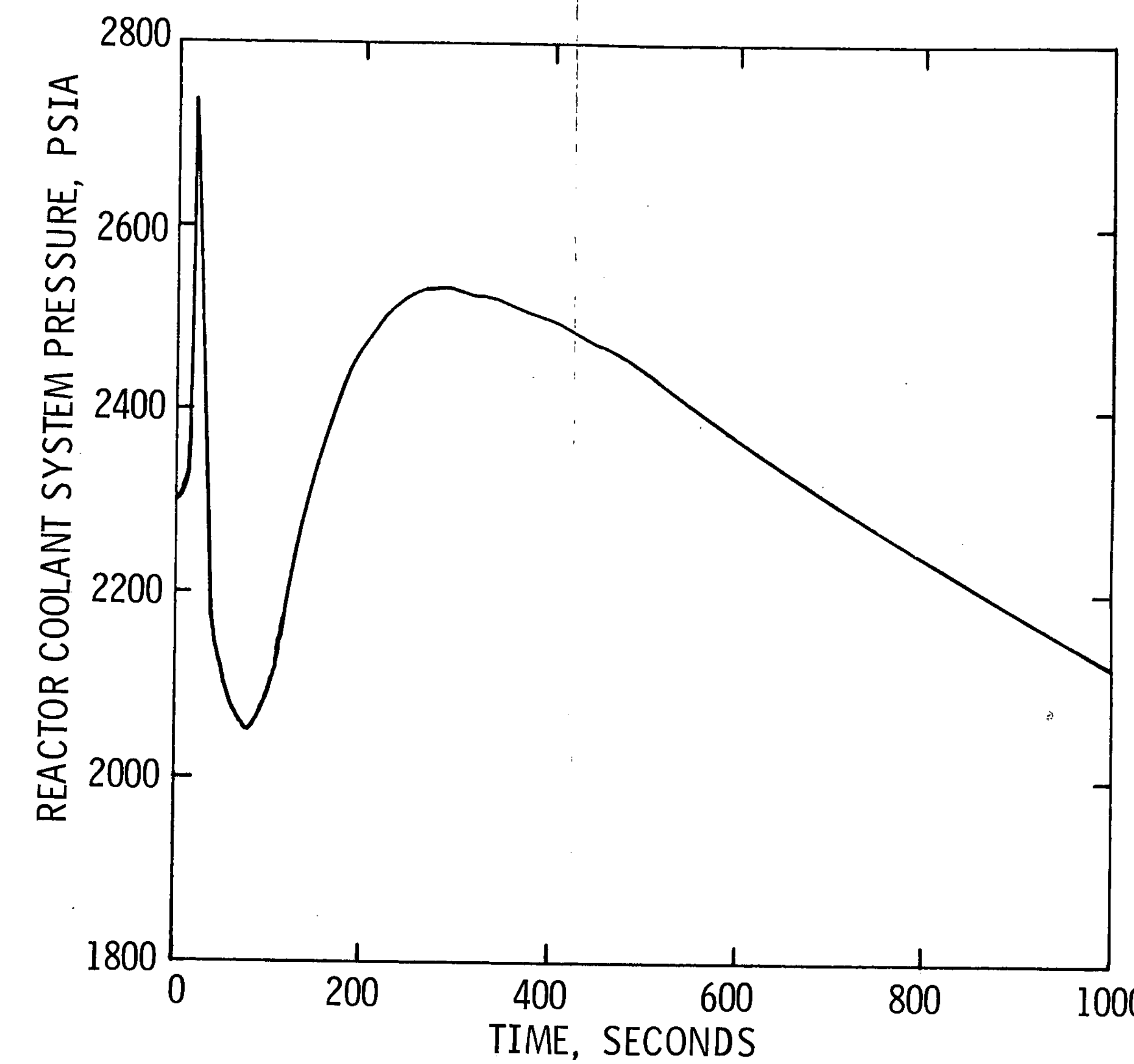


10079-11D-1023
10OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
CORE HEAT FLUX vs. TIME

Figure 15.2-41

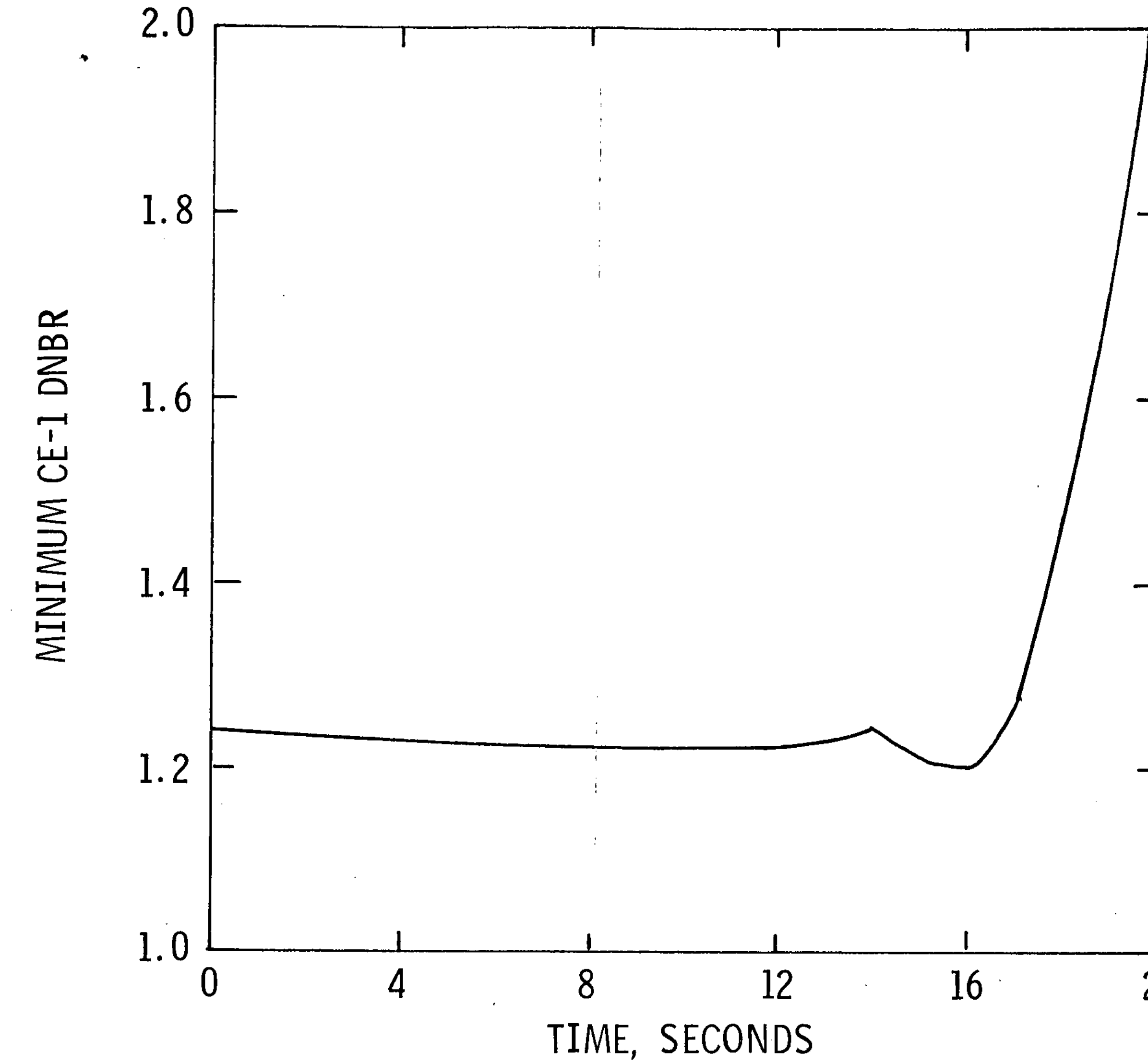


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
REACTOR COOLANT SYSTEM
PRESSURE vs. TIME

Figure 15.2-42

10079-11D-1024
10OC6

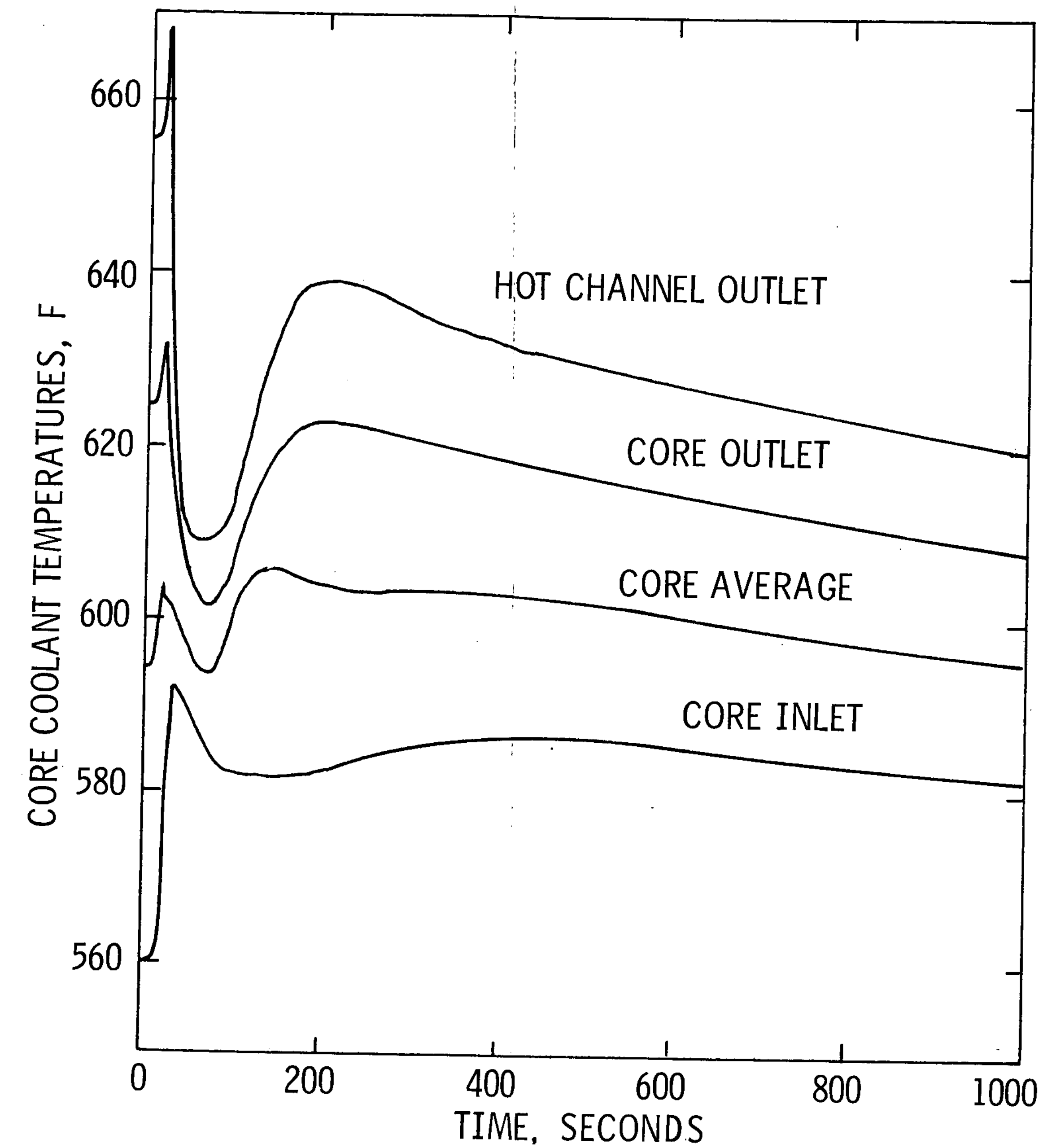


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
MINIMUM DNBR vs. TIME

Figure 15.2-43

10079-11D-1025
10OC6

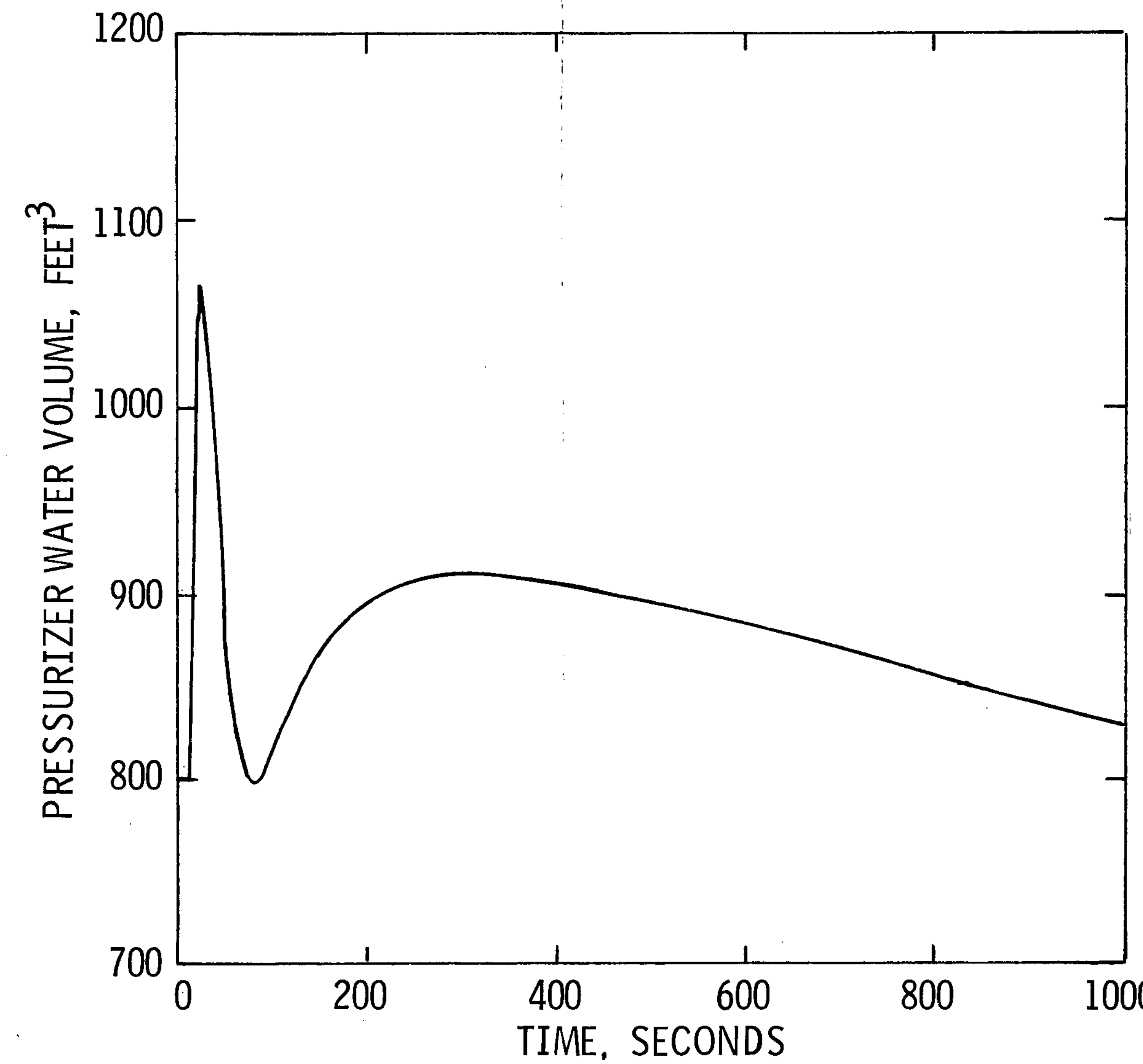


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
CORE COOLANT TEMPERATURES vs. TIME

Figure 15.2-44

10079-11D-1026
100C6

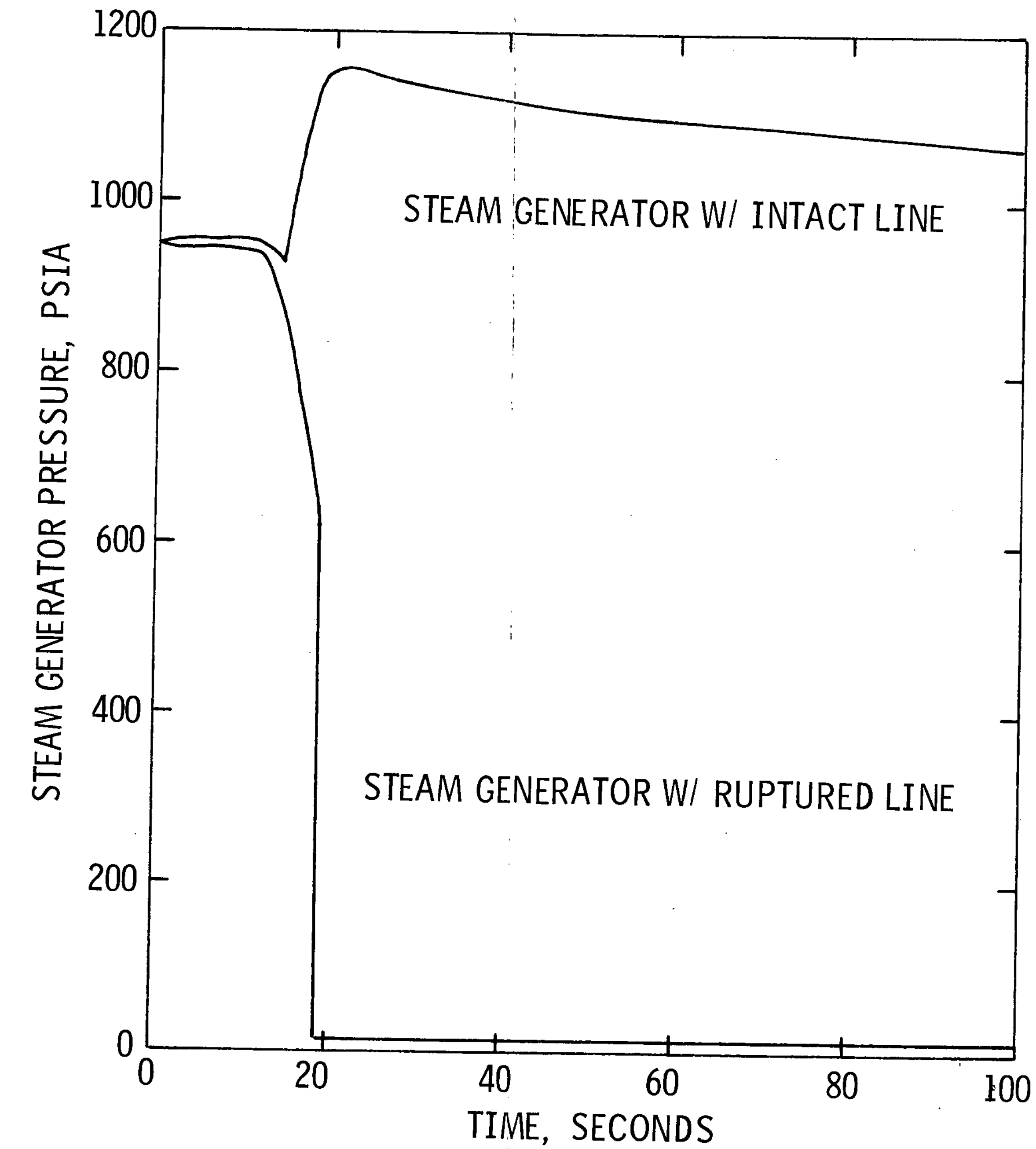


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
PRESSURIZER WATER VOLUME vs. TIME

Figure 15.2-45

10079-11D-1027
10OC6

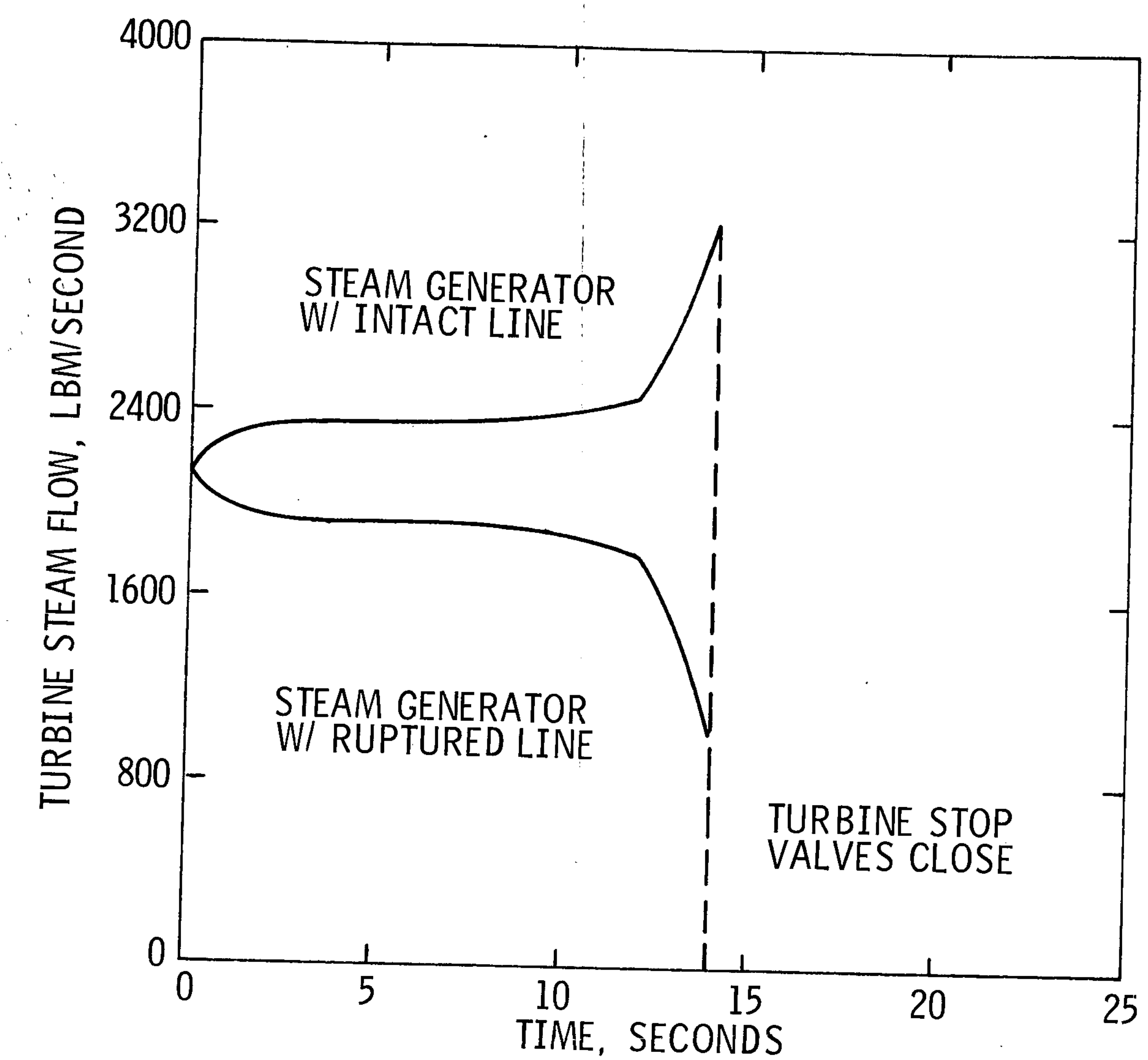


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
STEAM GENERATOR PRESSURE vs. TIME

Figure 15.2-46

10079-11D-1028
100C6

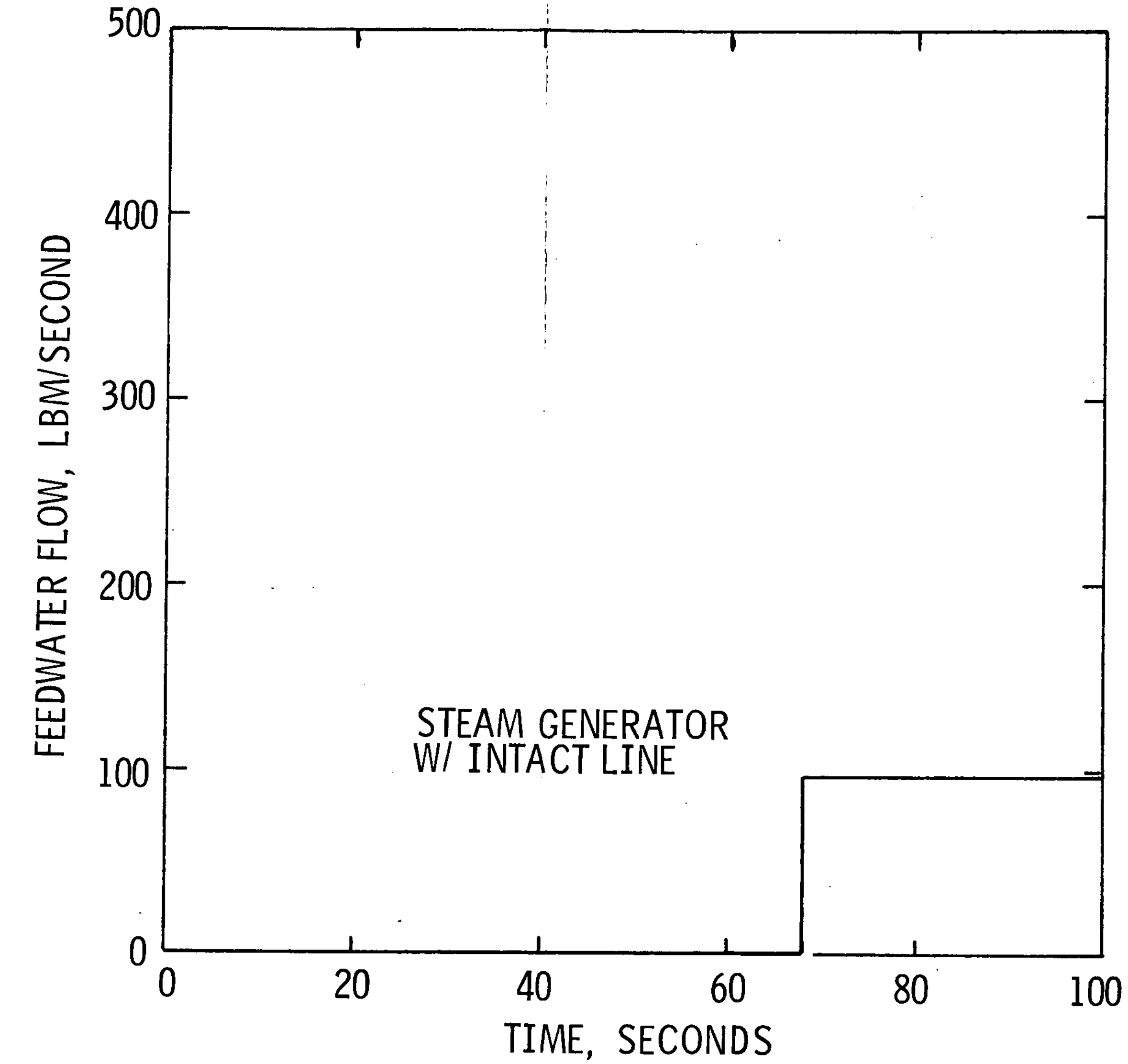


10079-11D-1029
10OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
TURBINE STEAM FLOW vs. TIME

Figure 15.2-47

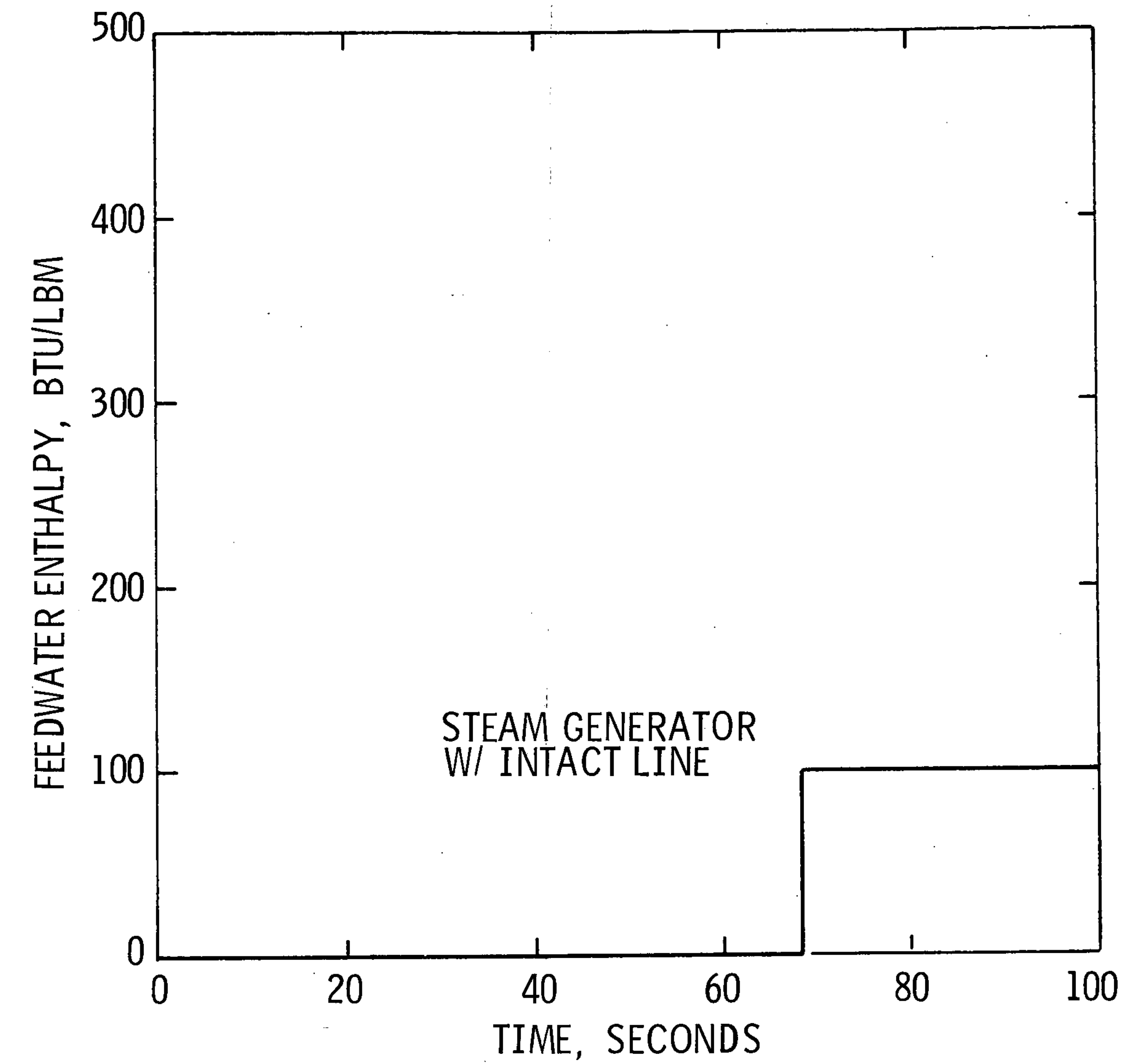


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
FEEDWATER FLOW vs. TIME

Figure 15.2-48

10079-11D-1030
100C6

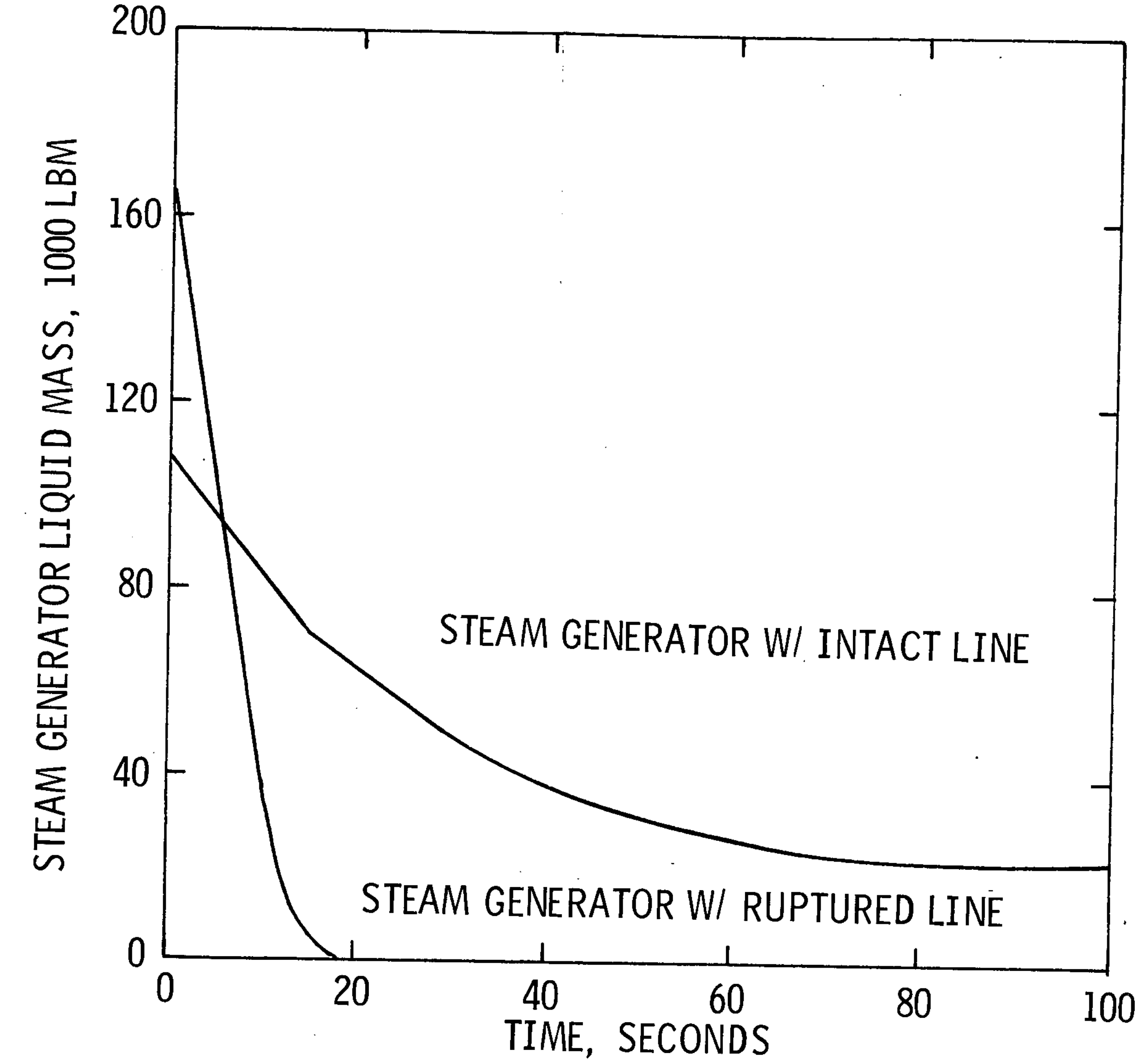


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
FEEDWATER ENTHALPY vs. TIME

Figure 15.2-49

10079-11D-1031
10OC6

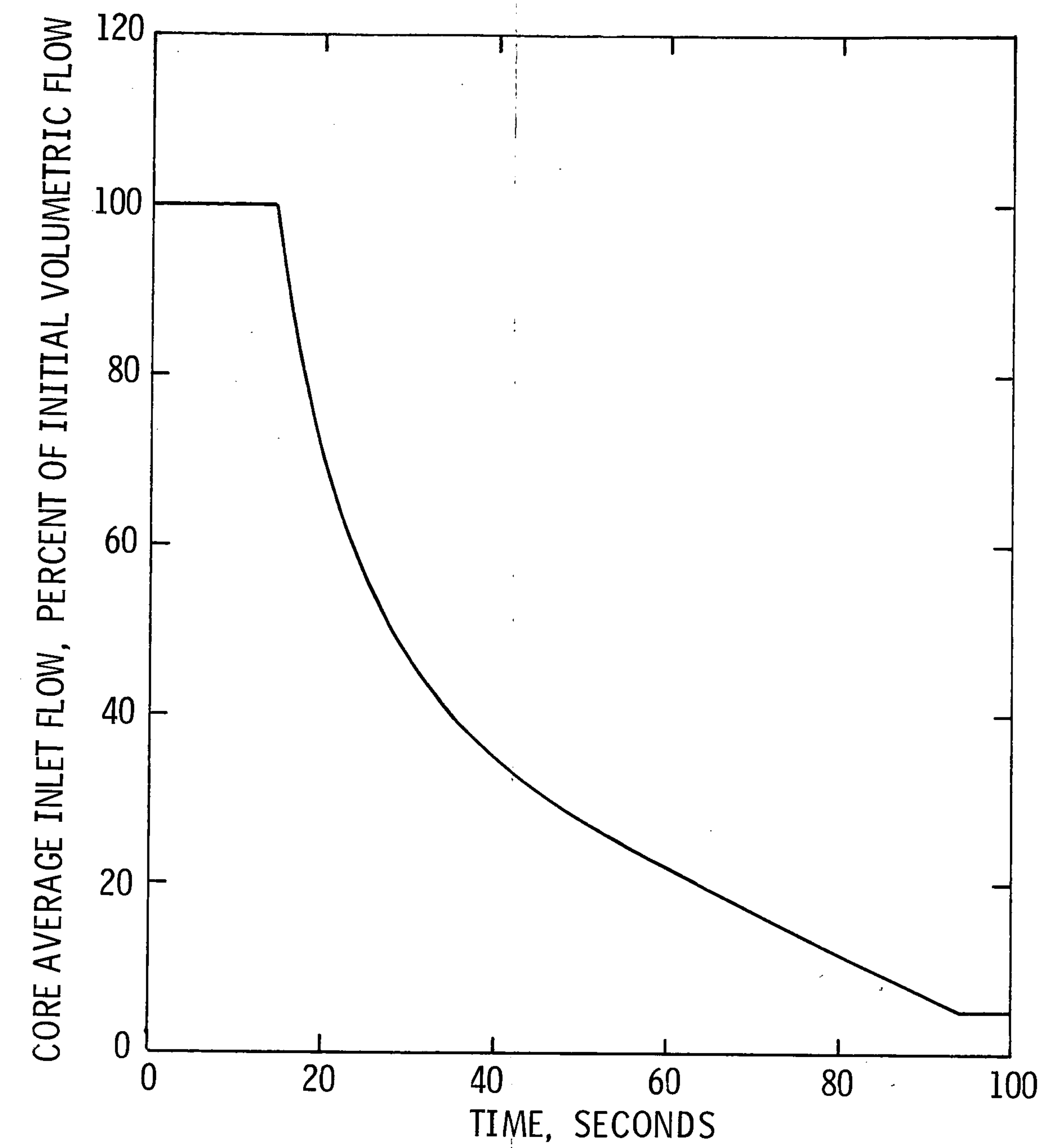


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
STEAM GENERATOR LIQUID
MASS vs. TIME

Figure 15.2-50

10079-11D-1032
100C6

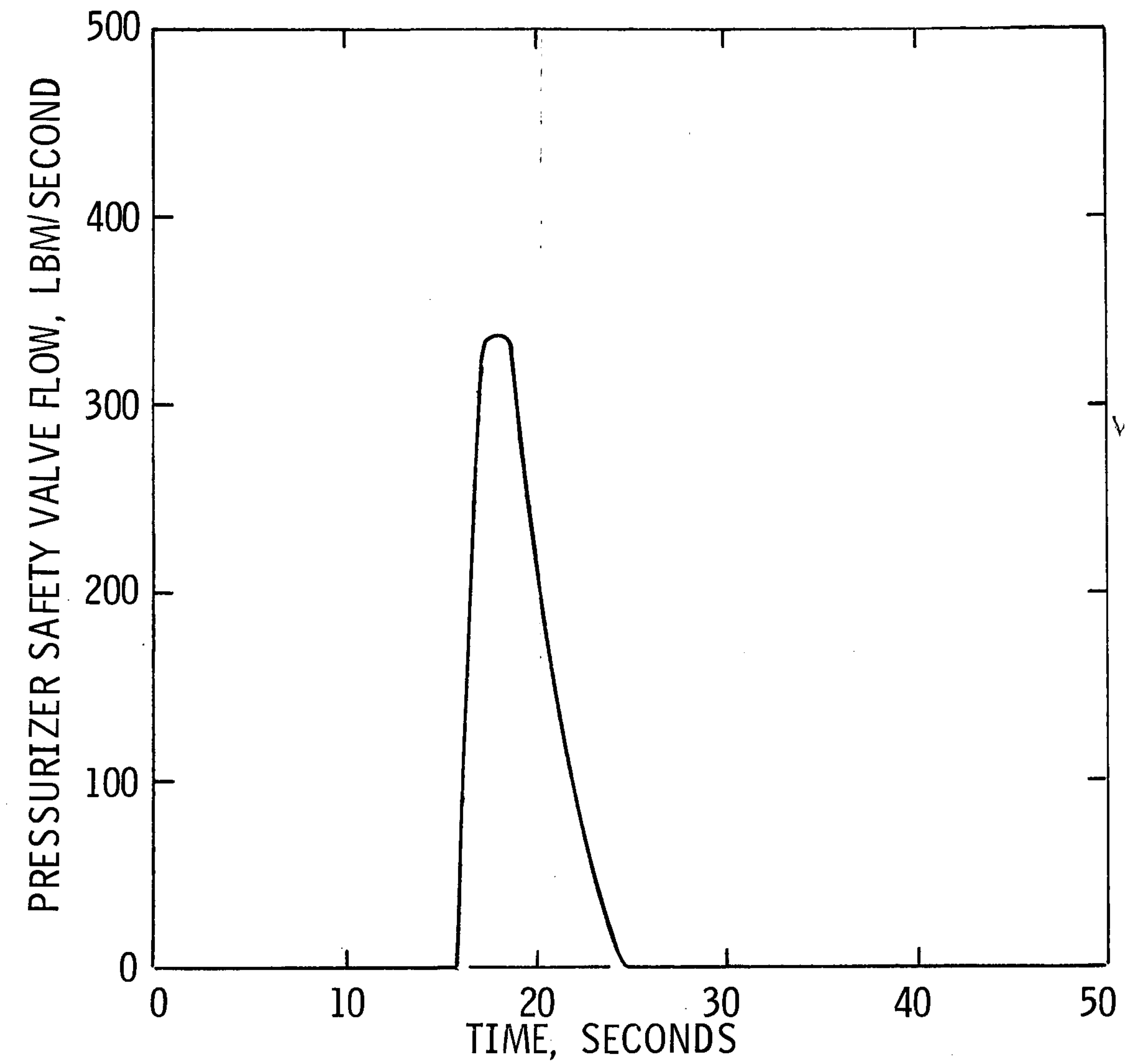


10079-11D-1033
10OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
CORE AVERAGE INLET FLOW vs. TIME

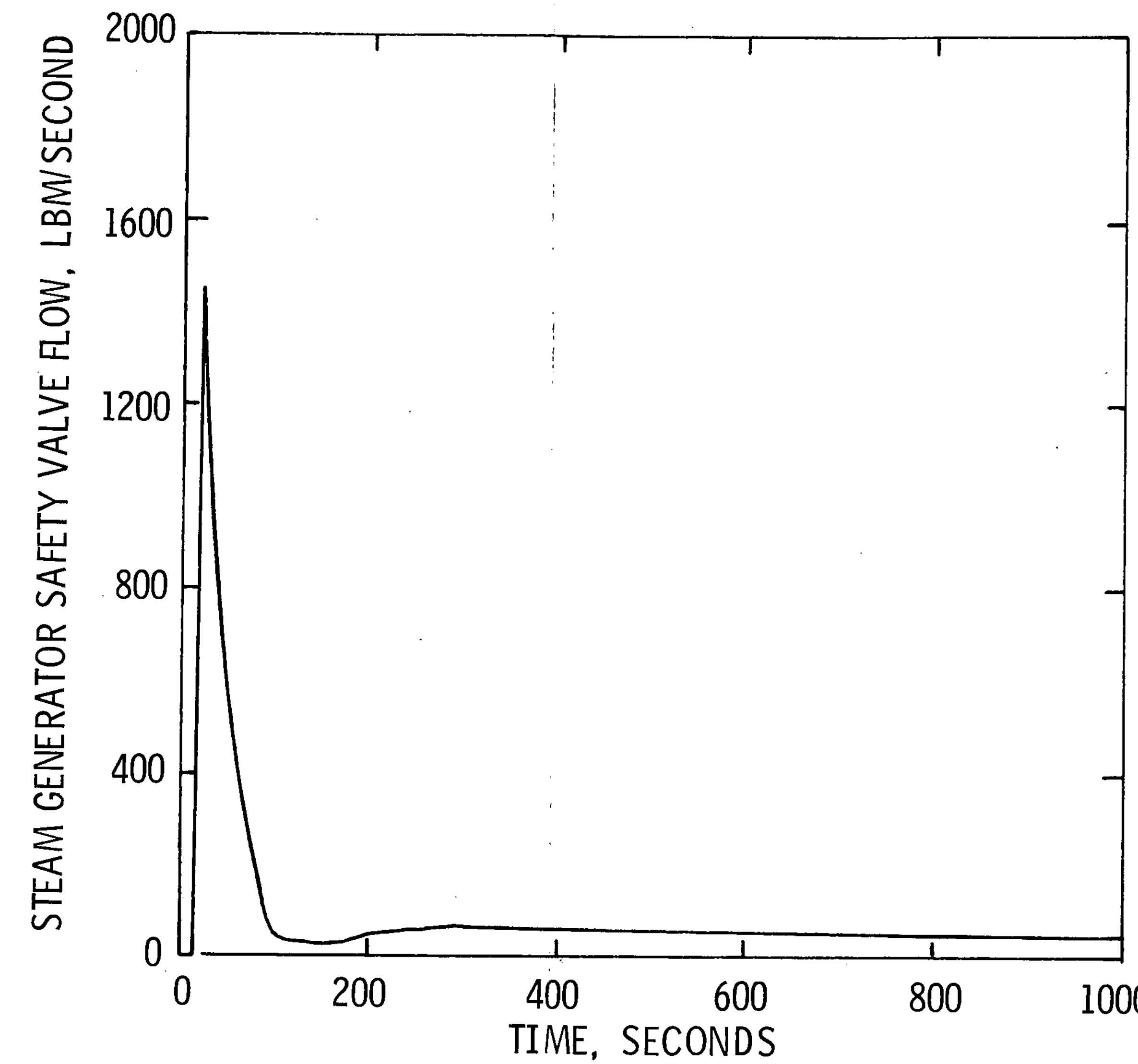
Figure 15.2-51



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
PRESSURIZER SAFETY VALVE
FLOW vs. TIME

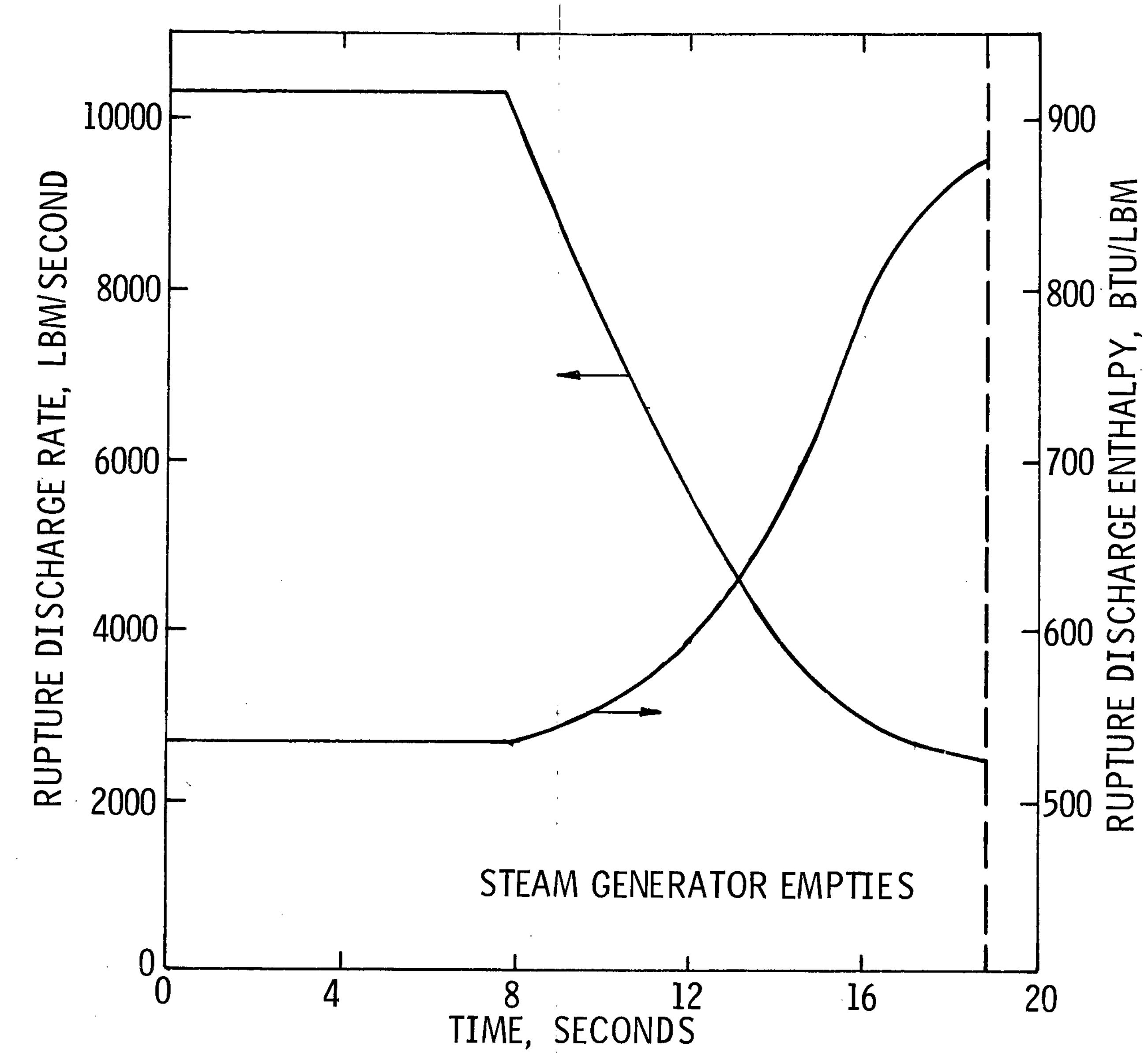
Figure 15.2-52



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
STEAM GENERATOR SAFETY VALVE
FLOW vs. TIME

Figure 15.2-53

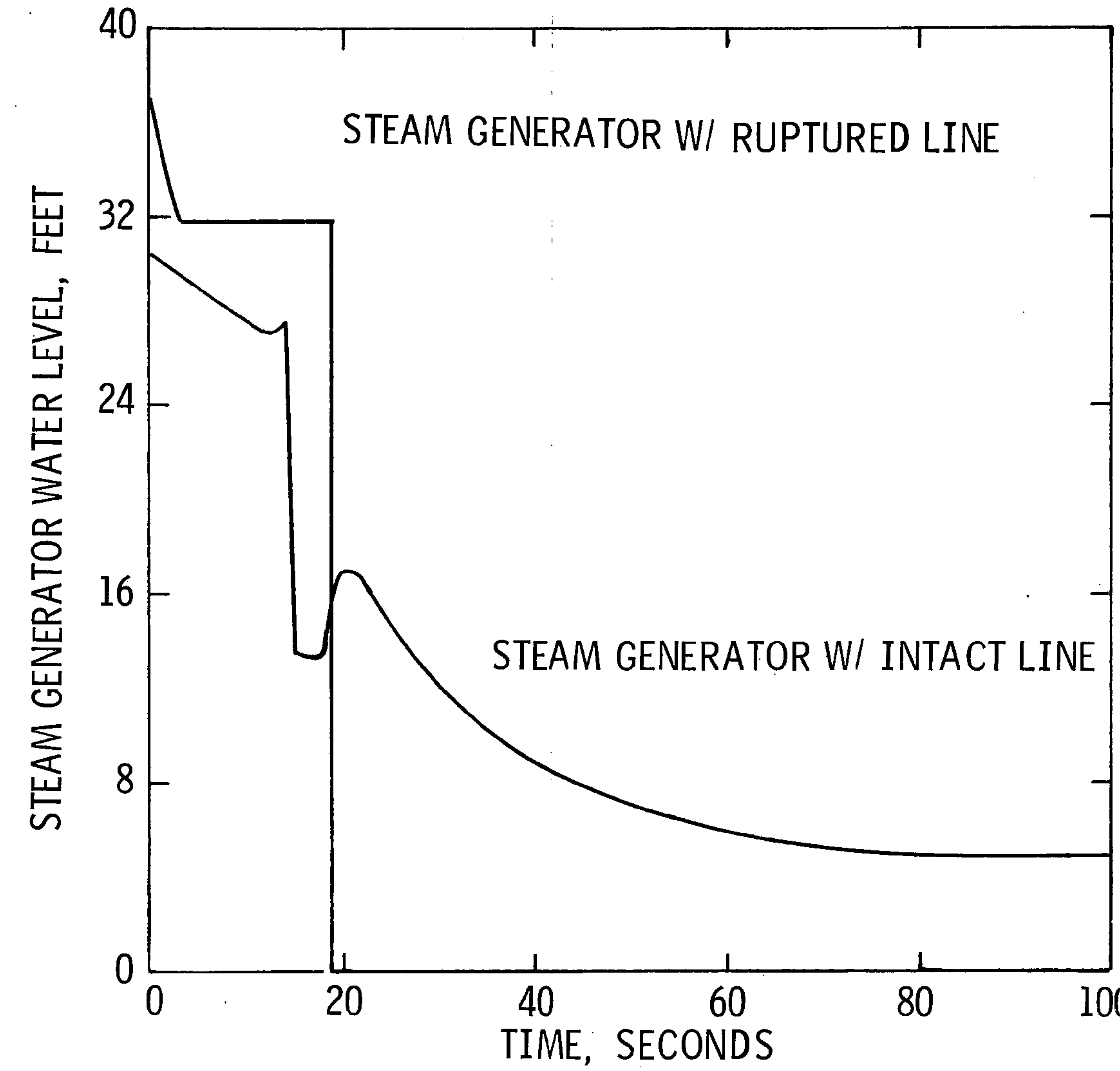


10079-11D-1036
10OC6

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
RUPTURE DISCHARGE RATE
AND ENTHALPY

Figure 15.2-54

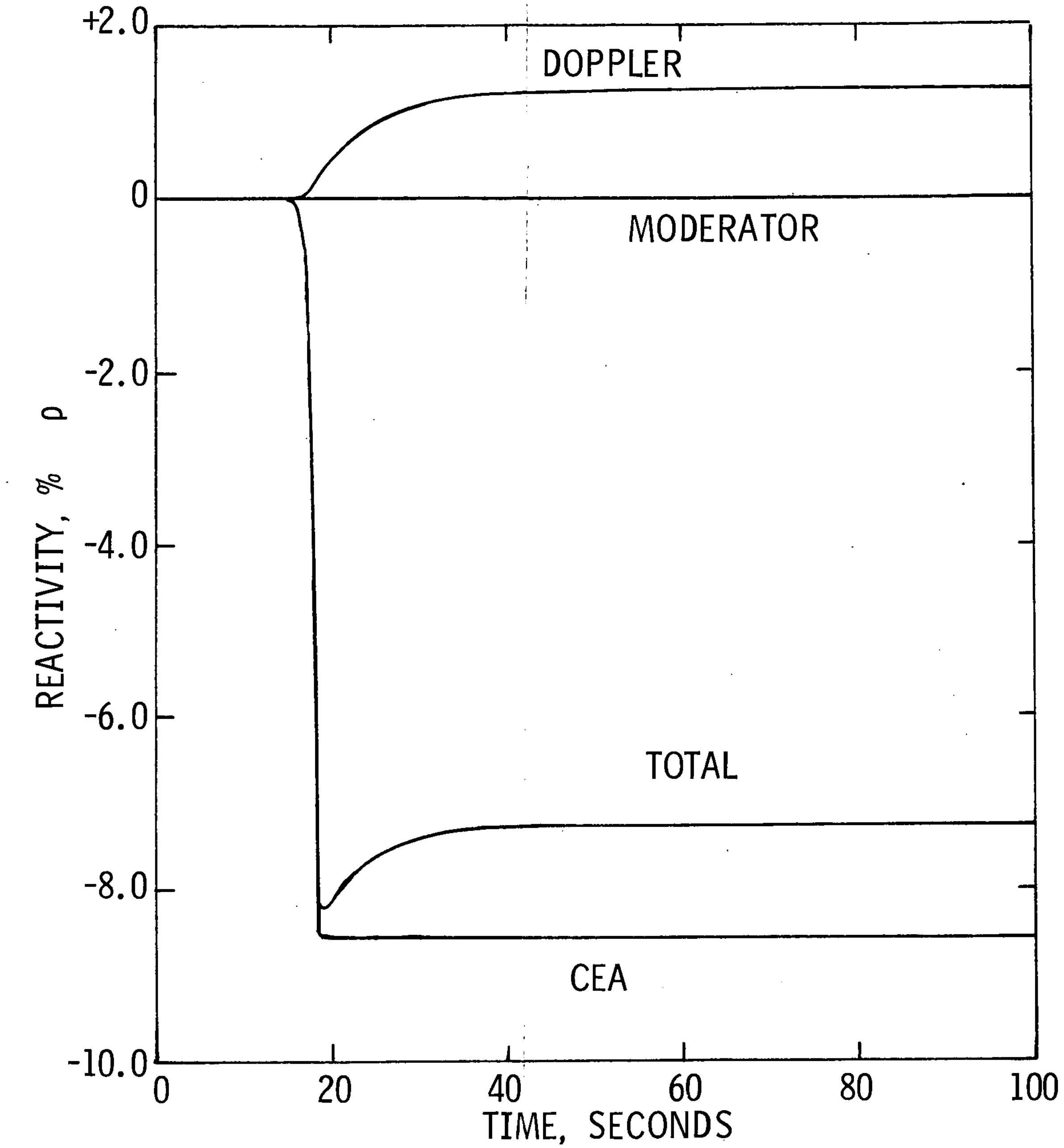


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
STEAM GENERATOR WATER
LEVEL vs. TIME

Figure 15.2-55

10079-11D-1037
10OC6

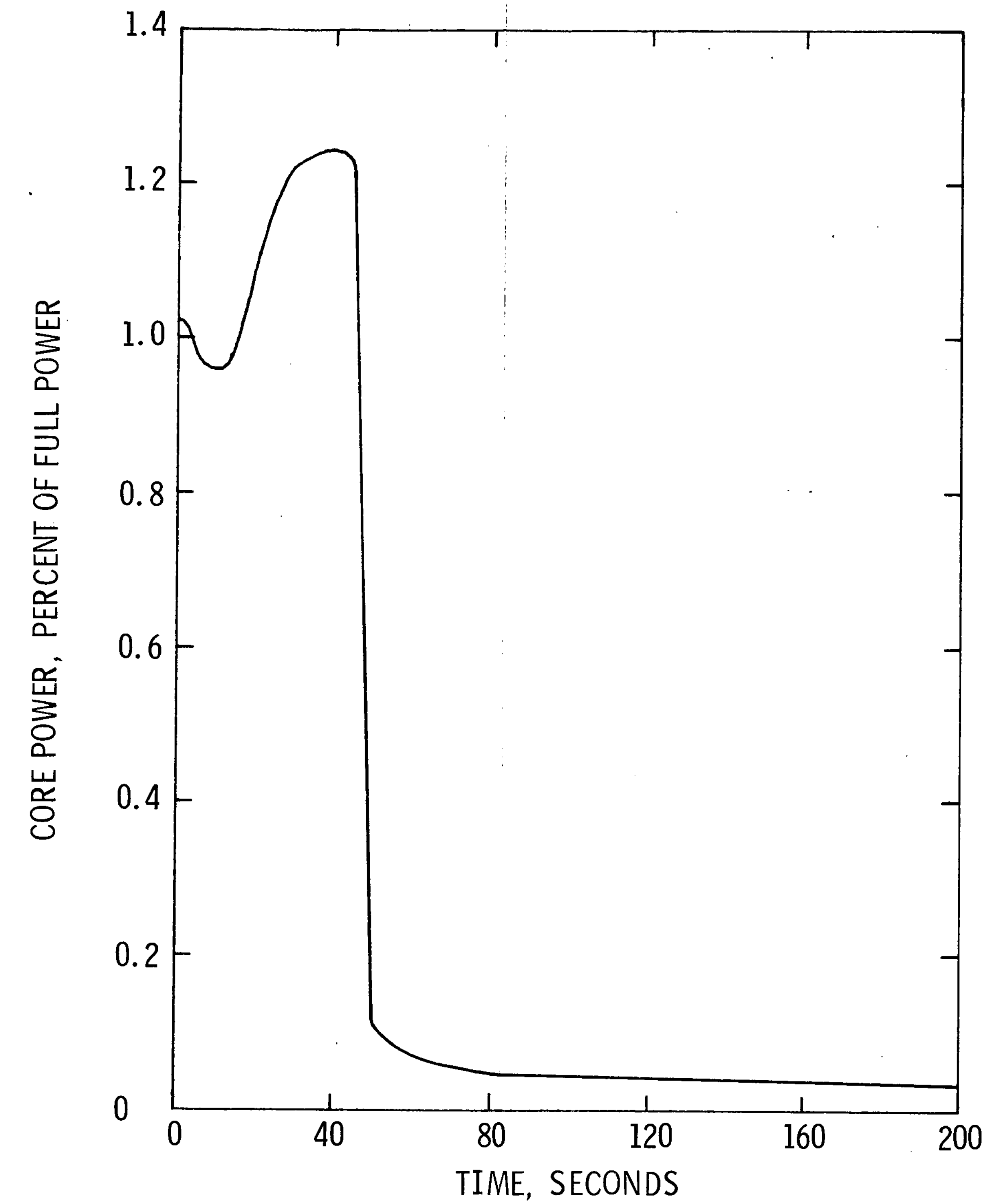


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FEEDWATER SYSTEM PIPE BREAK
REACTIVITY vs. TIME

Figure 15.2-56

10079-11D-1038
10OC6

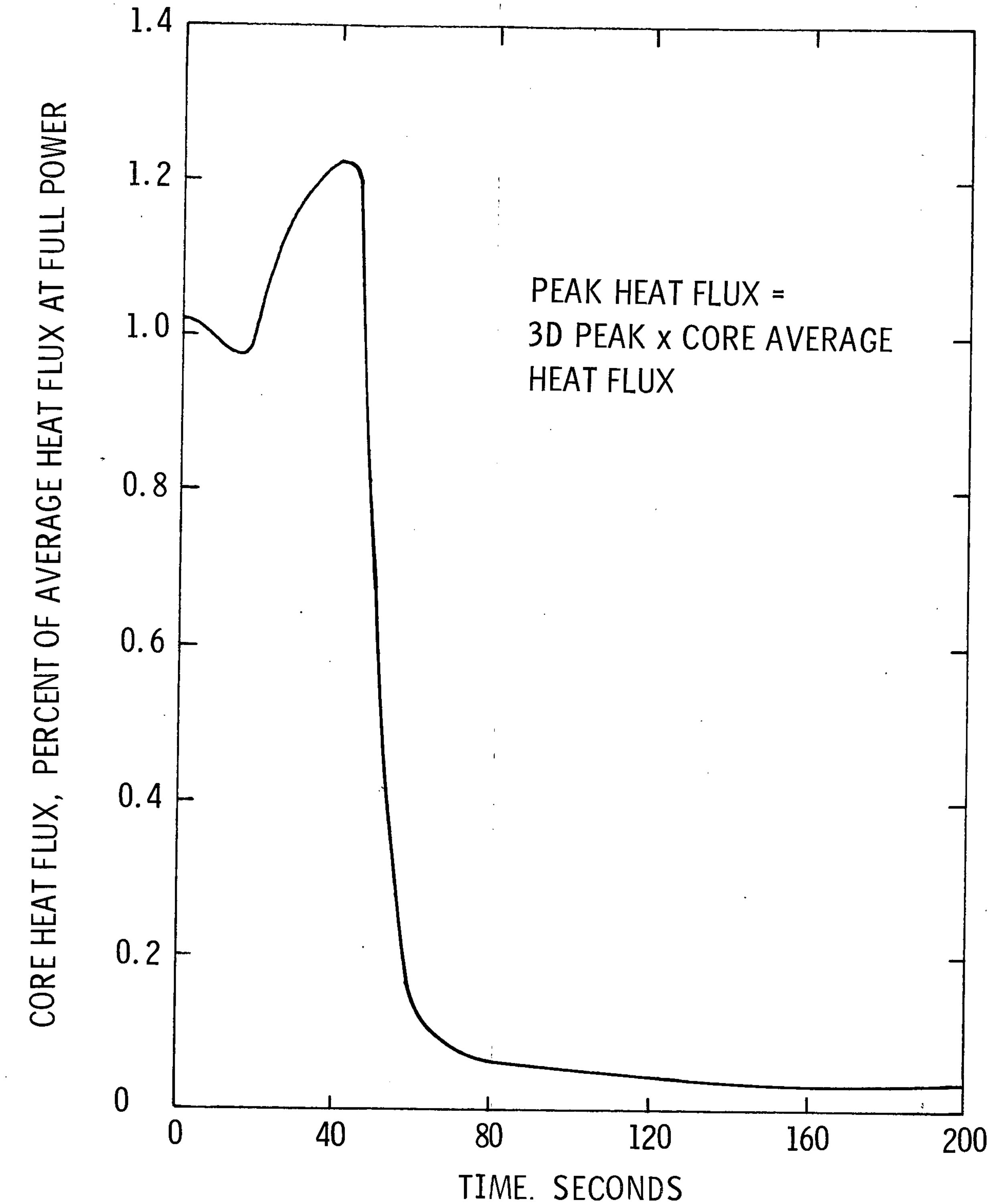


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW WITH TURBINE BYPASS VALVES OPEN CORE POWER vs. TIME

Figure 15.2-57

10079-11D-1039
10OC6

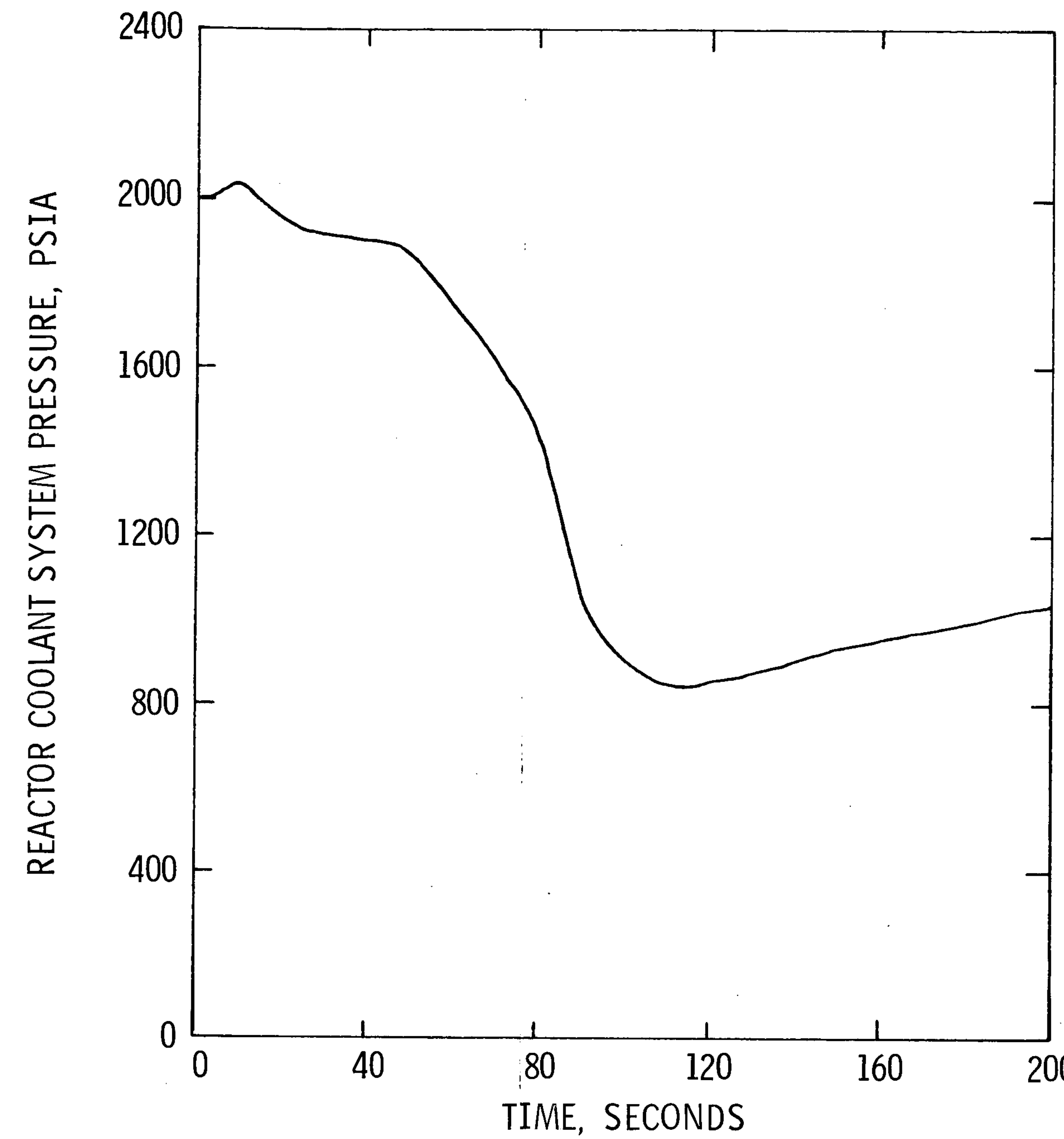


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW WITH TURBINE BYPASS VALVES OPEN CORE HEAT FLUX vs. TIME

Figure 15.2-58

10079-11D-1040
10OC6

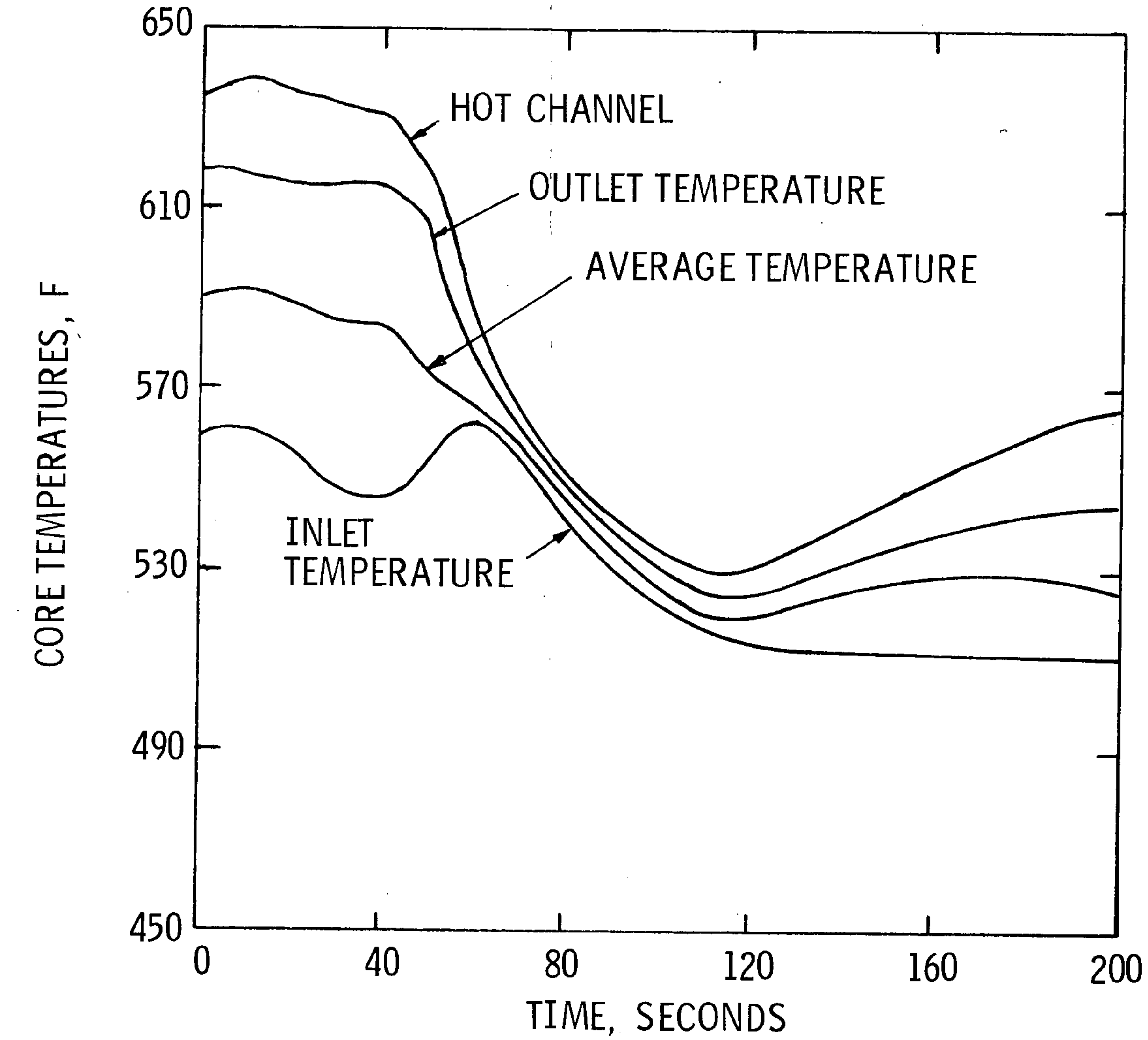


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW WITH TURBINE BYPASS VALVES OPEN REACTOR COOLANT SYSTEM PRESSURE vs. TIME

Figure 15.2-59

10079-11D-1041
10OC6

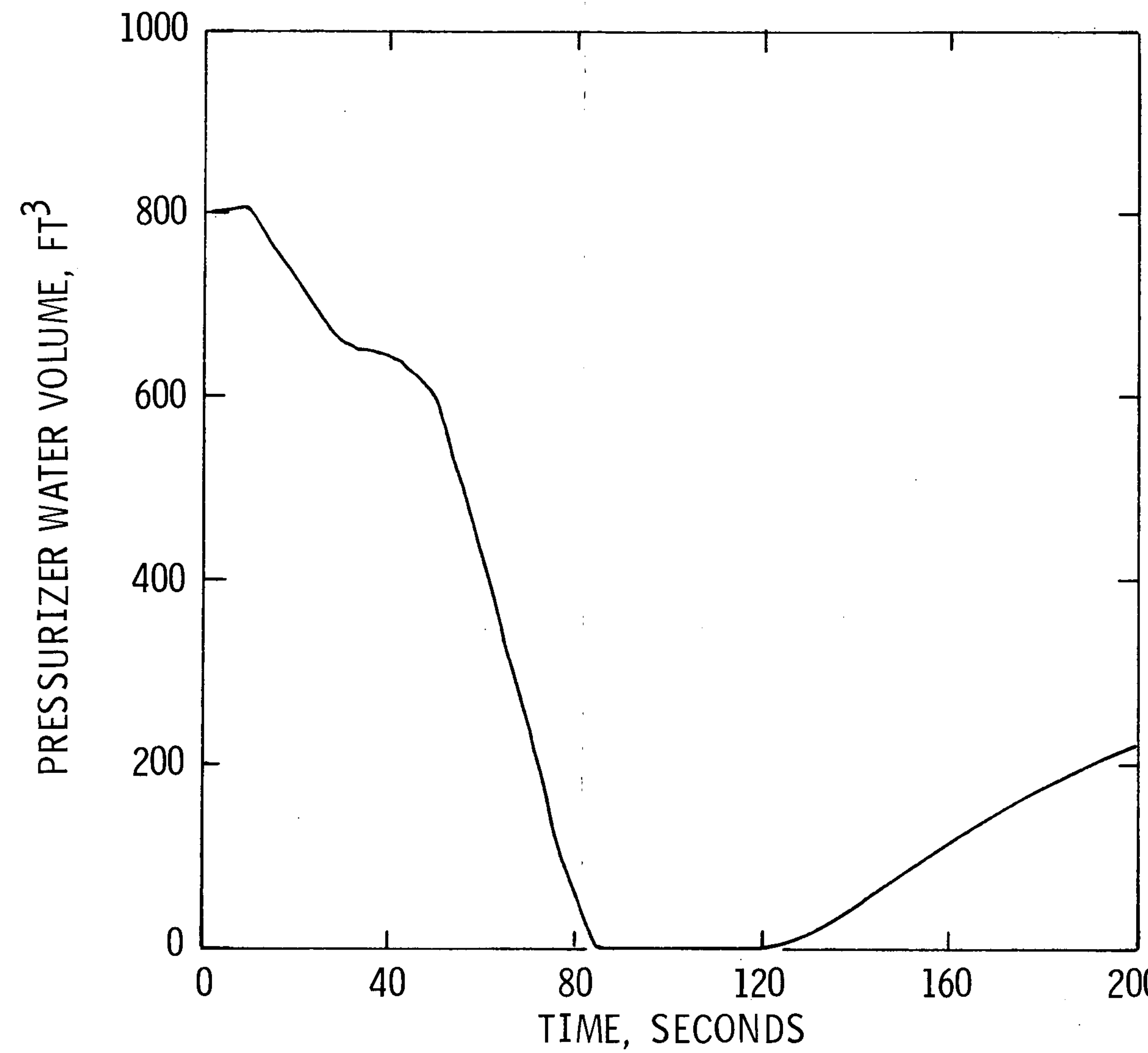


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW WITH TURBINE BYPASS VALVES OPEN CORE TEMPERATURES vs. TIME

Figure 15.2-60

10079-11D-1042
10OC6

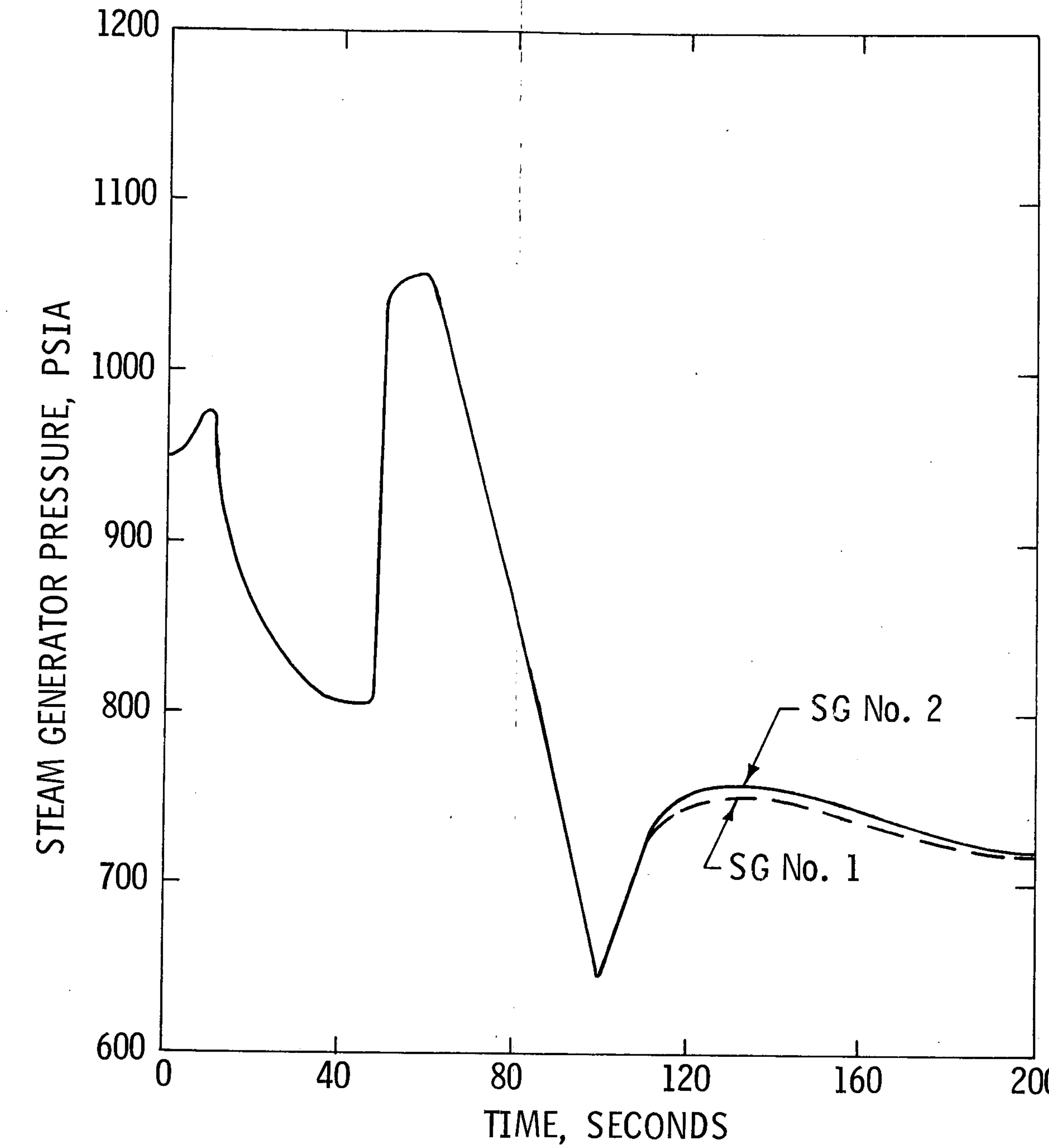


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW WITH TURBINE BYPASS VALVES OPEN PRESSURIZER WATER VOLUME vs. TIME

Figure 15.2-61

10079-11D-1043
10OC6

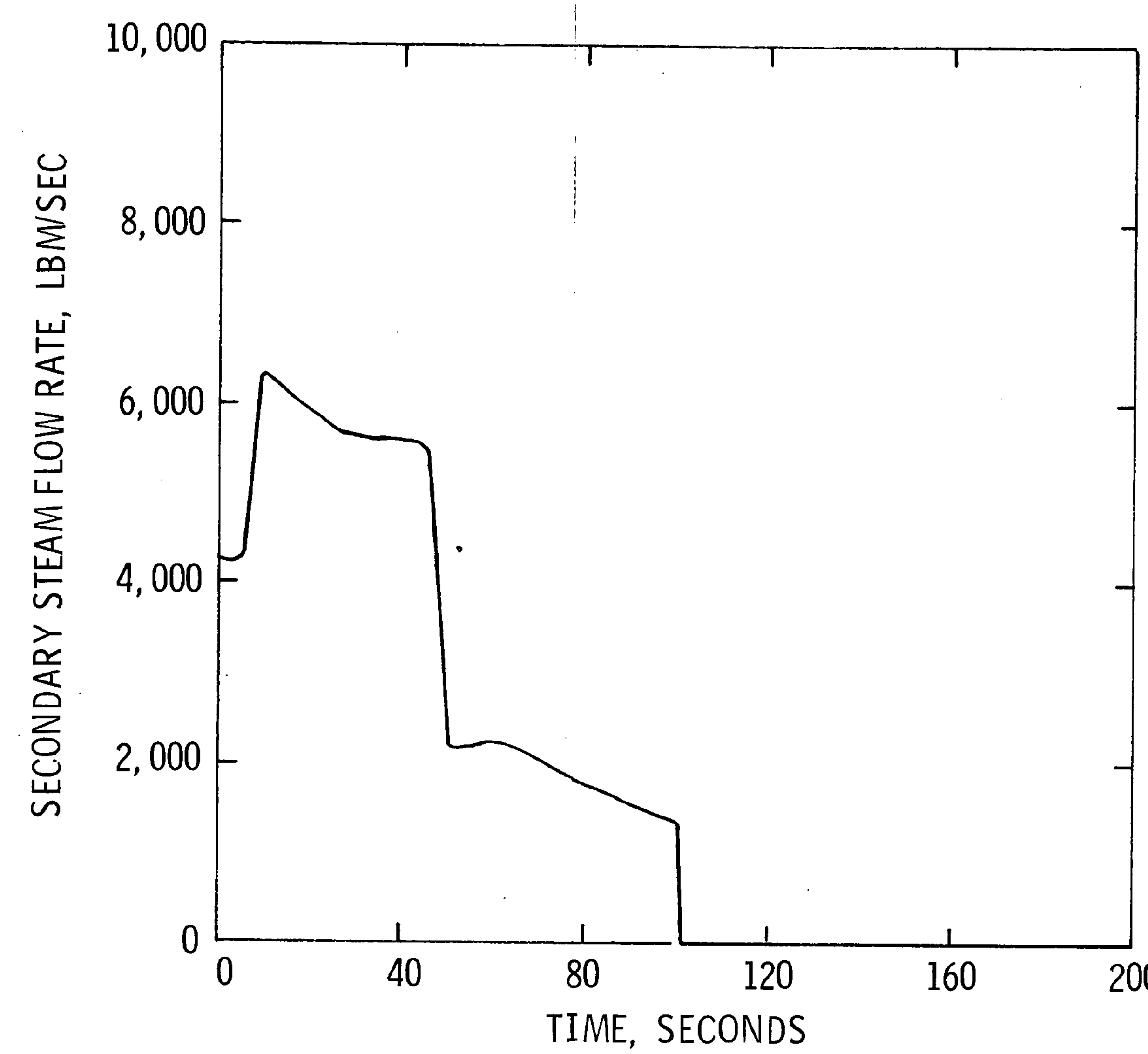


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW WITH TURBINE BYPASS VALVES OPEN STEAM GENERATOR PRESSURE vs. TIME

Figure 15.2-62

10079-11D-1044
10OC6

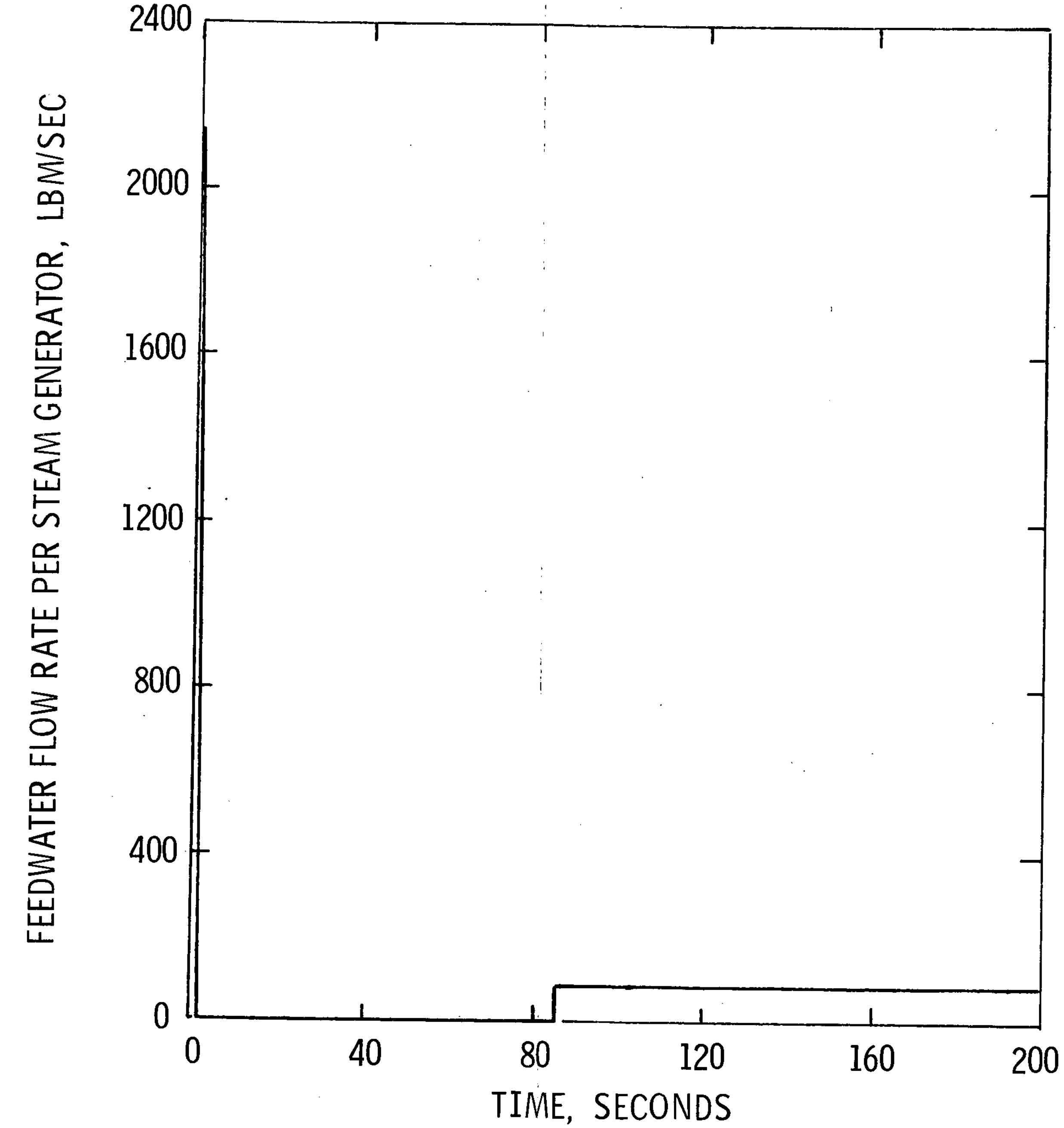


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
WITH TURBINE BYPASS VALVES
OPEN SECONDARY STEAM
FLOWRATE vs. TIME

Figure 15.2-63

10079-11D-1045
10OC6

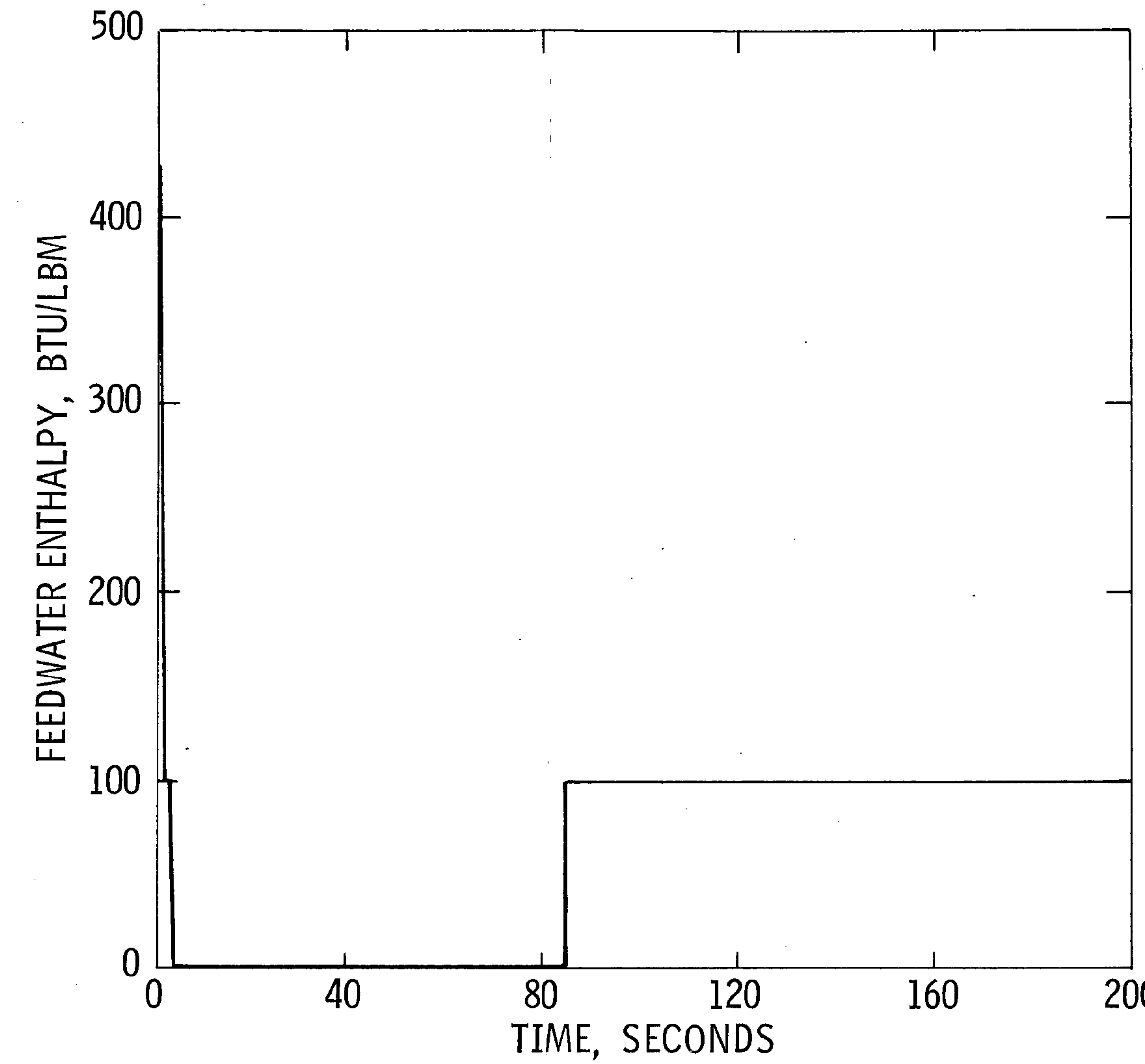


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
WITH TURBINE BYPASS VALVES
OPEN FEEDWATER
FLOWRATE vs. TIME

Figure 15.2-64

10079-11D-1046
10OC6

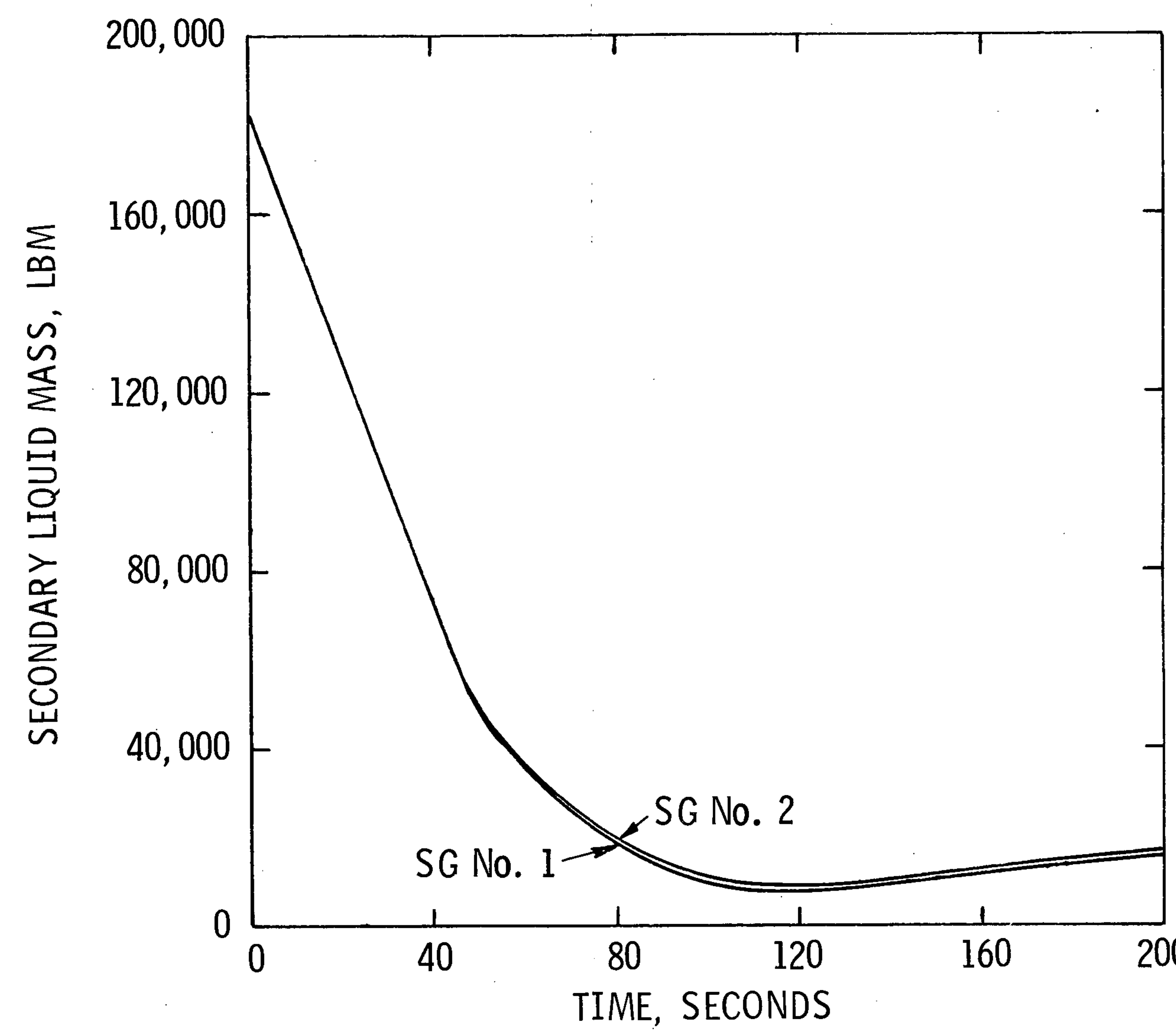


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
WITH TURBINE BYPASS VALVES
OPEN FEEDWATER
ENTHALPY vs. TIME

Figure 15.2-65

10079-11D-1047
10OC6

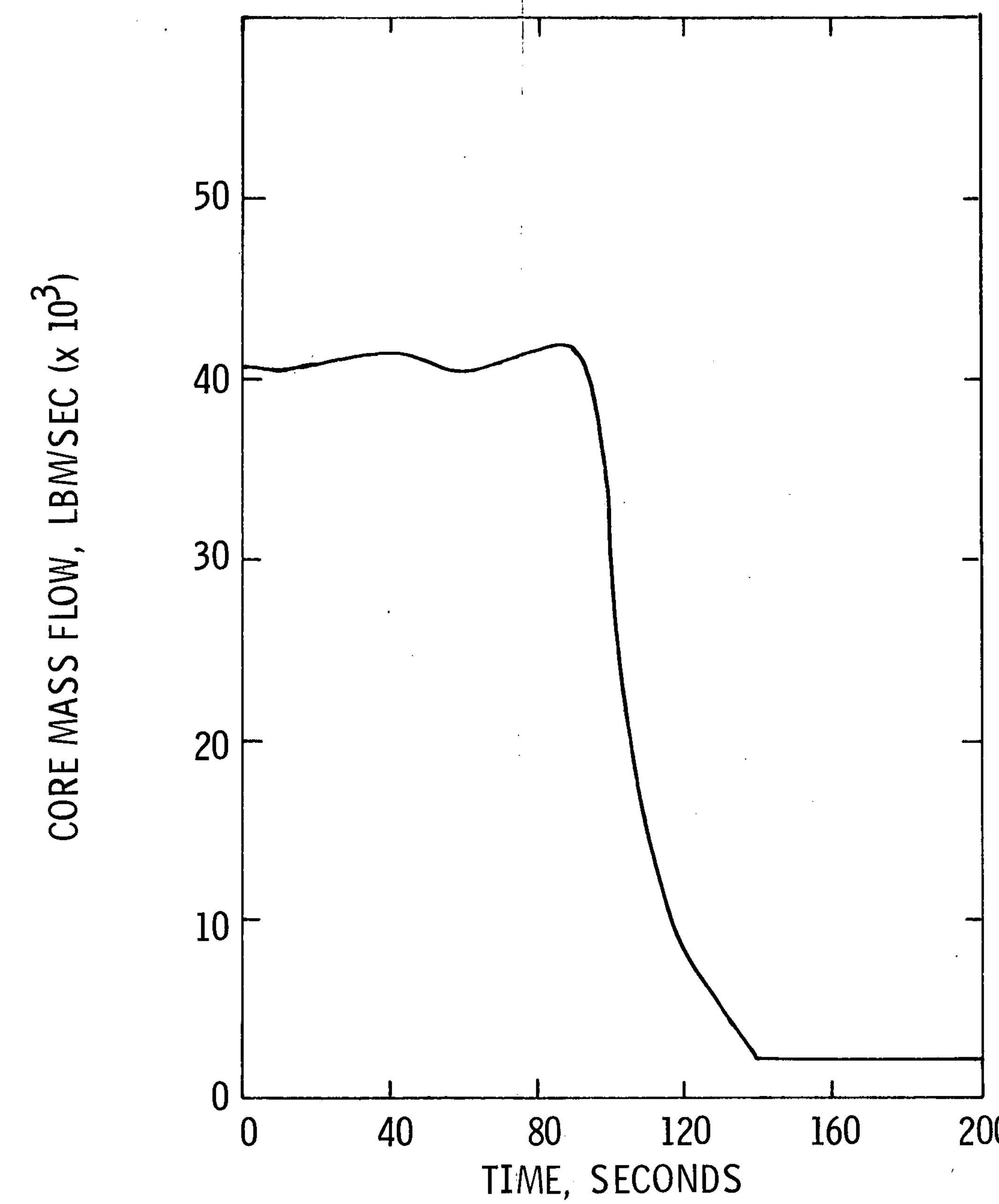


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
WITH TURBINE BYPASS VALVES
OPEN SECONDARY LIQUID MASS
vs. TIME

Figure 15.2-66

10079-11D-1048
10OC6



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LOSS OF NORMAL FEEDWATER FLOW
WITH TURBINE BYPASS VALVES
OPEN CORE MASS
FLOW vs. TIME

Figure 15.2-67

10079-11D-1049
10OC6