REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8510020422 DOC.DATE: 85/09/30 NOTARIZED: YES DOCKET #
FACIL:STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
STN-50-529 Palo Verde Nuclear Station, Unit 2, Arizona Publi 05000529
STN-50-530 Palo Verde Nuclear Station, Unit 3, Arizona Publi 05000530
AUTH.NAME AUTHOR AFFILIATION
VAN BRUNT,E.E. Arizona Nuclear Power Project (formerly Arizona Public Serv
RECIP.NAME RECIPIENT AFFILIATION
KNIGHTON,G.W. Licensing Branch 3

SUBJECT: Forwards proposed changes to FSAR Sections 1.9,2.4,6.3,1.4, 15.1.5 & app 15C incorporating results of main steam line break analyses.Changes will be incorporated into next FSAR amend per NRC approval.

"all repts"

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 6 SIZE: 120
TITLE: OR Submittal: General Distribution

NOTES:Standardized plant. 05000528
OL:12/31/84
Standardized plant. 05000529
Standardized plant. 05000530

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	NRR LB3 BC	01	7	2*				
INTERNAL:	ACRS	09	6	0	ADM/LFMB		1	0
	ELD/HDS3		1	0	NRR/DE/MTEB		1	0
	NRR/DL DIR		1	0	NRR/DL/ORAB		1	0
	NRR/DL/TSRG		1	0	NRR/DSI/METB		1	0
	NRR/DSI/RAB		1	0	REG FILE	04	1	1*
	RGN5		1	0				
EXTERNAL:	24X		1	1*	EG&G BRUSKE,S		1	1
	LPDR	03	1	1*	NRC PDR	02	1	1*
	NSIC	05	1	1*				

Revised Dist
* w/Encl

TOTAL NUMBER OF COPIES REQUIRED: LTTR 28 ENCL 25

[illegible]

Arizona Nuclear Power Project
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

Director of Nuclear Reactor Regulation
Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

September 30, 1985
ANPP-33611-EEVB/KLM

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528(License No. NPF-41)/529/530
Main Steam Line Break Analyses Results - Chapter 15 Reanalyses
File: 85-056-026; G.1.01.10

Reference: Letter to G. W. Knighton, NRC, from E. E. Van Brunt, Jr., ANPP, dated April 15, 1985 (ANPP-32401); Subject: Revised Chapter 15 Analyses

Dear Mr. Knighton:

The attached supplies the results of the following analyses:

FSAR Section	Event
15.1.5	Full Power Steam Line Break + LOP
15.1.5	Full Power Steam Line Break
15.1.5	Zero Power Steam Line Break + LOP
15.1.5	Zero Power Steam Line Break
15.1.5	Full Power Steam Line Break Outside Containment
15.1.5	Zero Power Steam Line Break Outside Containment + LOP

ANPP committed to the reanalysis and submittal of these events in the referenced letter.

A001
1/6

The results are presented as proposed changes to FSAR Sections 1.9.2.4, 6.3.1.4, 15.1.5 and Appendix 15C and are all within NRC acceptance criteria. Upon NRC approval, these FSAR changes will be incorporated into the next PVNGS FSAR Amendment.

Please contact Mr. W. F. Quinn, of my staff, if you should have any questions on this matter.

Very truly yours,

E. E. Van Brunt

E. E. Van Brunt, Jr.
Executive Vice President
Project Director

EEVB/KLM/slh
Attachment

cc: E. A. Licitra
R. P. Zimmerman
A. C. Gehr

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Edwin E. Van Brunt, Jr., represent that I am Executive Vice President, Arizona Nuclear Power Project, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true.

Edwin E. Van Brunt
Edwin E. Van Brunt, Jr.

Sworn to before me this 30 day of September 1985.

Nora E. Meador
Notary Public

My Commission Expires:

My Commission Expires April 6, 1987

1.9.2.4.7 Safety Injection System (CESSAR Section 6.3.1.3.M.8/ FSAR Section 6.3.1.4.M.8)

CESSAR Section 6.3.1.3.M.8.b requires that the volume in each safety injection pipe between the RCS and the first SI valve be less than 30 cubic feet. ~~The PVNGS design satisfies this requirement on the four 12-inch safety injection lines, where the volume is less than 5 cubic feet per pipe~~. INSERT A The PVNGS design has 41 cubic feet of water volume in the long-term recirculation 3-inch pipes. This 11 cubic feet difference is considered negligible since the requirement of 30 cubic feet was specified to minimize boration time response. The four 12-inch lines are used for boration. The long-term recirculation pipes are not used for boration and, therefore, the 41 cubic feet is considered satisfactory.

10 1.9.2.4.8 Containment Spray System (CESSAR Appendix 6A, Section 7.13.7.B/FSAR Section 6.5.2.8.A) (RA)7.13.7.B

CESSAR Appendix 6A, section 7.13.7.B discusses delay time for spray of borated water. This interface requirement uses the term "borated water" because the containment spray pump suction is from the RWT which is borated. The total volume of water in both spray headers inside containment when filled to the 115-foot elevation is less than 700 gallons, and filling this volume with fresh instead of borated water will have no impact on the performance of the containment spray system in removing containment heat or iodine, and negligible potential for boron dilution following sump recirculation after a loss of coolant accident. The intent of the CESSAR interface requirement is met as described in section 6.5.2.8.A (RA)7.13.7.B by filling the spray headers inside containment to the 115-foot elevation with fresh water.

8510020422

INSERT A

The PVNGS design satisfies this requirement on the four 12-inch safety injection lines, where the total unborated volume for all four cold leg injection lines is less than the 120 cubic ~~foot~~ feet requirement.

5. Safety injection system components are properly supported such that pipe stresses and support reactions are within the allowable limits as defined in section 3.9.3.

6. The PVNGS design of the SIS piping and any connecting system piping is such that the loadings imposed on the SIS/RCS nozzles or SIS nozzles are less than the C-E furnished nozzle design loadings for the C-E supplied SIS components.

7. In the event of a limited leakage passive failure in one SIS train during recirculation, personnel access to the intact train will not be precluded due to flooding.

8. Two safety injection check valves in each of the six safety injection lines are located as follows: one as close as practical to the RCS cold leg piping and the other as close as practical to the containment penetration. Additionally, a third check valve is located in each of the four cold leg injection lines upstream of the safety injection tank branch connection in order to minimize high energy piping runs.

   a. Space is provided to permit valve accessiability and maintenance.

   b. The total [unborated] water volume in the piping ~~from the RCS up to the four 12-inch safety injection check valves~~ is less than ~~5 ft^3 per line~~ 120 ft^3. The total water volume in the piping from the RCS up to the two long-term recirculation check valves (3 inch) is less than 41 ft^3 per line. Refer to section 1.9.

   c. Each check valve leakage line is connected to the safety injection line immediately upstream of the safety injection check valve.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.1.1 DECREASE IN FEEDWATER TEMPERATURE

Refer to CESSAR Section 15.1.1.

15.1.2 INCREASE IN FEEDWATER FLOW

Refer to CESSAR Section 15.1.2. 10

15.1.3 INCREASED MAIN STEAM FLOW

Refer to CESSAR Section 15.1.3.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

Refer to CESSAR Section 15.1.4, except that the main feedwater isolation valve closure time increased from 4.6 to 9.6 seconds. Refer to section 1.9.2.4.10 for a discussion of how this deviation does not affect the CESSAR safety analysis. 14

Insert New Section 15.1.5

~~15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT~~ 10

~~15.1.5.1 Identification of Event and Causes~~

Insert 1

~~Refer to CESSAR Section 15.1.5.1, except that the limiting steam line break, with respect to Return to Power CESSAR Case 1 (SLBFPLOP), is reanalyzed for the PVNGS specific feedwater isolation valve closure time and the ruptured generator differential pressure (ΔP) isolation (lockout) setpoint. The primary effect of these two PVNGS specific changes is the additional reactivity inserted due to the additional blowdown of the affected steam generator.~~ CESSAR Cases 2 through 4 are bounded by the Case 1 analysis and have not been revised. 14

CESSAR Cases 5 and 6 were chosen to conservatively maximize the potential for degradation of fuel performance and dose at the site EAB. The Palo Verde changes do not affect the CESSAR Case 5 conclusions regarding the potential for fuel

damage as the minimum DNBR occurs prior to valve actuation. CESSAR Case 6 is bounded by Case 5. Although the delay in valve actuation results in an increase in the amount of
14 condensate released to the atmosphere, there is no appreciable increase in offsite dose due to the negligible activity of the condensate. Therefore, CESSAR Cases 5 and 6 have not been revised.

Refer to section 1.9.2.4.10 for further details.

10 15.1.5.2 Sequence of Events and Systems Operation

The sequence of events for CESSAR Case 1, SLBFPLOP, is presented in table 15.1-1. For Cases 2 through 6, refer to
14 CESSAR Section 15.1.5.2. Also, Palo Verde utilizes other reactor trip signals - steam generator ΔP low flow and variable overpower. These signals are redundant to the CPC trip signal. The effect of these reactor trip signals has been included in section 15.1.5.3. Additionally, CIAS or SIAS signals will actuate control room and fuel building essential ventilation systems. See sections 6.4 and 9.4 for details. CIAS or SIAS will terminate the containment power
10 access purge as described in section 9.4.

15.1.5.3 Analysis of Effects and Consequences

Refer to CESSAR Section 15.1.5.3 since the 0 to 2 hour χ/Q value presented in section 2.3 is smaller than that assumed in the CESSAR radiological consequence analysis.

A. Mathematical Model
12 Refer to CESSAR Section 15.1.5.3.A.

B. Input Parameters and Initial Conditions
Refer to CESSAR Section 15.1.5.3.B, except that the volume to be swept out of the safety injection lines before boron enters the RCS is 40 ft^3 for PVNGS, compared to 120 ft^3 for CESSAR.

Additionally, Case 1 utilizes the PVNGS specific feedwater isolation valve closure time of 9.6 seconds and the steam generator ΔP isolation setpoint of 325 psid. 14

C. Results

Case 1: Large Steam Line Break During Full Power Operation with Concurrent Loss of Offsite Power (SLBFPLOP) 12

The dynamic behavior of the salient NSSS parameters following the SLBFPLOP is presented in figures 15.1-1 through 15.1-16.

Concurrent with the steam line break, a loss of offsite power occurs. At this time, an actuation signal for the emergency diesel generators is initiated. An auxiliary feedwater actuation signal (AFAS) is conservatively assumed to be immediately initiated for the ruptured steam generator. Due to decreasing core flow following loss of power to the reactor coolant pumps, conditions exist for a steam generator ΔP low flow or low DNBR trip. Additionally, for inside containment breaks, a trip can be initiated by a high containment pressure trip signal. At 0.6 seconds, a trip signal is initiated. At 0.75 seconds, the reactor trip breakers open. After a 0.34 second coil decay delay, the CEAs begin to drop into the core at 1.09 seconds. Turbine trip occurs upon reactor trip. Main steam flow is not affected since the SLB is a double-ended guillotine break of a main steam line. At 7.6 seconds, voids begin to form in the upper head of the reactor vessel. 14

At 9.1 seconds, the steam generator pressure drops below the main steam isolation signal (MSIS) analysis setpoint of 810 psia. This results in generation of an MSIS at 10.1 seconds. The MSIS initiates closure of the main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs). The MSIVs close by 14.7 seconds and the MFIVs close by 19.7 seconds. At 28 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 325 psid for lockout of auxiliary feedwater (AFW) to the ruptured steam generator. Isolation of AFW from the ruptured steam generator begins at 29 seconds. By 44 seconds switching of the AFW to the intact steam generator is complete; AFW to the ruptured steam generator has been isolated and the AFW valves to the intact steam generator are fully open. At 94 seconds, the pressurizer empties. At 130 seconds, the pressurizer pressure has dropped below the analysis setpoint of 1600 psia for safety injection actuation, resulting in initiation of a safety injection actuation signal (SIAS) at 140 seconds. Within 30 seconds of SIAS, the high pressure safety injection (HPSI) valves are fully open and the operable HPSI pump delivers full flow. At 230 seconds, the affected steam generator empties. Safety injection boron begins to reach the core at 249 seconds. At 276 seconds, the maximum core reactivity (-0.03% Δρ) occurs. As shown by figure 15.1-16 the values of DNBR remain above those for which fuel damage would be indicated. At a maximum of 30 minutes, the operator commences a controlled plant cooldown via the appropriate atmospheric dump

valves, assuming that offsite power has not been restored. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions. 14

The SLBFPLOP sequence of events is summarized in table 15.1-1.

Case 2: Large Steam Line Break During Full Power Operation with Offsite Power Available (SLBFP) 12

Refer to CESSAR Section 15.1.5.3.C.

Additionally, after a steam line break a trip can be initiated by a variable overpower trip signal, or for breaks inside containment, a high containment pressure trip signal. 14

Case 3: Large Steam Line Break During Zero Power Operation with Concurrent Loss of Offsite Power (SLBZPLOP) 12

Refer to CESSAR Section 15.1.5.3.C.

Additionally, after a steam line break with the loss of offsite power a trip can be initiated by a steam generator ΔP low flow trip signal, or for breaks inside containment, a high containment pressure trip signal. 14

Case 4: Large Steam Line Break Zero Power Operation with Offsite Power Available (SLBZP)

See CESSAR Section 15.1.5.3.C. 12

Case 5: Small Steam Line Break Outside Containment During Full Power Operation with Offsite Power Available (SSLBFP)

See CESSAR Section 15.1.5.3.C.

Case 6: Large Steam Line Break Outside Containment from Zero Power Operation with Loss of Offsite Power (SLBZPLOPD)

12 See CESSAR Section 15.1.5.3.C.

15.1.5.4. <u>Conclusion</u>

Refer to CESSAR Section 15.1.5.4.

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT

15.1.5.1 Identification of Event and Causes

A steam line break (SLB) is defined as a pipe break in the main steam system. SLB cases are chosen to maximize potential for a post-trip return to power, to maximize potential for degradation in fuel cladding performance, and to maximize dose at the site Exclusion Area Boundary. The results show that fission power levels remain sufficiently low following reactor trip to preclude degradation in fuel performance as a result of post-trip return to power, that degradation in fuel performance prior to trip is of sufficiently limited extent, that the core will remain in place and intact with no loss of core cooling capability, and that doses are within 10CFR100 guidelines. The steam line breaks presented are:

A. Cases chosen to maximize potential for a post-trip return to power:

1. A large steam line break inside containment during full power operation with concurrent loss of offsite power in combination with a single failure, and a stuck CEA (SLBFPLOP).

2. A large steam line break inside containment during full power operation with offsite power available in combination with a single failure and a stuck CEA (SLBFP).

3. A large steam line break inside containment during zero power operation with concurrent loss of offsite power in combination with a single failure, and a stuck CEA (SLBZPLOP).

4. A large steam line break inside containment during zero power operation with offsite power available in combination with a single failure and a stuck CEA (SLBZP).

B. Cases chosen to maximize potential for degradation in fuel performance and dose at the site Exclusion Area Boundary:

5. A steam line break outside of containment upstream of the main steam isolation valve (MSIV) during full power operation with offsite power available in combination with a single failure, technical specification steam generator tube leakage, and a stuck CEA (SLBFPD).

6. A large steam line break outside of containment upstream of the MSIV during zero power operation with concurrent loss of offsite power in combination with a single failure, technical specification steam generator tube leakage, iodine spike, and a stuck CEA (SLBZPLOPD).

The largest possible steam line break size is the double ended rupture of a steam line upstream of the MSIV. In the System 80 design, an integral flow restrictor exists in each steam generator outlet nozzle. The largest effective steam blowdown area for each steam line is limited by the flow restrictor throat area which is 1.28 square feet.

Parameters and conditions for maximizing post-trip degradation in fuel performance are discussed in Appendix 15C. Parameters and conditions for maximizing pre-trip degradation in fuel performance and for maximizing the secondary system contribution to radiological releases are also discussed in Appendix 15C.

These six cases are analyzed for an end of equilibrium self-generated plutonium recycle (SGR) core. Minimum safety injection flow rates, feedwater isolation valve closure time, and ruptured generator differential pressure (ΔP) isolation (lockout) setpoint are employed in the current analysis. In addition, an auxiliary feedwater (AFW) system model has been used in the analysis along with improvements to the boron transport and mixing model, and the upper head void model. The model changes and improvements are discussed in Appendix 15C.

The primary effect of using PVNGS specific safety injection minimum flow rates is to delay and reduce the effect of boron injection into the RCS. The use of a PVNGS specific feedwater isolation valve closure time and steam generator differential pressure lockout setpoint results in additional moderator reactivity inserted due to the additional blowdown of the affected steam generator and resulting cooldown of the RCS.

15.1.5.2 Sequence of Events and Systems Operation

Steam line breaks are characterized as cooldown events due to the increased steam flow rate, which causes excessive energy removal from the steam generators and the reactor coolant system (RCS). This results in a decrease in reactor coolant temperatures and in RCS and steam generator pressure. The cooldown causes an increase in core reactivity due to the assumed negative moderator and Doppler reactivity coefficients.

Detection of the cooldown is accomplished by the pressurizer and steam generator low pressure alarms, by the high reactor power alarm and by the low steam generator water level alarm. Reactor trip as a consequence of a steam line break is provided by one of several available reactor trip signals including low steam generator pressure, low RCS pressure, low steam generator water level, high reactor power, low DNBR trip initiated by the core protection calculators (CPC) and, for inside containment breaks, high containment pressure, steam generator ΔP low flow, and variable overpower.

Following reactor trip the most reactive control rod is conservatively assumed to be held in the fully withdrawn position. The depressurization of the affected steam generator results in the actuation of a main steam isolation signal (MSIS). This closes the MSIVs, isolating the unaffected steam generator from blowdown and closes the main feedwater isolation valves (MFIVs), terminating main feedwater flow to both steam generators. After the reduction of steam flow that occurs with MSIV closure, the level in the intact steam generator falls below the auxiliary feedwater actuation signal (AFAS) setpoint. The resulting AFAS causes auxiliary feedwater (AFW) flow to be initiated to both steam generators. When the differential pressure between the two steam generators exceeds the setpoint, the AFW logic isolates flow to the affected steam generator and diverts the flow from both AFW pumps to the

intact steam generator. The pressurizer pressure decreases to the point where a safety injection actuation signal (SIAS) is initiated. The isolation of the unaffected steam generator and subsequent emptying of the affected steam generator terminates the cooldown. The introduction of safety injection boron upon SIAS causes core reactivity to decrease. The operator, via the appropriate emergency procedures, may initiate plant cooldown by manual control of the atmospheric steam dump valves, or, in the event that offsite power is available, by using the MSIV bypass valves associated with the unaffected steam generator and the turbine bypass valves, any time after the affected steam generator empties. The analysis presented herein conservatively assumes operator action is delayed until 30 minutes after first indication of the event. The plant is then cooled to 350°F and 400 psia, at which point shutdown cooling is initiated. In addition, CIAS or SIAS signals will activate control room and fuel building essential ventilation systems. See Sections 6.4 and 9.4 for details. CIAS or SIAS will terminate the containment power access purge as described in Section 9.4.

A parametric study of single failures (See Appendix 15C) that would have an adverse impact on the SLB has determined that the failure of one of the high pressure safety injection (HPSI) pumps to start following SIAS has the most adverse effect for all cases except Case 5. Consequently, one HPSI pump is conservatively assumed to fail for these cases. For Case 5 (SLBFPD) there is no single failure which increases the potential for degradation in fuel cladding performance or which increases the offsite dose. However, the failure of a HPSI pump to start was used in the analysis to be consistent with Case 2 (SLBFP). The evaluation shows that for the full power SLB without loss of offsite power (Case 2) deviates from CESSAR Case 2 since the failure of an MSIV on Palo Verde leads to a non-isolable leak of only 1.5% (.034 ft^2) of the design steam rate which is much less than the 11% (.256 ft^2) of the design steam rate assumed in CESSAR.

The sequence of events for Cases 1 through 5 above are presented in Tables 15.1-1 through 5, respectively. The sequence of events for Case 6 is the same as for Case 3.

15.1.5.3 Analysis of Effects and Consequences

A. Mathematical Models

The mathematical models and data transfer between codes used in the SLB analysis are presented in Appendix 15C.

B. Input Parameters and Initial Conditions

The initial conditions assumed in the analysis of the NSSS response to Cases 1 through 5 are presented in Tables 15.1-6 through 10, respectively. The initial conditions for Case 6 are the same as those for Case 3. Justification of the selection of initial conditions and input parameters is presented in Appendix 15C. The PVNGS specific feedwater isolation valve closure time of 9.6 seconds, steam generator differential pressure lockout setpoint of 325 psid and safety injection minimum pump flow rates are used in the analysis.

Case 1: Large Steam Line Break During Full Power Operation with Concurrent Loss of Offsite Power (SLBFPLOP)

The dynamic behavior of the salient NSSS parameters following the SLBFPLOP is presented in Figures 15.1.1-1 through 15.1.1-17. Table 15.1-1 summarizes the major events, times, and results for this transient.

Concurrent with the steam line break, a loss of offsite power occurs. This initiates turbine stop valve closure, termination of feedwater to both steam generators and the coastdown of the reactor coolant pumps. At this time, an actuation signal for the emergency diesel generators is initiated. Due to decreasing core flow following loss of power to the reactor coolant pumps, conditions exists for a low DNBR trip. In addition, after a steam line break with the loss of offsite power, a trip can be initiated by a steam generator ΔP low flow trip signal, or for breaks inside containment on high containment pressure. 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. Turbine trip occurs upon reactor trip. Main steam flow is not affected since the SLB is a double-ended quillotine break of a main steam line. At 7.7 seconds, voids begin to form in the upper head of the reactor vessel. At 9.6 seconds, the steam generator pressure drops below the MSIS setpoint of 810 psia. This results in generation of an MSIS at 10.6 seconds. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 15.2 seconds. AFW is automatically initiated at 15.2 seconds assuming no delay after the AFAS signal, which is assumed to be generated coincident with the MSIV closure. The MFIVs close by 20.2 seconds. At 30 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 325 psia for lockout of auxiliary feedwater to the ruptured steam generator. Isolation of AFW from the ruptured steam generator begins at 31 seconds. By 46 seconds, switching of the AFW to the intact steam generator is complete; AFW to the ruptured steam generator has been isolated and the AFW valves to the intact steam generator remain fully open. At 86 seconds, the pressurizer empties. At 129 seconds, the pressurizer pressure has dropped below 1600 psia, and at 130 seconds a SIAS is generated. Within 30 seconds of SIAS, the operable HPSI pump is loaded on the diesels and reaches full speed and the HPSI valves are fully open and the operable HPSI pump delivers full flow.

Safety injection boron begins to reach the core at 261 seconds. At 315 seconds, the affected steam generator empties. At 325 seconds, AFW to the intact steam generator is automatically terminated. At 372 seconds, the maximum transient core reactivity (+0.05 % Δρ) occurs. At 384 seconds, the minimum transient DNBR (3.86) occurs. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the atmospheric dump valves, assuming that offsite power has not been restored. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

Case 2: Large Steam Line Break During Full Power Operation with Offsite Power Available (SLBFP)

The dynamic behavior of the salient NSSS parameters following the SLBFP is presented in Figures 15.1.2-1 through 15.1.2-16. Table 15.1-2 summarizes the major events, times, and results for this transient.

At 6.95 seconds after the initiation of the steam line break, a trip signal is initiated by the core protection calculators on a projected DNBR of 1.19. In addition, a trip can be initiated by a variable overpower trip signal, or for breaks inside containment, a high containment pressure trip signal. At 710 seconds, the reactor trip breakers open. At 12.0 seconds voids begin to form in the upper head of the reactor vessel. At 14.2 seconds, the steam generator pressure drops below the MSIS setpoint of 810 psia. This results in generation of an MSIS at 15.2 seconds. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 19.8 seconds. AFW is automatically initiated at 19.8 seconds assuming no delay after the AFAS signal, which is assumed to be generated coincident with the MSIV closure.

The MFIV's close by 24.8 seconds. At 40 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 325 psid for lockout of auxiliary feedwater (AFW) to the ruptured steam generator. Isolation of AFW from the ruptured steam generator begins at 41 seconds. By 56 seconds, switching of the AFW to the intact steam generator is complete; AFW to the ruptured steam generator has been isolated and the AFW valves to the intact steam generator remain fully open.

At 69 seconds, the pressurizer empties. At 104 seconds, the pressurizer pressure drops below 1600 psia, and at 105 seconds a SIAS is generated. Within 30 seconds of SIAS, the HPSI pumps reach full speed and the HPSI valves are fully open. At 186 seconds, the maximum core reactivity (-0.45% $\Delta\rho$) occurs. At 205 seconds, the affected steam generator empties. Safety injection boron begins to reach the core at 270 seconds. DNB remains above 10 during the post-trip portion of this transient. At 359 seconds, AFW to the intact steam generator is automatically terminated. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the turbine bypass valves. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

Case 3: Large Steam Line Break During Zero Power Operation with Concurrent Loss of Offsite Power (SLBZPLOP)

The dynamic behavior of the salient NSSS parameters following the SLBZPLOP is presented in Figures 15.1.3-1 through 15.1.3-16. Table 15.1-3 summarizes the major events, times, and results for this transient.

Concurrent with the steam line break, a loss of offsite power occurs. At this time, an actuation signal for the emergency diesel generators is initiated. Due to decreasing core flow following loss of power to the reactor coolant pumps, conditions exist for a low DNBR trip. In addition, after a steam line break with the loss of offsite power a trip can be initiated by a steam generator ΔP low flow trip signal, or for breaks inside containment, a high

containment pressure trip signal. At 0.6 second a low DNBR trip signal is initiated by the core protection calculators. At 0.75 seconds, the reactor trip breakers open. At 8.5 seconds, the steam generator pressure drops below the MSIS setpoint of 810 psia. This results in generation of an MSIS at 9.5 seconds. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 14.1 seconds. AFW is automatically initiated at 14.1 seconds assuming no delay after the AFAS signal, which is assumed to be generated coincident with the MSIV closure. The MFIVs close by 19.1 seconds. At 35 seconds, the pressurizer empties.

At 36 seconds, the pressurizer pressure drops below 1600 psia and at 37 seconds a SIAS is generated. Within 30 seconds of SIAS, the operable HPSI pump is loaded on the diesels and reaches full speed and the HPSI valves are fully open. At 38 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 325 psid for lockout of auxiliary feedwater (AFW) to the ruptured steam generator. Isolation of AFW from the ruptured steam generator begins at 39 seconds. At 48 seconds, voids begin to form in the upper head of the reactor vessel. By 54 seconds, switching of the AFW to the intact steam generator is complete, AFW to the ruptured steam generator has been isolated, and the AFW valves to the intact steam generator remain fully open. At 65 seconds, AFW to the intact steam generator is automatically terminated. Safety injection boron begins to reach the core at 126 seconds. At 211 seconds, the maximum core reactivity (-0.22% $\Delta\rho$) occurs. DNBR remains above 10 during this transient. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the atmospheric dump valves, assuming that offsite power has not been restored. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

Case 4: Large Steam Line Break Zero Power Operation with Offsite Power Available (SLBZP)

The dynamic behavior of the salient NSSS parameters following the SLBZP is presented in Figures 15.1.4-1 through 15.1.4-16. Table 15.1-4 summarizes the major events, times, and results of this transient.

At 8.6 seconds after initiation of the steam line break, the steam generator pressure drops below the low steam generator pressure trip and MSIS setpoint of 810 psia. At 9.6 seconds, a low steam generator pressure trip signal and MSIS are generated. At 9.75 seconds, the reactor trip breakers open. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 14.2 seconds. AFW is automatically initiated at 14.2 seconds assuming no delay after the AFAS signal, which is assumed to be generated coincident with the MSIV closure. The MFIVs close by 19.2 seconds.

At 41 seconds, the pressurizer empties. At 42 seconds, the pressurizer pressure drops below 1600 psia, and at 43 seconds an SIAS is generated. Also at 42 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 325 psid for lockout of auxiliary feedwater (AFW) to the ruptured steam generator. Isolation of AFW from the ruptured steam generator begins at 43 seconds. At 51 seconds, voids begin to form in the upper head of the reactor vessel. By 58 seconds, switching of the AFW to the

intact steam generator is complete; AFW to the ruptured steam generator has been isolated, and the AFW valves to the intact steam generator remain fully open. At 72 seconds, AFW to the intact steam generator is automatically terminated. At 73 seconds (30 seconds following the SIAS), the operable HPSI pump reaches full speed and the HPSI valves are fully open. Safety injection boron begins to reach the core at 134 seconds. At 326 seconds, the maximum core reactivity (-0.154% $\Delta\rho$) occurs. At 419 seconds, the affected steam generator empties.

At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by manual control of the MSIV bypass valves associated with the unaffected steam generator and turbine bypass valves. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

Case 5: Steam Line Break Outside Containment During Full Power Operation with Offsite Power available (SLBFPD)

The dynamic behavior of the salient NSSS parameters following a typical limiting SLBFPD is presented in Figures 15.1.5-1 through 15.1.5-8. Table 15.1-5 summarizes the major events, times and results for this transient.

The consequences of this transient (fraction of fuel rods predicted to experience DNB) are nearly the same as those for SLBFPDs for a spectrum of break sizes, due to the protective action of the core protection calculators (CPCs). See the discussion in CESSAR Section 15C.3.2 and Figure 15C-1 of CESSAR Appendix 15C. The largest break size yields the minimum DNBR. Therefore, the transient presented here is that which results from the double ended break of a main steam line.

Not later than 5.85 seconds after initiation of the steam line break, a trip signal is initiated by the CPCs on a projected DNBR of 1.19. At 6.0 seconds, the reactor trip breakers open. At 7.49 seconds, a minimum transient DNBR of 1.11 is calculated to occur, after which DNBR rapidly increases. At 9.0 seconds voids begin to form in the upper head of the reactor vessel. At 12.2 seconds, the steam generator pressure drops below the MSIS setpoint of 810 psia. This results in generation of an MSIS at 13.2 seconds. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 17.8 seconds. AFW is automatically initiated at 17.8 seconds assuming no delay after the AFAS signal, which is assumed to be generated coincident with the MSIV closure. The MFIVs close by 22.8 seconds. At 34 seconds, the pressure difference between the steam generators reaches the analysis setpoint of 325 psid for lockout of auxiliary feedwater (AFW) to the ruptured steam generator. Isolation of AFW from the ruptured steam generator begins at 35 seconds. By 50 seconds, switching of the AFW to the intact steam generator is complete; AFW to the ruptured steam generator has been isolated and the AFW valves to the intact steam generator remain fully open.

Subsequently, the events of this transient follow a sequence similar to those of the SLBFP (Case 2). Since the cooldown is less severe, the potential for post-trip degradation in the fuel cladding performance is less for this case

(SLBFPD) than for Case 2 (SLBFP). At a maximum of 30 minutes, the operator, using the appropriate emergency procedure, initiates plant cooldown by manual control of the turbine bypass valves. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

At the point of the minimum transient DNBR, no more than 0.7% of the fuel rods are predicted to experience DNB. The minimum DNBR and assumed percentage of failed fuel are identical to the values stated in CESSAR Section 15.1-5 for this case. All of the activity in the fuel gap for fuel rods that are assumed to fail is assumed to be uniformly mixed with the reactor coolant. The activity in the fuel clad gap is assumed to be 10% of the iodines and 10% of the noble gases accumulated in the fuel at the end of core life, assuming continuous full power operation. This results in a primary coolant activity of 618 μCi/gm. Assuming one gpm steam generator tube leakage, during a period of two hours after initiation of the SLBFPD the integral leakage from the RCS through the affected steam generator is 890 lbm, which is assumed to be released to the atmosphere with a DF of 1. This mass release results in a contribution to the inhalation thyroid dose at the Exclusion Area Boundary (EAB) of not more than 40 rem.

The total steam released from the affected steam generator is 167,000 lbm. The affected steam generator will empty in two hours; therefore, all the mass release from the affected steam generator to the atmosphere has a DF of 1. The calculated inhalation thyroid dose is not more than 1.5 rem for the blowdown originating from the secondary system fluid discharge from the affected steam generator.

Less than 74,000 lbm of steam from the unaffected steam generator will be released through the steam line break. During the SLBFPD the MSIVs will isolate the unaffected steam generator from the break and prevent it from emptying. Therefore, a DF of 100 is assumed in calculating iodine activity released from the unaffected steam generator. The resulting contribution to the inhalation thyroid dose at the EAB is less than 0.1 rem.

The foregoing doses are calculated by the methods outlined in Section 15.0.4. In summary, the total two-hour inhalation thyroid dose at the EAB as a consequence of the SSLBFP is no more than 42 rem.

Case 6: Large Steam Line Break Outside Containment from Zero Power Operation with Loss of Offsite Power (SLBZPLOPD)

Case 6 in included in Case 3, since the break of the latter can be either inside or outside of containment. The Figures, Tables, and Discussion for Case 3 apply to Case 6.

Assuming one gpm steam generator tube leakage, during a period of two hours after initiation of the SLBZPLOPD the integral leakage from the RCS through the affected steam generator is 890 lbm, which is assumed to be released to the atmosphere with a DF of 1. This mass release results in a contribution to the inhalation thyroid doses at the EAB of:

(a) 0.1 rem, assuming technical specification primary coolant activity;
(b) 4.0 rem, assuming a pre-existing iodine spike; or
(c) 2.3 rem, assuming an event-induced iodine spike.

The total steam released from the affected steam generator is 315,000 lbm. The affected steam generator will empty in two hours; therefore, all the mass release from the affected steam generator to atmosphere has a DF of 1. The calculated inhalation thyroid dose is 2.4 rem for the blowdown steam originating from the affected steam generator.

Less than 53,000 lbm of steam from the unaffected steam generator will be released through the steam line break. During the SLBZPLOPD, the MSIVs will isolate the unaffected steam generator and prevent it from emptying. Therefore, a DF of 100 is assumed in calculating iodine activity released from the unaffected steam generator. The resulting contribution to the inhalation thyroid dose at the EAB is less than 0.1 rem.

The foregoing doses are calculated by the methods outlined in CESSAR Section 15.0.4. ~~Table 15.1.5-11 presents the major assumptions, parameters, and radiological consequences for this transient.~~

In summary, the total two-hour inhalation thyroid dose at the EAB as a consequence of the SLBZPLOPD is no more than 6.5 rem.

15.1.5.4 Conclusion

For the large steam line break in combination with a single failure and stuck CEA, with or without a loss of offsite power, fission power remains sufficiently low following reactor trip to preclude fuel damage as a result of post-trip return to power.

For a large steam line break during zero power operation in combination with a loss of offsite power and technical specification tube leakage the two-hour inhalation thyroid dose at the EAB is well within 10CFR100 guidelines.

(a) 2.6 rem, assuming technical specification primary coolant activity;
(b) 6.5 rem, assuming a pre-existing iodine spike; or
(c) 4.8 rem, assuming an event-induced iodine spike.

The maximum potential for radiological releases due to fuel failure occurs for full power steam line breaks outside containment in combination with a stuck CEA. For these cases, the maximum potential for degradation in fuel cladding performance occurs prior to and during reactor trip. With the assumption of one gallon per minute steam generator tube leakage and a bounding assumption of 0.7% fuel failure the two-hour inhalation thyroid dose at the EAB is calculated to be no more than 42 rem, which is within the 10 CFR100 guidelines.

Potential fuel failure is sufficiently limited to ensure that the core will remain in place and intact with no loss of core cooling capabilities.

Table 15.1-1

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBFPLOP) (Sheet 1 of 2) 12 14

Time (sec)	Event	Setpoint or Value
0.0	Steam Line Break and Loss of Off-site Power Occur, AFAS Assumed to be Generated for Ruptured Steam Generator	--
0.6	CPC Low DNBR Trip Signal or Steam Generator ΔP Low Flow Trip Signal Generated, % of Initial Full Power Mass Flow	1.19 70
0.75	Reactor Trip Breakers Open	--
7.6	Voids Begin to Form in RV Upper Head	--
9.1	Steam Generator Pressure reaches Main Steam Isolation Signal (MSIS) Analysis Setpoint, psia	810
10.1	MSIS Generated	--
14.7	MSIVs Completely Closed	--
19.7	MFIVs Completely Closed	--
28.0	Difference Between Steam Generator Pressures reaches Analysis Set-point for Lockout of AFW to Ruptured Steam Generator, psid	325
29.0	Signal to Isolate AFW from Ruptured Steam Generator Generated	--
44.0	AFW Isolated from Ruptured Steam Generator; AFW Valves to Intact Steam Generator Fully Open	--
94	Pressurizer Empties	--

12 14

Table 15.1-1

12 SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS

14 OF OFFSITE POWER (SLBFPLOP) (Sheet 2 of 2)

12

Time (sec)	Event	Setpoint or Value
139	Pressurizer Pressure reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1600
140	SIAS Generated	--
170	Safety Injection Flow Begins	--
230	Affected Steam Generator Empties	--
249	Safety Injection Boron Begins to Reach Reactor Core	--
276	Maximum Transient Reactivity, $10^{-2}\ \Delta\rho$	-0.03
285	Minimum Post-Trip DNBR	5.4
1800	Operator Initiates Cooldown	--

14

Table 15.1-2

DELETED

Table 15.1-3

DELETED

TABLE 15.1.5-1 (page 1 of 3)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBFPLOP)

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break and Loss of Offsite Power Occur	--
0.6	CPC Low DNBR Trip Signal, Projected DNBR or	1.19
	Steam Generator ΔP Low Flow Trip Signal Generated, % of Initial Full Power Mass Flow	70
0.75	Reactor Trip Breakers Open	--
7.7	Voids Begin to Form in RV Upper Head	--
9.6	Steam Generator Pressure reaches Main Steam Isolation Signal (MSIS) Analysis Setpoint, psia	810
10.6	MSIS Generated	--
15.2	MSIVs Completely Closed	--
15.2	AFW Flow Initiated	--
20.2	MFIVs Completely Closed	--
30	Difference Between Steam Generator Pressures reaches Analysis Setpoint for Lockout of AFW to Ruptured Steam Generator, psid	325

TABLE 15.1.[illegible]-1 (page 2 of 3)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBFPLOP)

Time (Sec)	Event	Setpoint or Value
31	Signal to Isolate AFW from Ruptured Steam Generator Generated	— —
46	AFW Isolated from Ruptured Steam Generator; AFW Valves to Intact Steam Generator Fully Open	— —
86	Pressurizer Empties	— —
129	Pressurizer Pressures reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1600
130	SIAS Generated	— —
160	Safety Injection Flow Begins	— —
261	Safety Injection Boron Begins to Reach Reactor Core	— —
315	Affected Steam Generator Empties	— —
325	AFW to Intact Steam Generator Terminated, % of wide range	80

TABLE 15.1.8-1 (page 3 of 3)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBFPLOP)

Time (Sec)	Event	Setpoint or Value
372	Maximum Transient Reactivity, $10^{-2}\Delta\rho$	+0.05
384	Minimum Transient DNBR	3.86
1800	Operator Initiates Cooldown	--

TABLE 15.1.~~X~~-2 (page 1 of 2)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	--
6.95	CPC Low DNBR Trip Signal Generated, Projected DNBR	1.19
7.10	Reactor Trip Breakers Open	--
12.0	Voids Begin to Form in RV Upper Head	--
14.2	Steam Generator Pressure reaches Main Steam Isolation Signal (MSIS) Analysis Setpoint, psia	810
15.2	MSIS Generated	--
19.8	MSIVs Completely Closed	--
19.8	AFW Flow Initiated	--
24.8	MFIVs Completely Closed	--
40	Difference Between Steam Generator Pressures reaches Analysis Setpoint for Lockout of AFW to Ruptured Steam Generator, psid	325
41	Signal to Isolate AFW from Ruptured Steam Generator Generated	--

TABLE 15.1.X-2 (page 2 of 2)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Time (Sec)	Event	Setpoint or Value
56	AFW Isolated from Ruptured Steam Generator; AFW Valves to Intact Steam Generator Fully Open	— —
69	Pressurizer Empties	— —
104 (14.2)	Pressurizer Pressure reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1600
105	SIAS Generated	— —
135 (15.2)	Safety Injection Flow Begins	— —
186	Maximum Transient Reactivity, $10^{-2}\ \Delta\rho$	-0.45
205	Affected Steam Generator Empties	— —
270	Safety Injection Boron Begins to Reach Reactor Core	— —
359	AFW to Intact Steam Generator Terminated,	80
1800 (41)	Operator Initiates Cooldown	— —

TABLE 15.1.8-3 (page 1 of 2)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING ZERO POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBZPLOP AND SLBZPLOPD)

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break and Loss of Offsite Power Occur	--
0.6	CPC Low DNBR Trip Signal, Projected DNBR or Steam Generator ΔP Low Flow Trip Signal Generated, % of Initial Full Power Mass Flow	1.19 70
0.75	Reactor Trip Breaker Open	--
8.5	Steam Generator Pressure reaches Main Steam Isolation Signal (MSIS) Analysis Setpoint, psia	810
9.5	MSIS Generated	--
14.1	MSIVs Completely Closed	--
14.1	AFW Flow Initiated	--
19.1	MFIVs Completely Closed	--
35	Pressurizer Empties	--
36	Pressurizer Pressure reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1600

TABLE 15.1.5-3 (page 2 of 2)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING ZERO POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBZPLOP AND SLBZPLOPD)

Time (Sec)	Event	Setpoint or Value
37	SIAS Generated	--
38	Difference Between Steam Generator Pressures reaches Analysis Setpoint for Lockout of AFW to Ruptured Steam Generator, psid	325
39	Signal to Isolate AFW from Ruptured Steam Generator Generated	--
48	Voids Begin to Form in RV Upper Head	--
54	AFW Isolated from Ruptured Steam Generator; AFW Valves to Intact Steam Generator Fully Open	--
65	AFW to Intact Steam Generator Terminated, % of wide range.	80
67	Safety Injection Flow Begins	--
126	Safety Injection Boron Begins to Reach Reactor Core	--
211	Maximum Transient Reactivity, $10^{-2}\Delta\rho$	-0.22
1800	Operator Initiates Cooldown	--

TABLE 15.1.X-4 (page 1 of 2)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING ZERO POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBZP)

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	--
8.6	Steam Generator Pressure reaches Main Steam Isolation Signal (MSIS) Analysis Setpoint and Low Steam Generator Pressure Trip Setpoint, psia.	810
9.6	Low Steam Generator Pressure Trip signal and MSIS generated	--
9.75	Reactor Trip Breakers Open	--
14.2	MSIVs Completely Closed	--
14.2	AFW Flow Initiated	--
19.2	MFIVs Completely Closed	--
41	Pressurizer Empties	--
42	Pressurizer Pressure reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1600
42	Difference Between Steam Generator Pressures reaches Analysis Setpoint for Lockout of AFW to Ruptured Steam Generator, psid	325

TABLE 15.1.X-4 (page 2 of 2)

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING ZERO POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBZP)

Time (Sec)	Event	Setpoint or Value
43	Signal to Isolate AFW from Ruptured Steam Generator Generated	--
43	SIAS Generated	--
51	Voids Begin to Form in RV Upper Head	--
58	AFW Isolated from Ruptured Steam Generator; AFW Valves to Intact Steam Generator Fully Open	--
72	AFW to Intact Steam Generator Terminated, % of wide range.	80
73	Safety Injection Flow Begins	--
134	Safety Injection Boron Begins to Reach Reactor Core	--
326	Maximum Transient Reactivity, $10^{-2}\Delta\rho$	-0.154
419	Affected Steam Generator Empties	--
1800	Operator Initiates Cooldown	--

TABLE 15.1.~~X~~-5 (page 1 of 2)

SEQUENCE OF EVENTS FOR A STEAM LINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFPD)

Time (Sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	--
5.85	CPC Low DNBR Trip Signal Generated, Projected DNBR	1.19
6.0	Reactor Trip Breakers Open	--
7.49	Minimum Transient DNBR	1.11
9.0	Voids Begin to Form in RV Upper Head	--
12.2	Steam Generator Pressure reaches Main Steam Isolation Signal (MSIS) Analysis Setpoint, psia	810
13.2	MSIS Generated	--
17.8	MSIVs Completely Closed	--
17.8	AFW Flow Initiated	--
22.8	MFIVs Completely Closed	--
34	Difference Between Steam Generator Pressures reaches Analysis Setpoint for Lockout of AFW to Ruptured Steam Generator, psid	325

TABLE 15.1.~~X~~-5 (page 2 of 2)

SEQUENCE OF EVENTS FOR A STEAM LINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFPD)

Time (Sec)	Event	Setpoint or Value
35. 5.95	Signal to Isolate AFW from Ruptured Steam Generator Generated	— —
50	AFW Isolated from Ruptured Steam Generator; AFW Valves to Intact Steam Generator Fully Open	— —
80.10	Pressurizer Pressure reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, psia	1600
82	SIAS Generated	— —
93	Maximum Transient Reactivity, $10^{-2} \Delta \rho$	-1.95
112	Safety Injection Flow Begins	— —
113	Affected Steam Generator Empties	— —
1800	Operator Initiates Cooldown	— —

PVNGS FSAR

TABLE 15.1.X-6

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBFPLOP)

Parameter	Assumed Value
Initial Core Power Level, MWt	3876
Initial Core Inlet Coolant Temperature, F	570
Initial Core Mass Flow Rate, 10^6 lbm/hr	148.
Initial Pressurizer Pressure, psia	2400
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	1.15
Moderator Coefficient Multiplier	1.10
Axial Shape Index	+.3
CEA Worth for Trip, 10^{-2} $\Delta\rho$	~~-8.8~~ -9.0
Initial Steam Generator Inventory, lbm, ~~affected~~ ~~intact~~	~~182000~~ ~~148000~~ 197,000
One High Pressure Safety Injection Pump	Inoperative
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown Area for Each Steam Line, ft^2	1.283

TABLE 15.1.~~X~~-7

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAM LINE BREAK DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFP)

Parameter	Assumed Value
Initial Core Power Level, MWt	3876
Initial Core Inlet Coolant Temperature, F	570
Initial Core Mass Flow Rate, 10^6 lbm/hr	148
Initial Pressurizer Pressure, psia	2400
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	1.15
Moderator Coefficient Multiplier	1.10
Axial Shape Index	+.3
CEA Worth for Trip, 10^{-2} $\Delta\rho$	~~-10.8~~ -9.0
Initial Steam Generator Inventory, lbm~~, affected~~ ~~intact~~	~~182000~~ ~~140000~~ 197,200
~~One Main Steam Isolation Valve on Intact Steam Generator~~	~~Inoperative~~
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown Area for Each Steam Line, ft^2	1.283
One High Pressure Safety Injection Pump	Inoperative

PVNGS FSAR

TABLE 15.1.~~X~~-8

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAM LINE BREAK DURING ZERO POWER OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBZPLOP AND SLBZPLOPD)

Parameters	Assumed Value
Initial Core Power Level, MWt	10
Initial Core Inlet Coolant Temperature, F	575
Initial Core Mass Flow Rate, 10^6 lbm/hr	147.
Initial Pressurizer Pressure, psia	2400
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	1.15
Moderator Coefficient Multiplier	1.10
Axial Shape Index	+.3
CEA Worth for Trip, $10^{-2}\Delta\rho$	-6.0
Initial Steam Generator Inventory, lbm, ~~affected~~ ~~intact~~	~~279000~~ ~~143000~~ 311,000
One High Pressure Safety Injection Pump	Inoperative
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown Area for Each Steam Line, ft^2	1.283

TABLE 15.1.X-9

ASSUMPTIONS AND INITIAL CONDITIONS FOR A LARGE STEAM LINE BREAK DURING ZERO POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBZP)

Parameter	Assumed Value
Initial Core Power Level, MWt	10
Initial Core Inlet Coolant Temperature, F	575
Initial Core Mass Flow Rate, 10^6 lbm/hr	147.
Initial Pressurizer Pressure, psia	2400
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	1.15
Moderator Coefficient Multiplier	1.10
Axial Shape Index	+.3
CEA Worth for Trip, $10^{-2}\,\Delta\rho$	-6.0
Initial Steam Generator Inventory, lbm, ~~affected~~ ~~intact~~	~~279000~~ 311,000 ~~163000~~
One High Pressure Safety Injection Pump	Inoperative
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown Area for Each Steam Line, ft^2	1.283

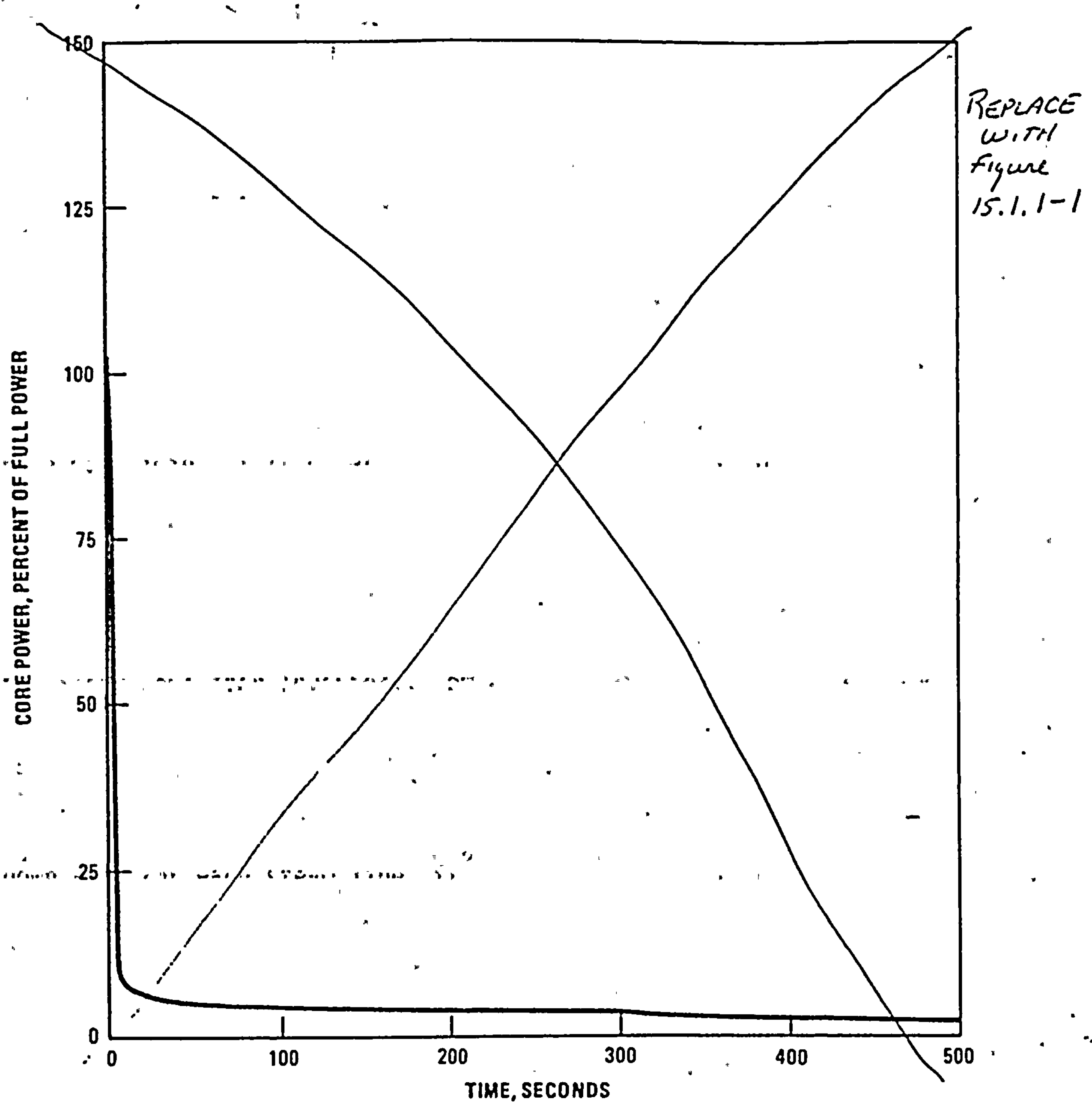
PVNGS FSAR

TABLE 15.1.~~X~~-10

ASSUMPTIONS AND INITIAL POWER CONDITIONS FOR A STEAM LINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFPD)

Parameter	Assumptions
Initial Core Power Level, MWt	3876
Initial Core Inlet Coolant Temperature, F	570
Initial Core Mass Flow Rate, 10^6 lbm/hr	148
Initial Pressurizer Pressure, psia	~~2199~~ 2139
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	0.85
Moderate Coefficient Multiplier	1.10
Axial Shape Index	+0.3
Radial Peaking Factor, F_R	1.42
CEA Worth for Trip, 10^{-2} $\Delta\rho$	~~8.0~~ -9.0
Initial Steam Generator Inventory, lbm ~~affected~~ ~~intact~~	~~122000~~ 122,000 ~~122000~~
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown area for each steam line, ft^2	1.283

9

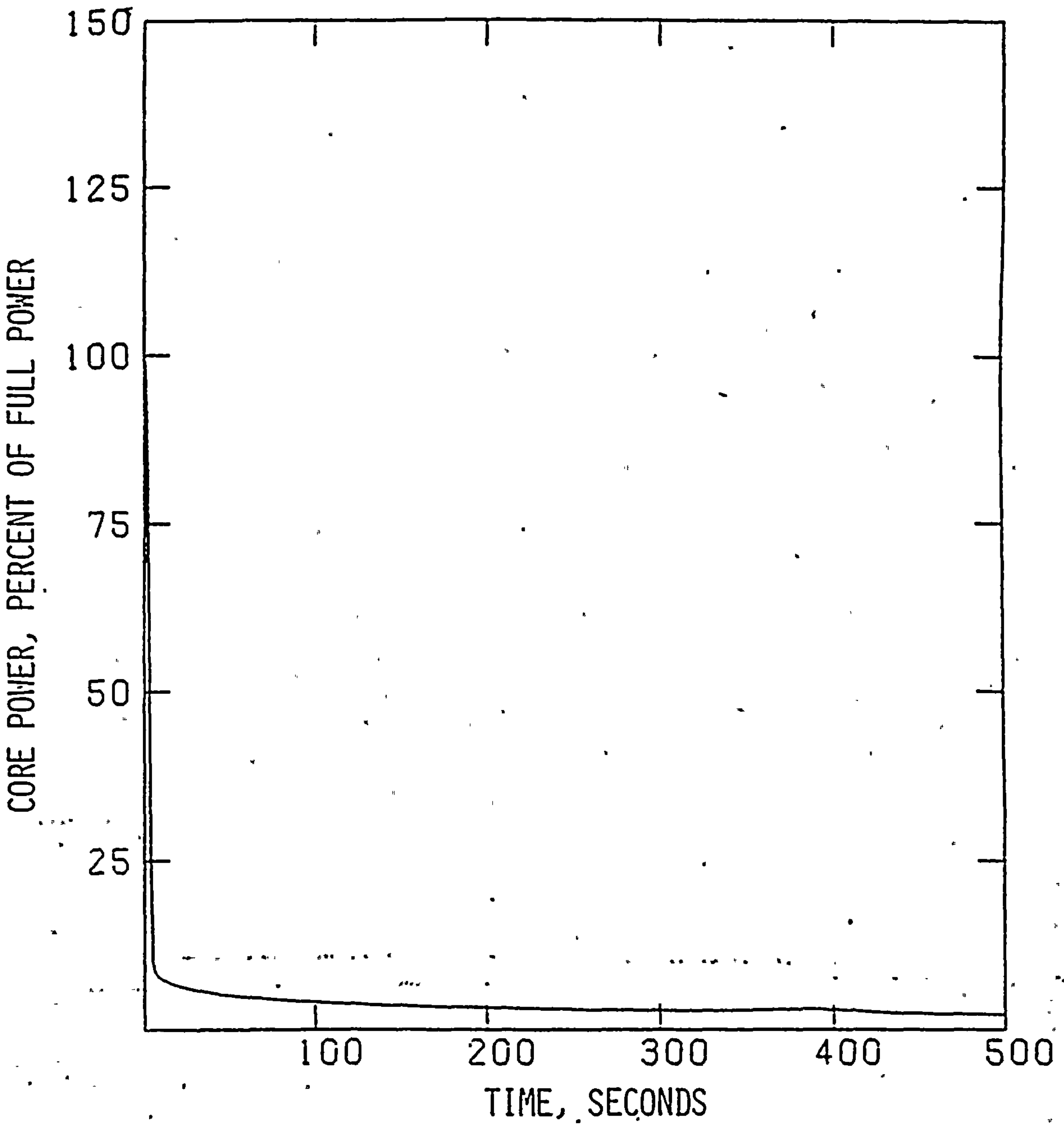


14

Palo Verde Nuclear Generating Station
FSAR

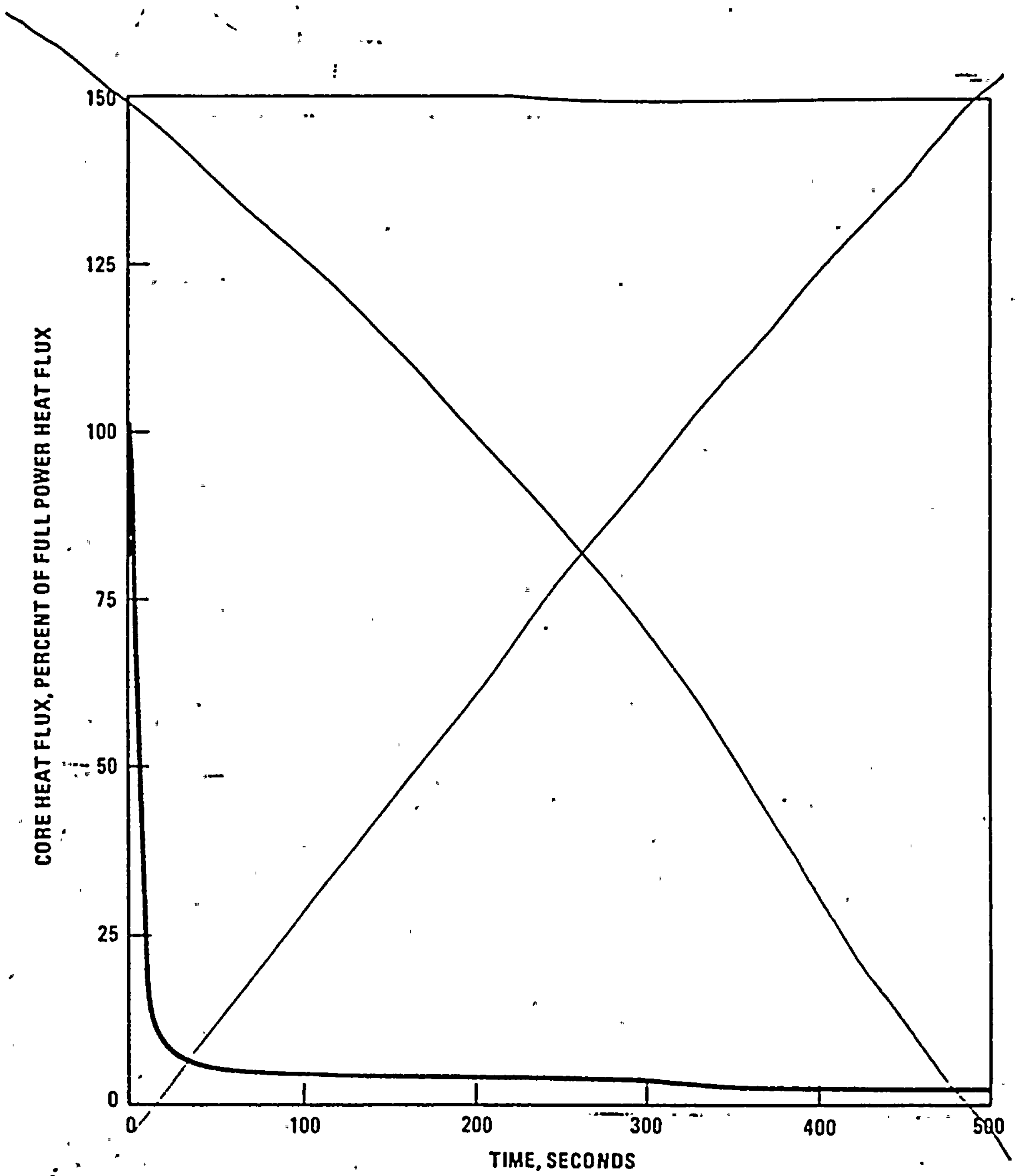
FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER
CORE POWER VS TIME

Figure 15.1-1



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER
CORE POWER VS TIME
Figure 15.1.1-1

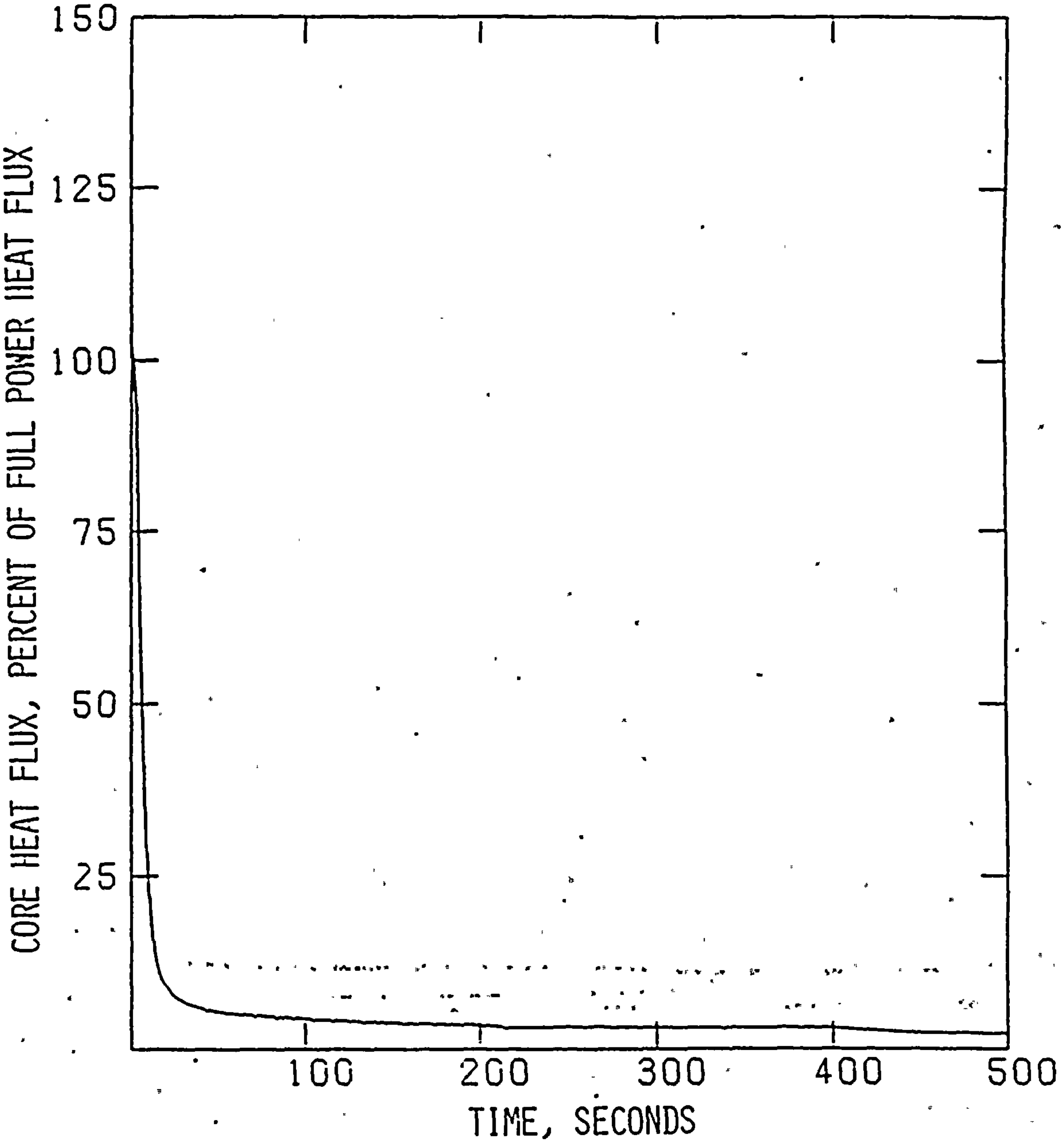


Replace WITH figure 15.1.1-2

Palo Verde Nuclear Generating Station
FSAR

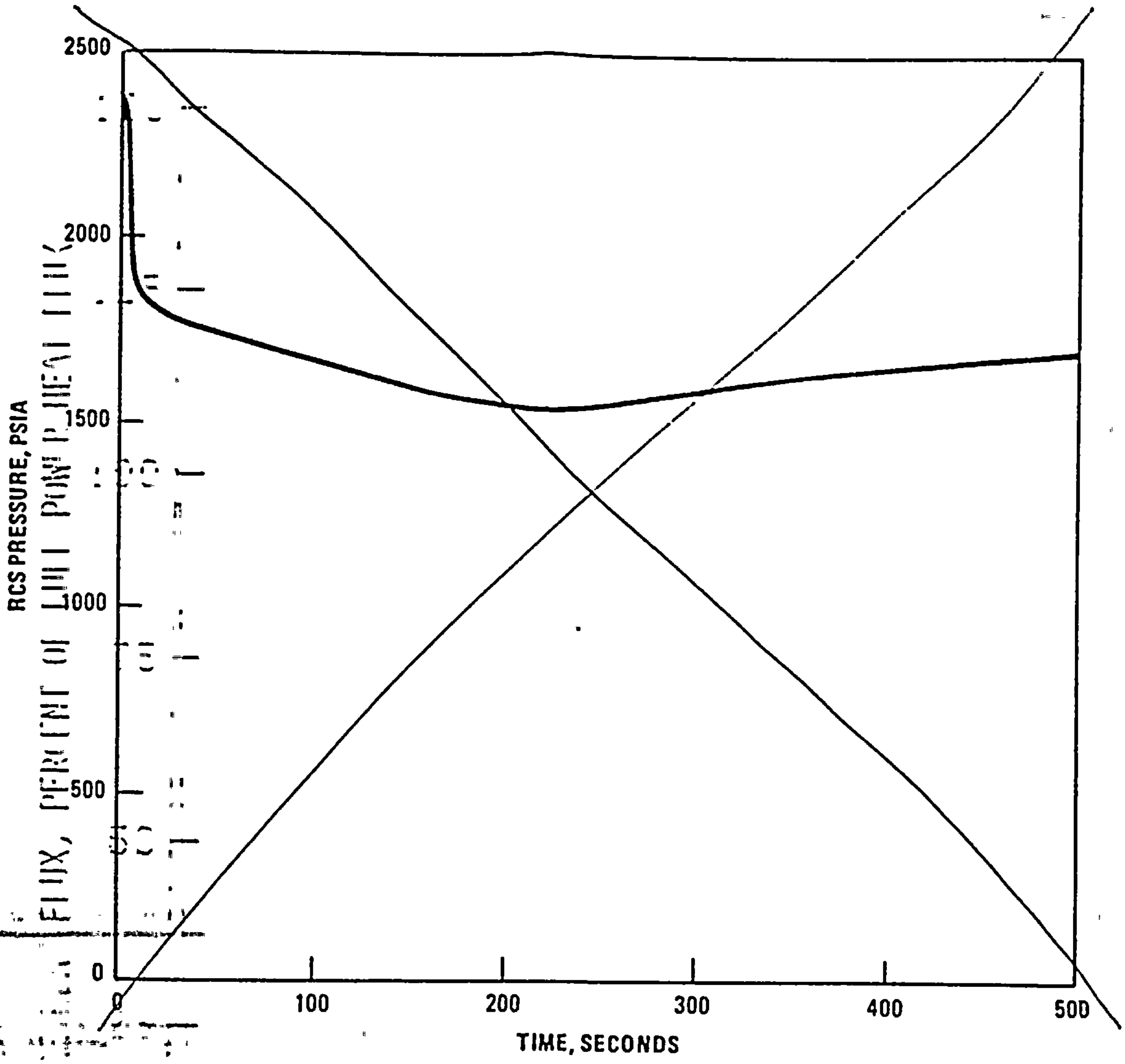
FULL POWER LARGE STEAM LINE BREAK WITH
CONCURRENT LOSS OF OFFSITE POWER
CORE HEAT FLUX VS TIME

Figure 15.1-2



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER
CORE HEAT FLUX VS TIME
Figure 15.1:1-2



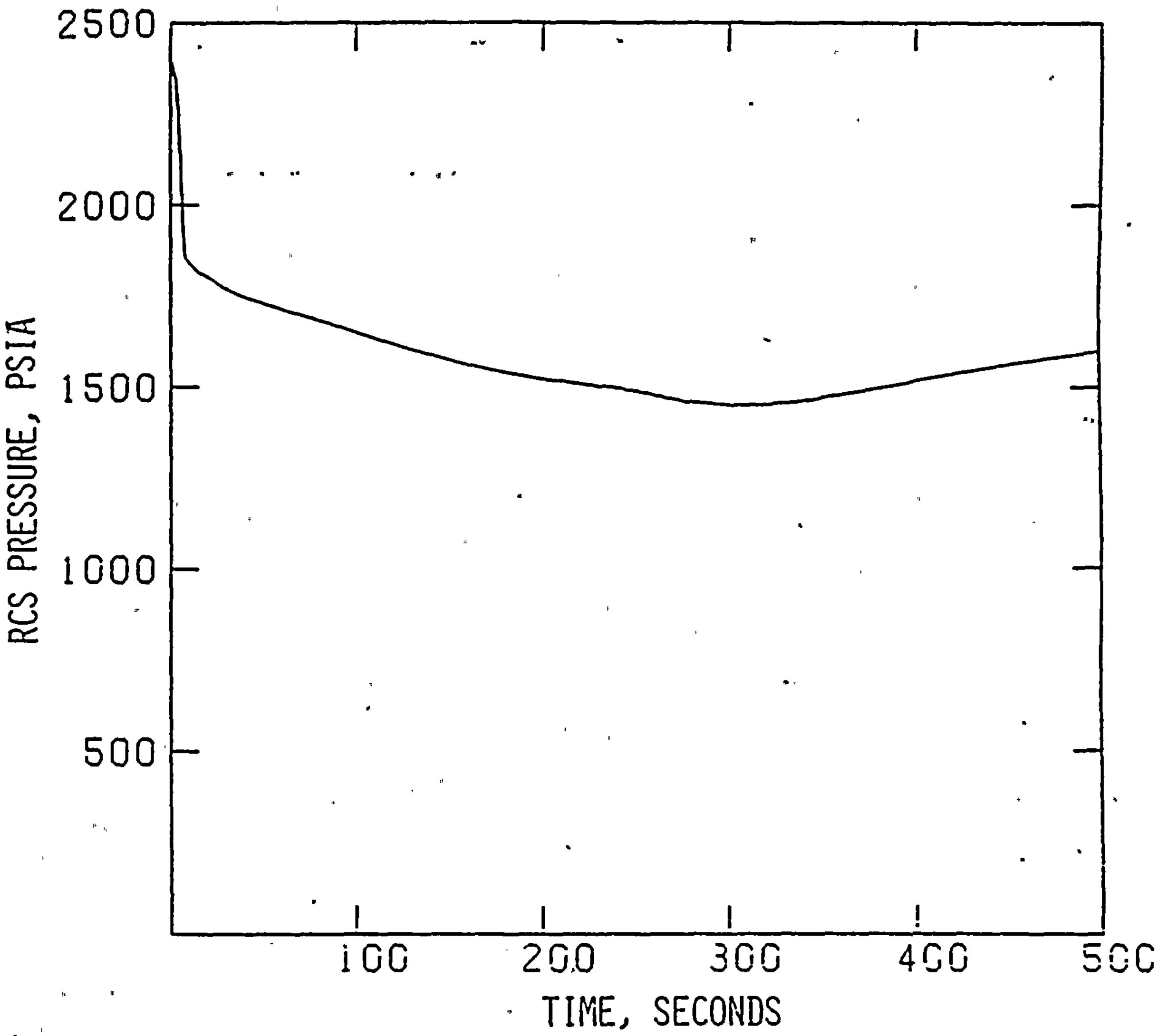
Replace with Figure 15.1.1-3

1

Palo Verde Nuclear Generating Station
FSAR

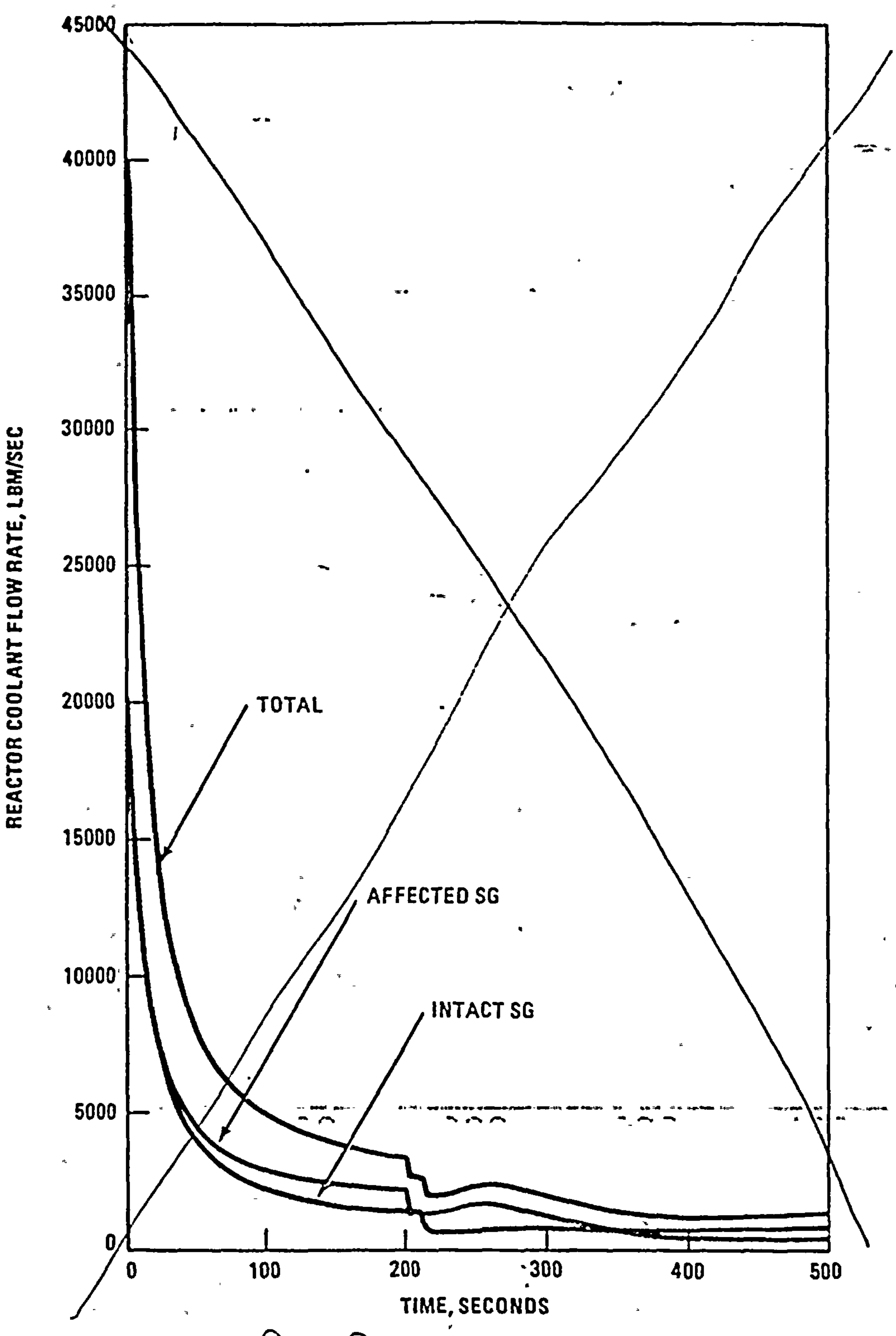
FULL POWER LARGE STEAM LINE BREAK WITH
CONCURRENT LOSS OF OFFSITE POWER
RCS PRESSURE VS TIME

Figure 15.1-3



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH
CONCURRENT LOSS OF OFFSITE POWER
RCS PRESSURE VS TIME
Figure 15.1.1-3

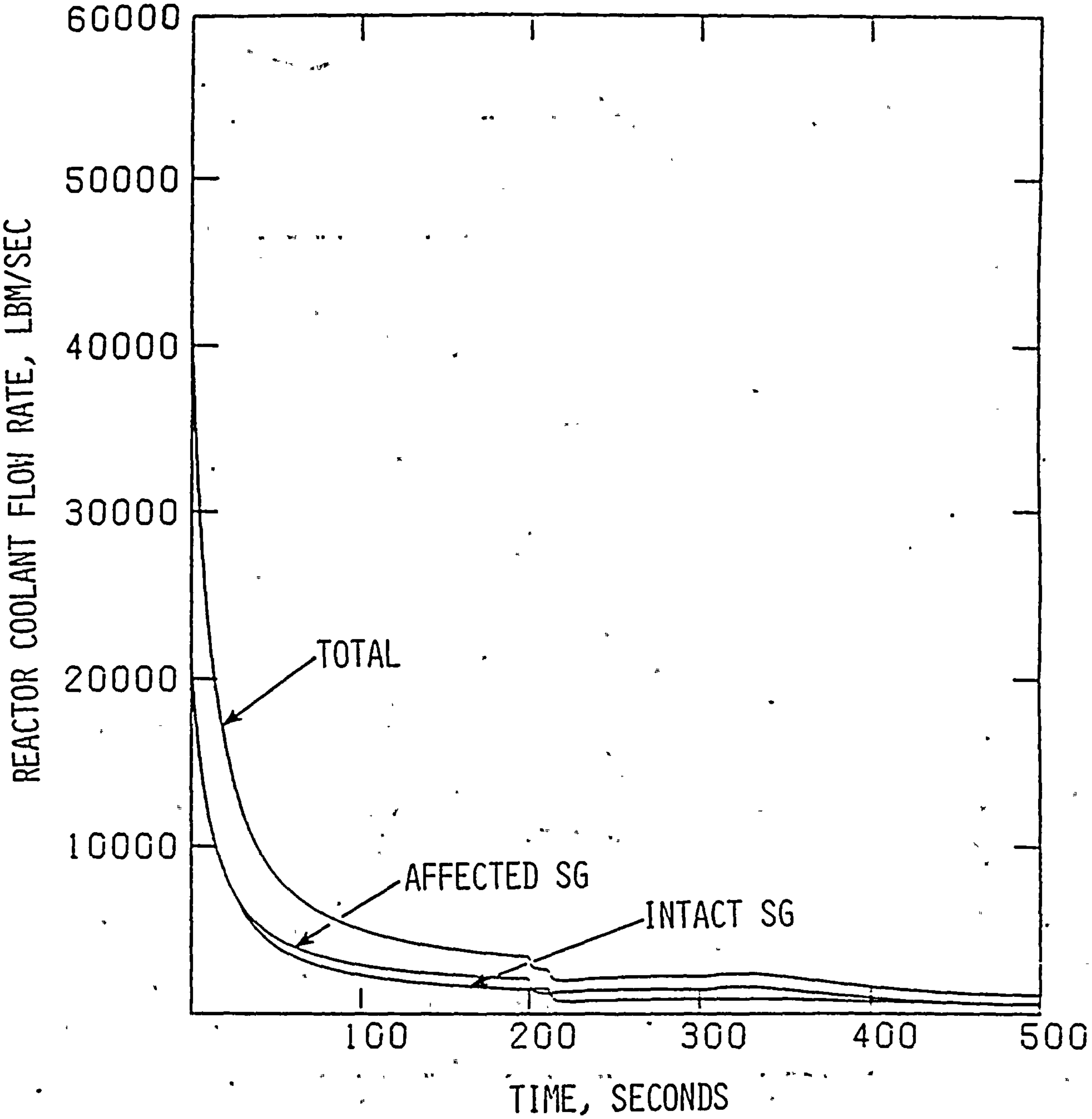


Replace with figure 15.1.1-4

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT FLOW RATE VS TIME

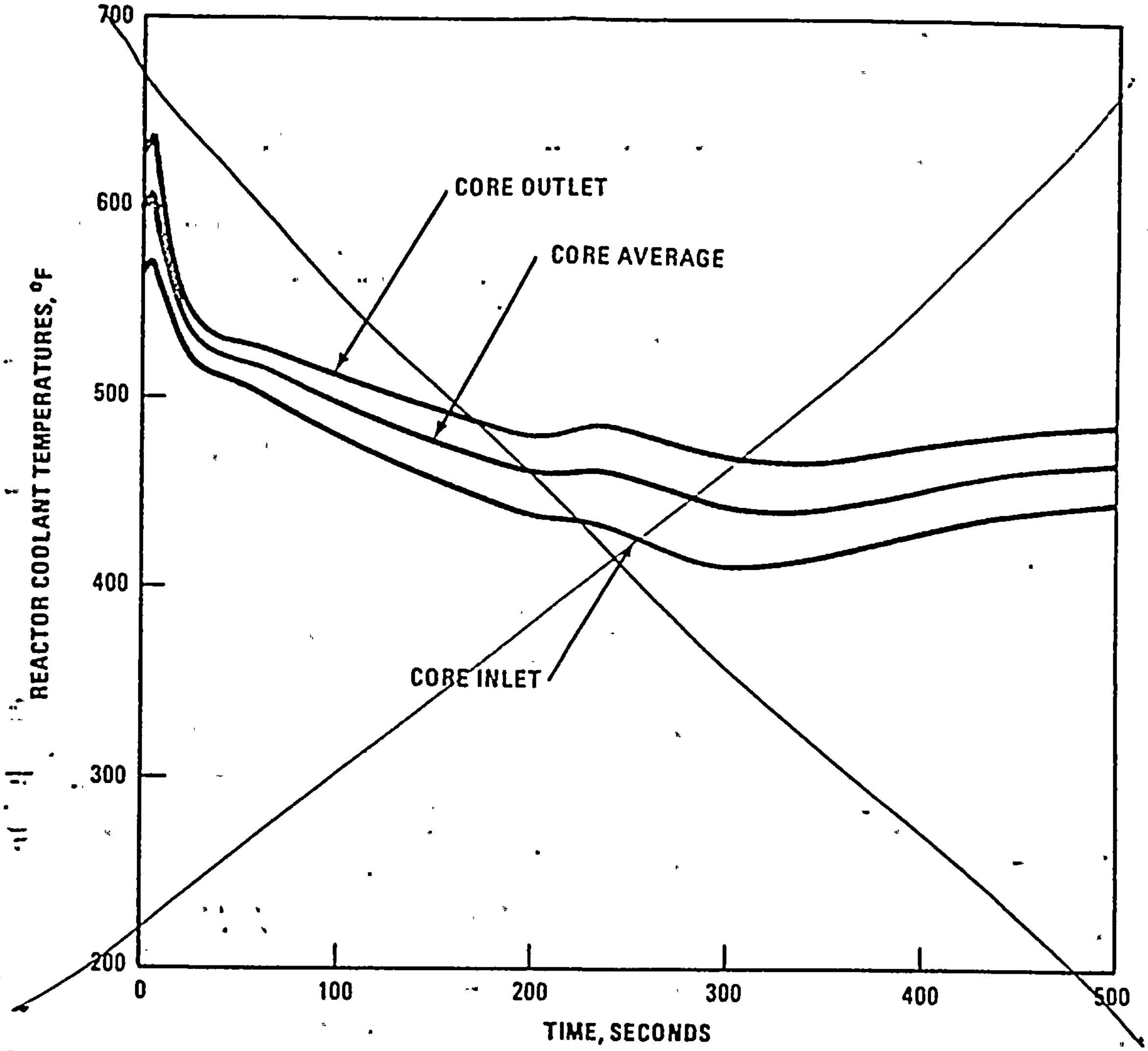
Figure 15.1-4



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT FLOW RATE VS TIME

Figure 15.1.1-4

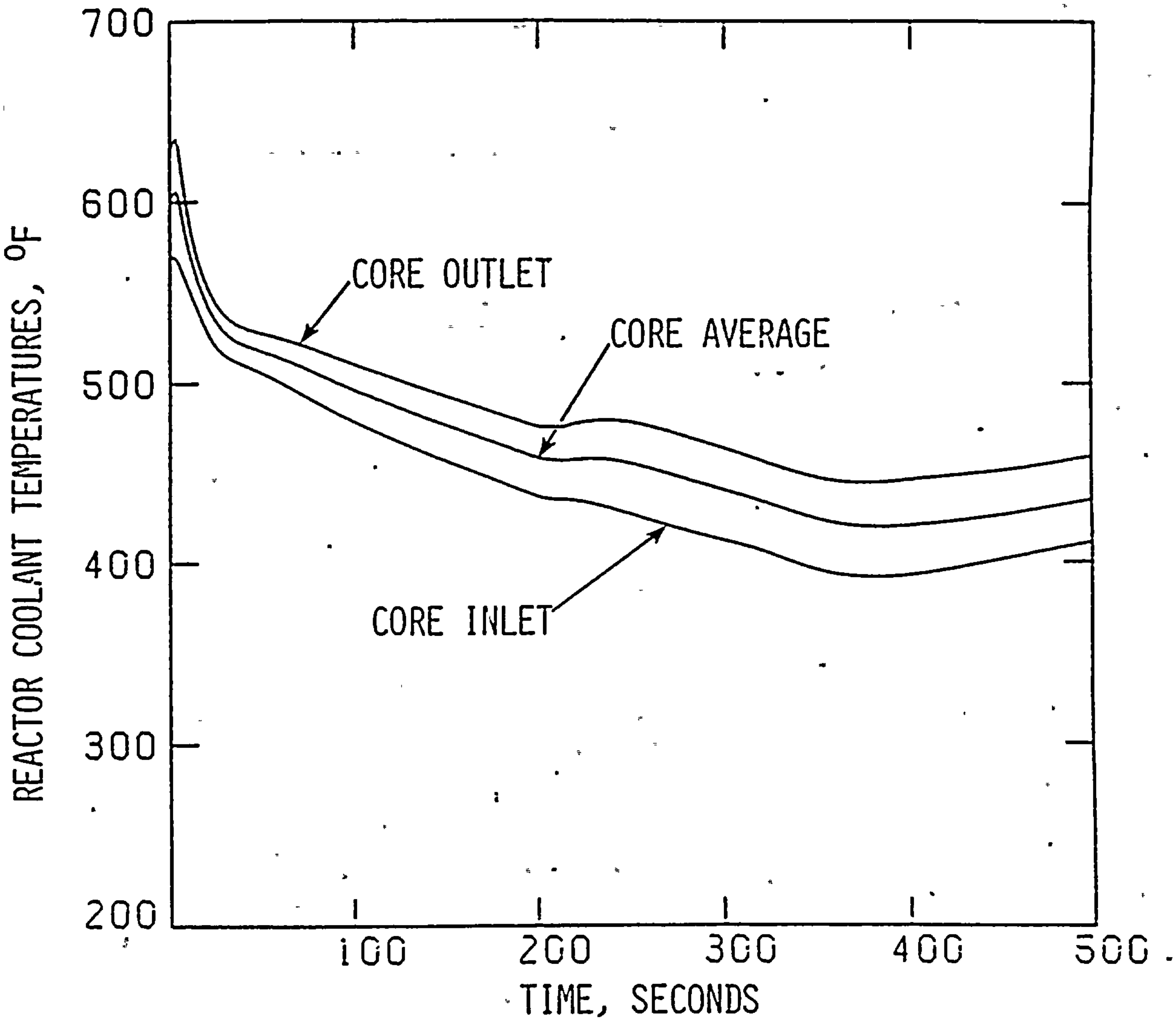


Replace with Figure 15.1.1-5

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT TEMPERATURES VS TIME (Sheet 1 of 2)

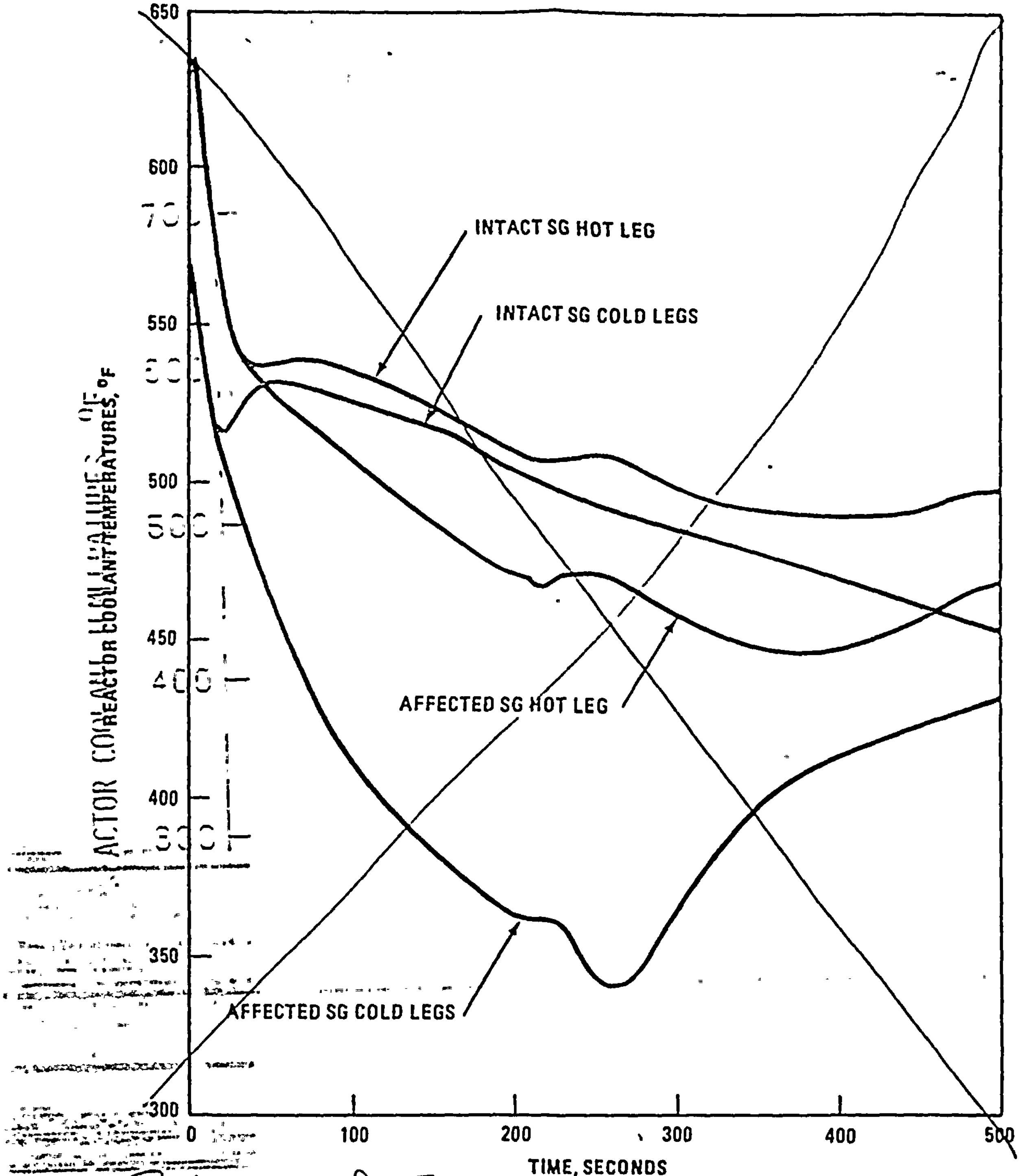
Figure 15.1-5



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT TEMPERATURES VS TIME

Figure 15.1.1-5

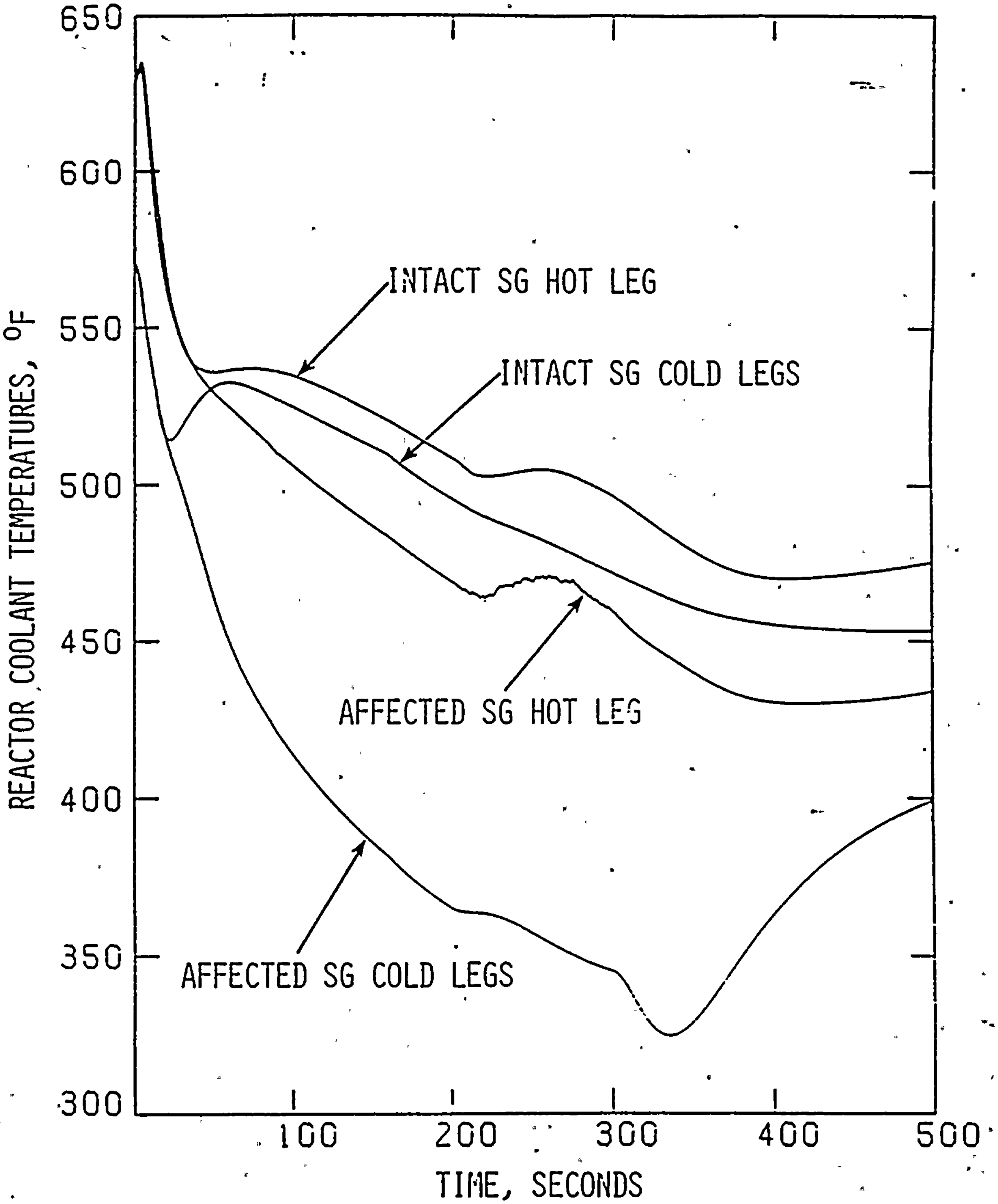


Replace with Figure 15.1.1-6

Palo Verde Nuclear Generating Station FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT TEMPERATURES VS TIME (Sheet 2 of 2)

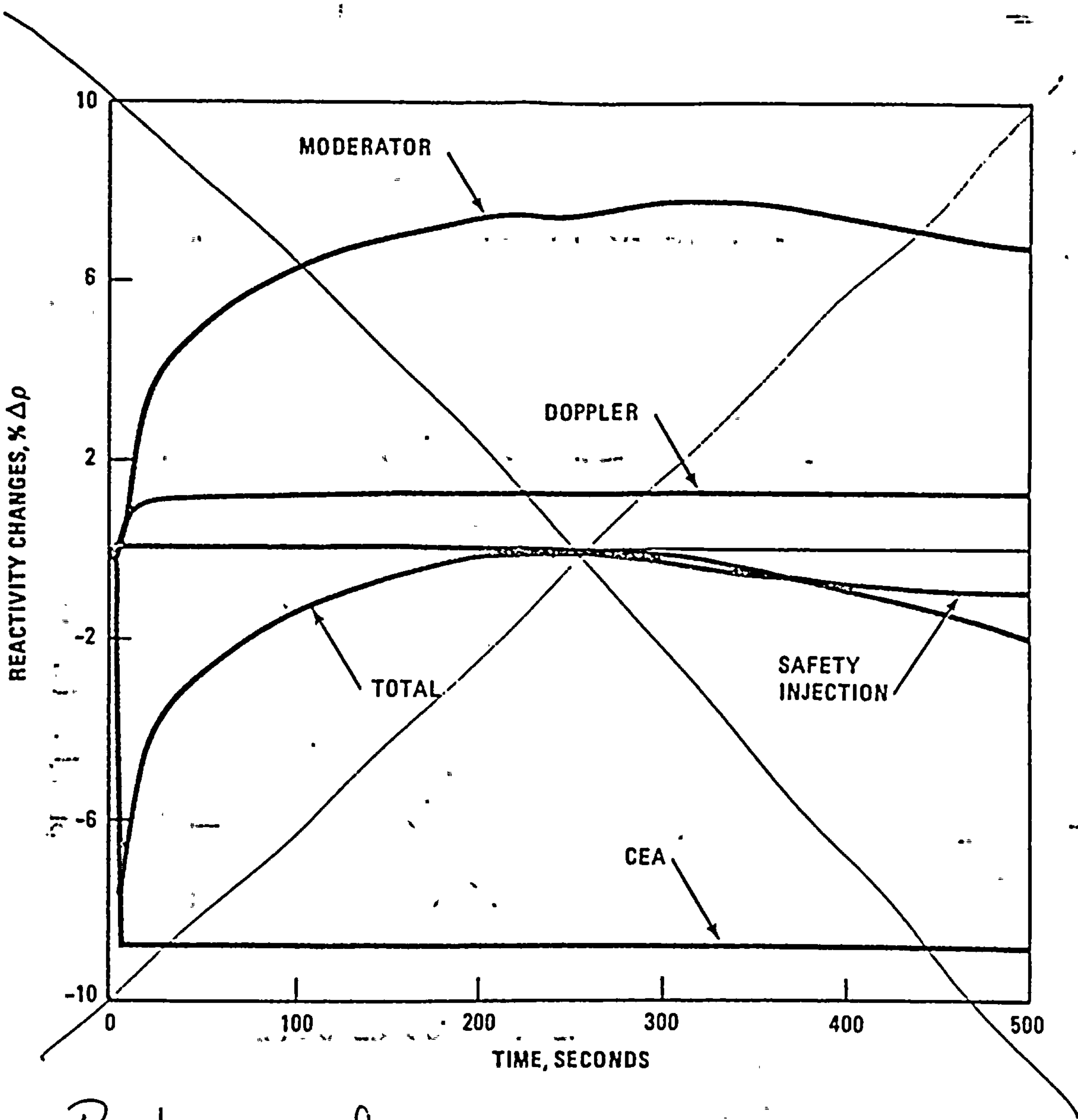
Figure 15.1-5



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT TEMPERATURES VS TIME

Figure 15.1.1-6



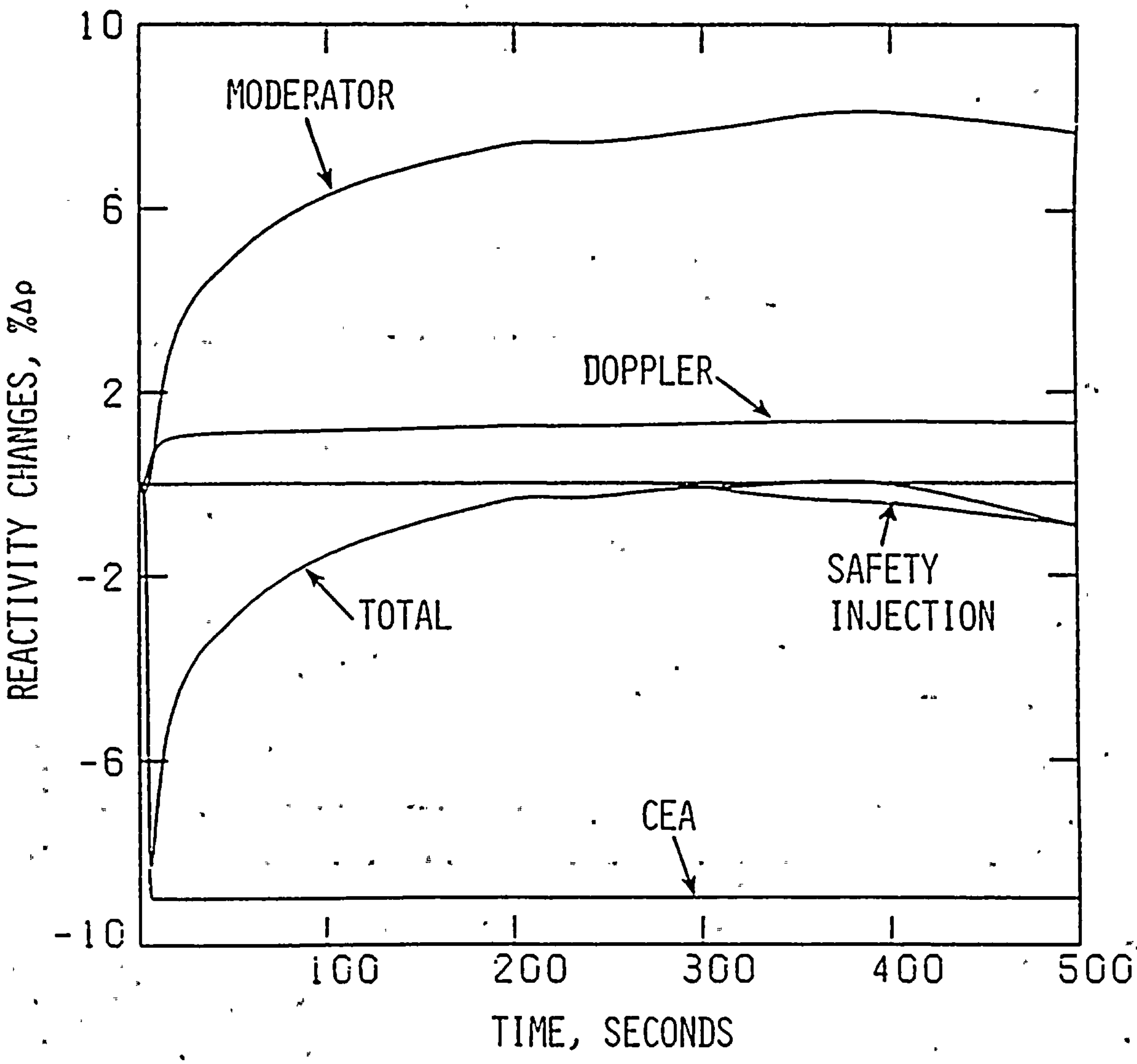
Replace with Figure 15.1.1-7

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTIVITY CHANGES VS TIME

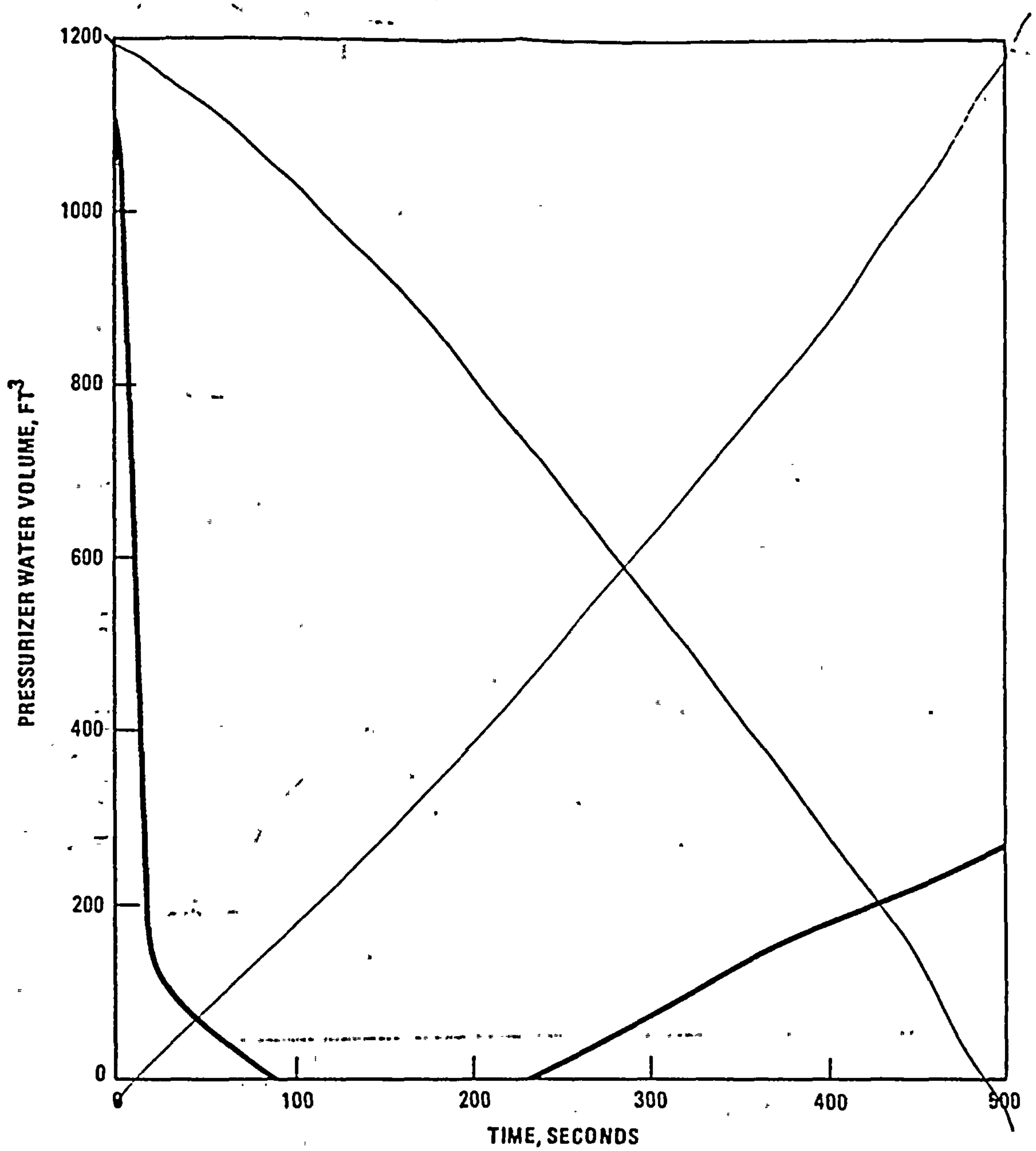
Figure 15.1-6



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTIVITY CHANGES VS TIME

Figure 15.1.1-7



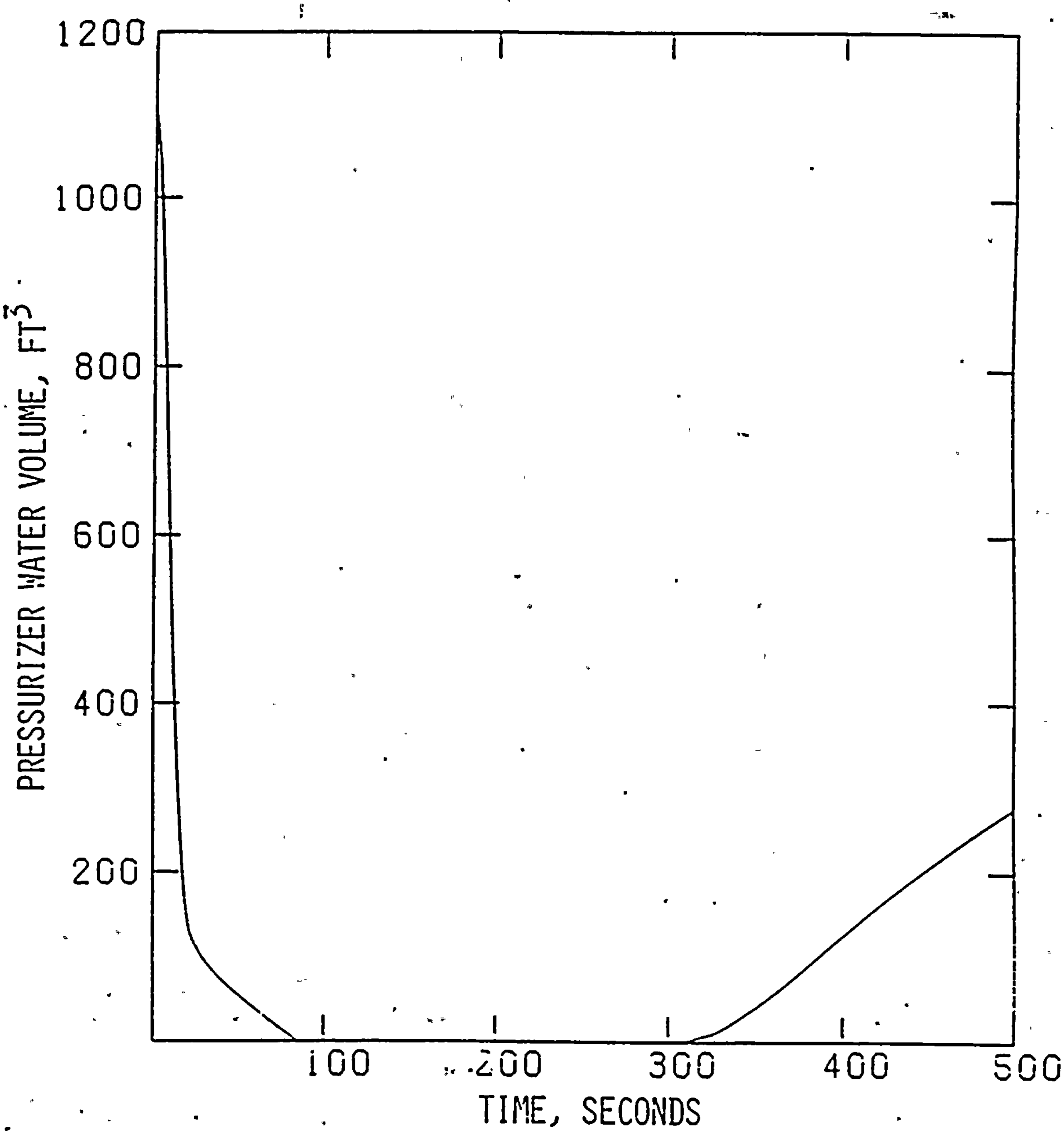
Replace WITH Figure 15.1.1-8

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER PRESSURIZER WATER VOLUME VS TIME

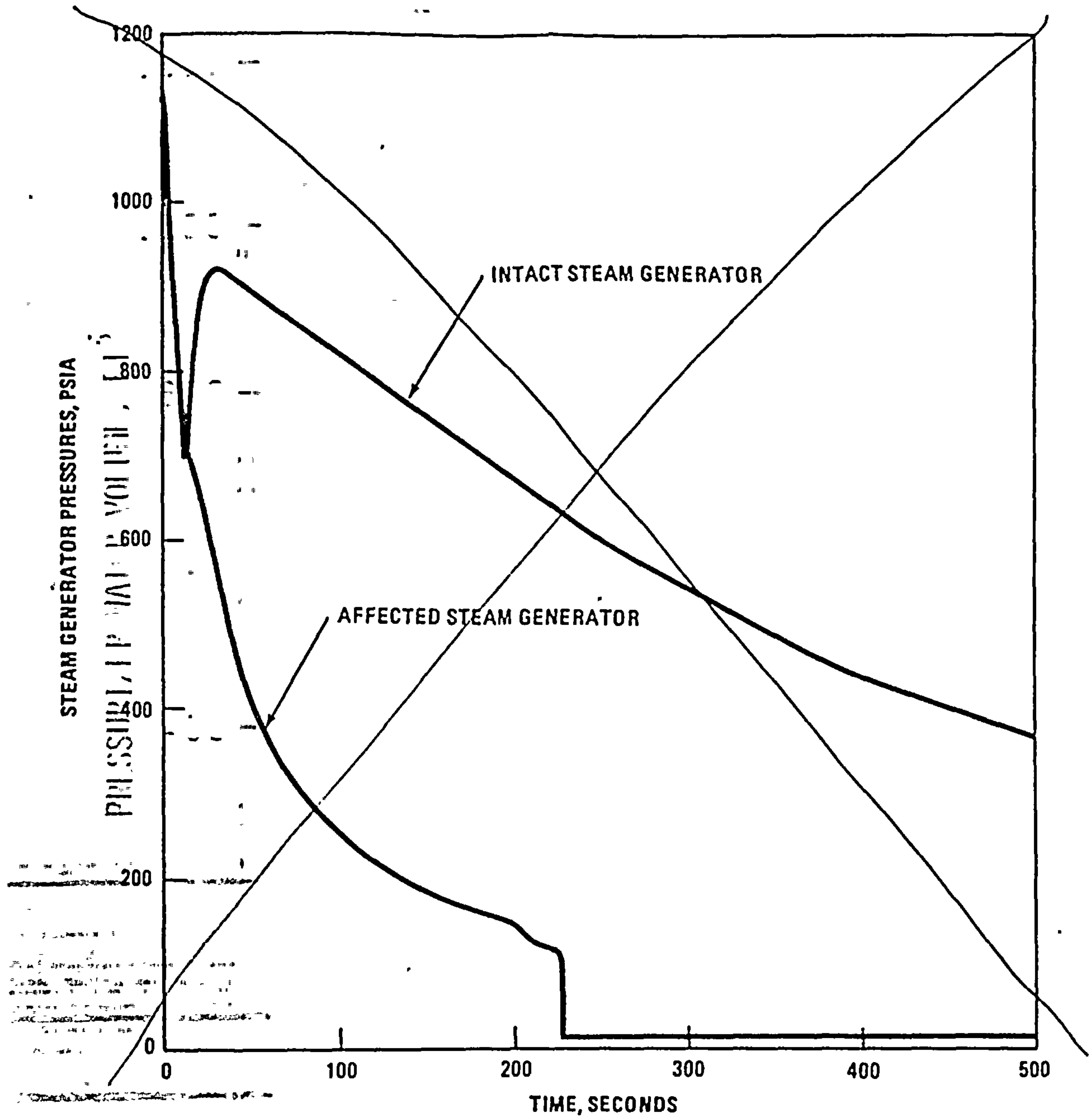
Figure 15.1-7



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER PRESSURIZER WATER VOLUME VS TIME

Figure 15.1.1-8



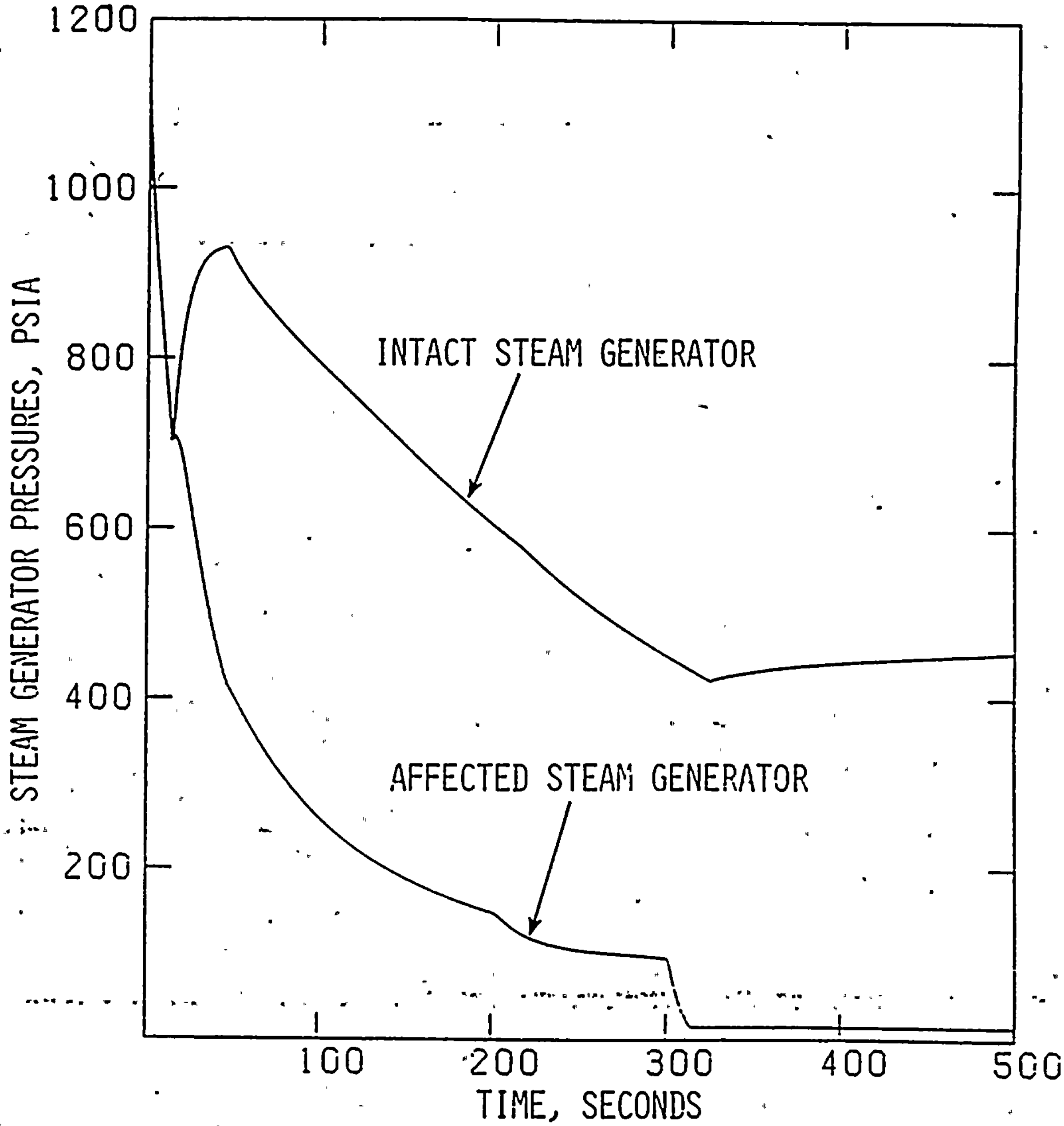
Replace with figure 15.1.1-9

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR PRESSURES VS TIME

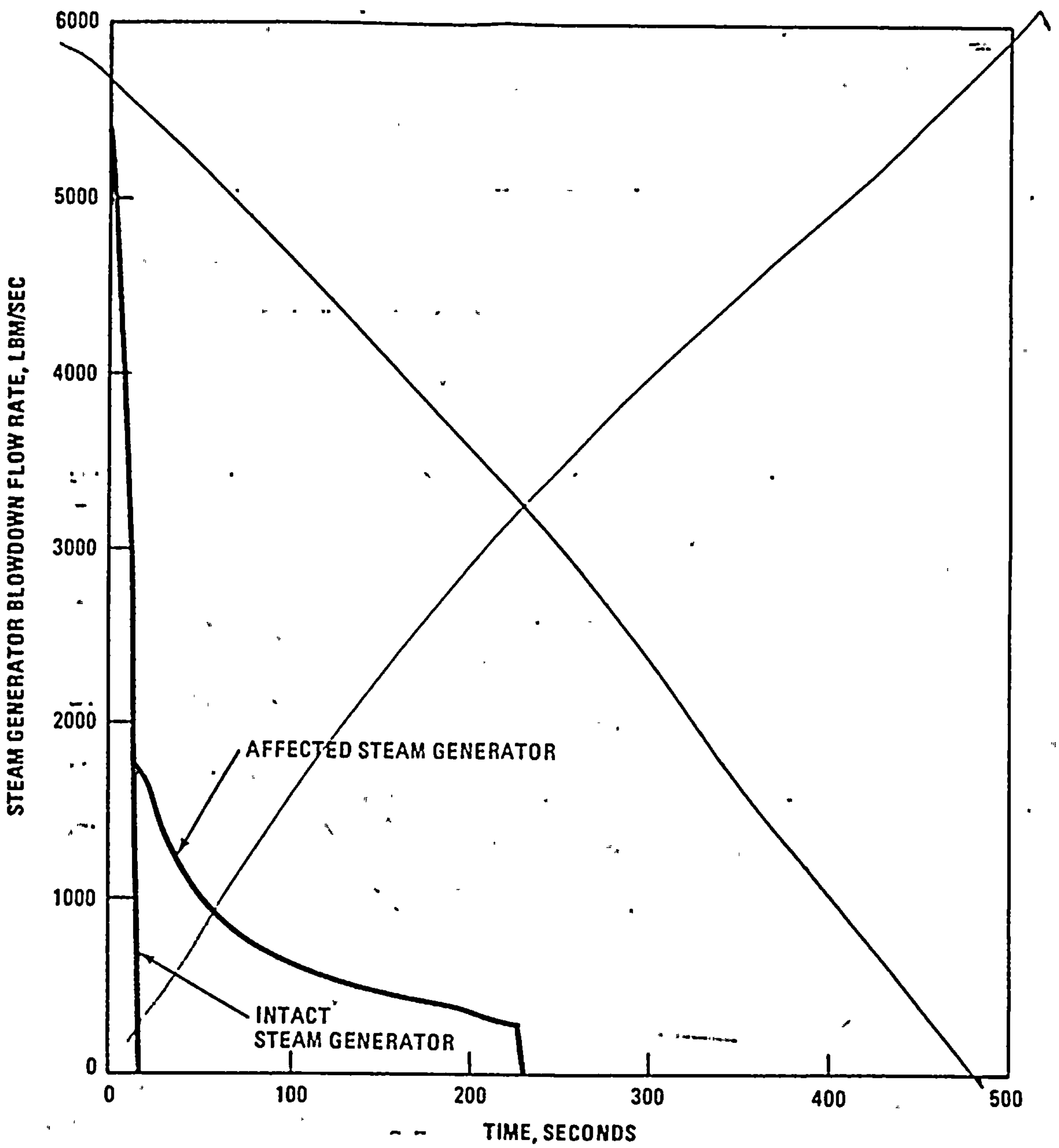
Figure 15.1-8



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR PRESSURES VS TIME

Figure 15.1.1-9



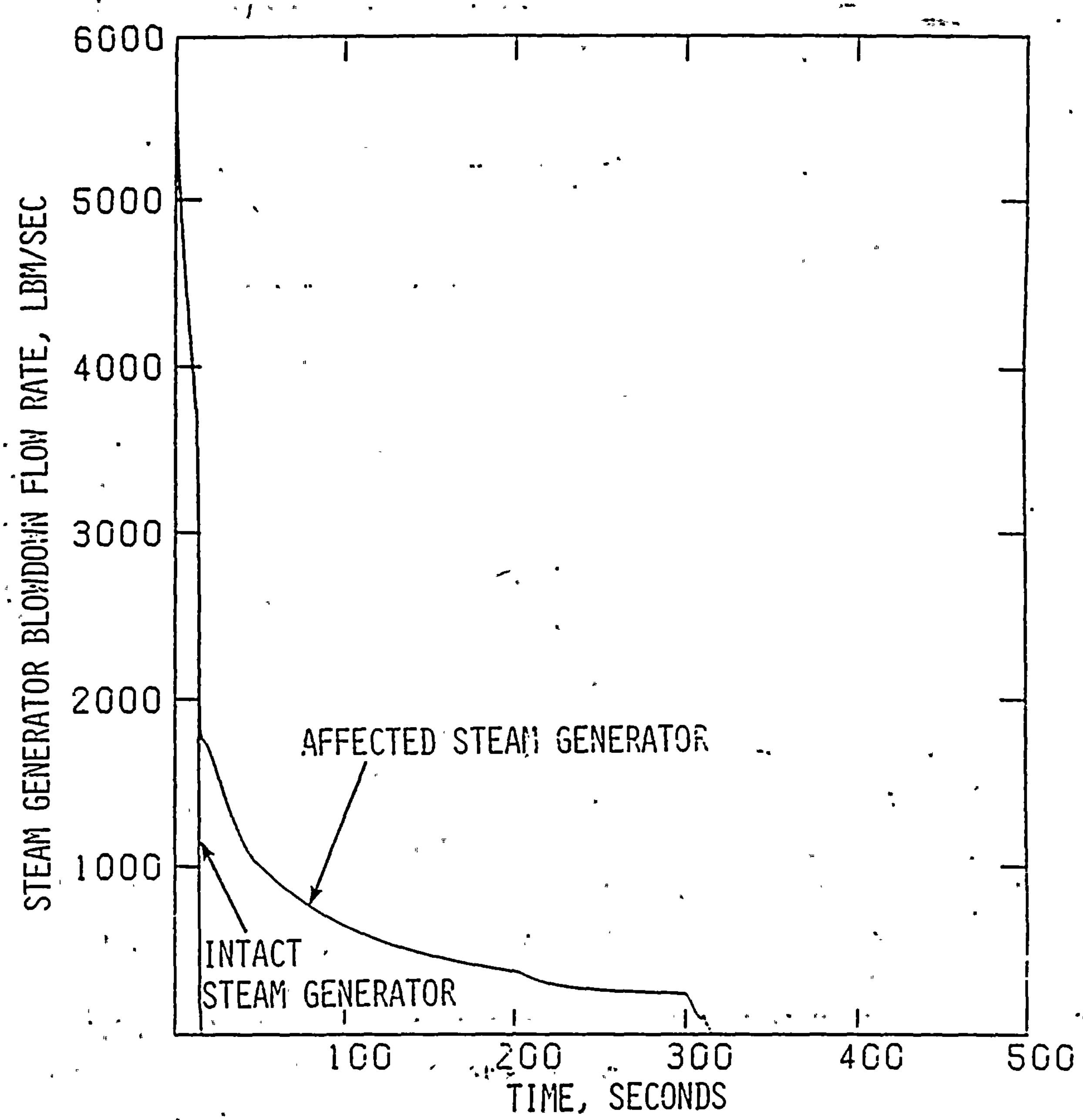
Replace with figure 15.1.1-10

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR BLOWDOWN RATES VS TIME

Figure 15.1-9

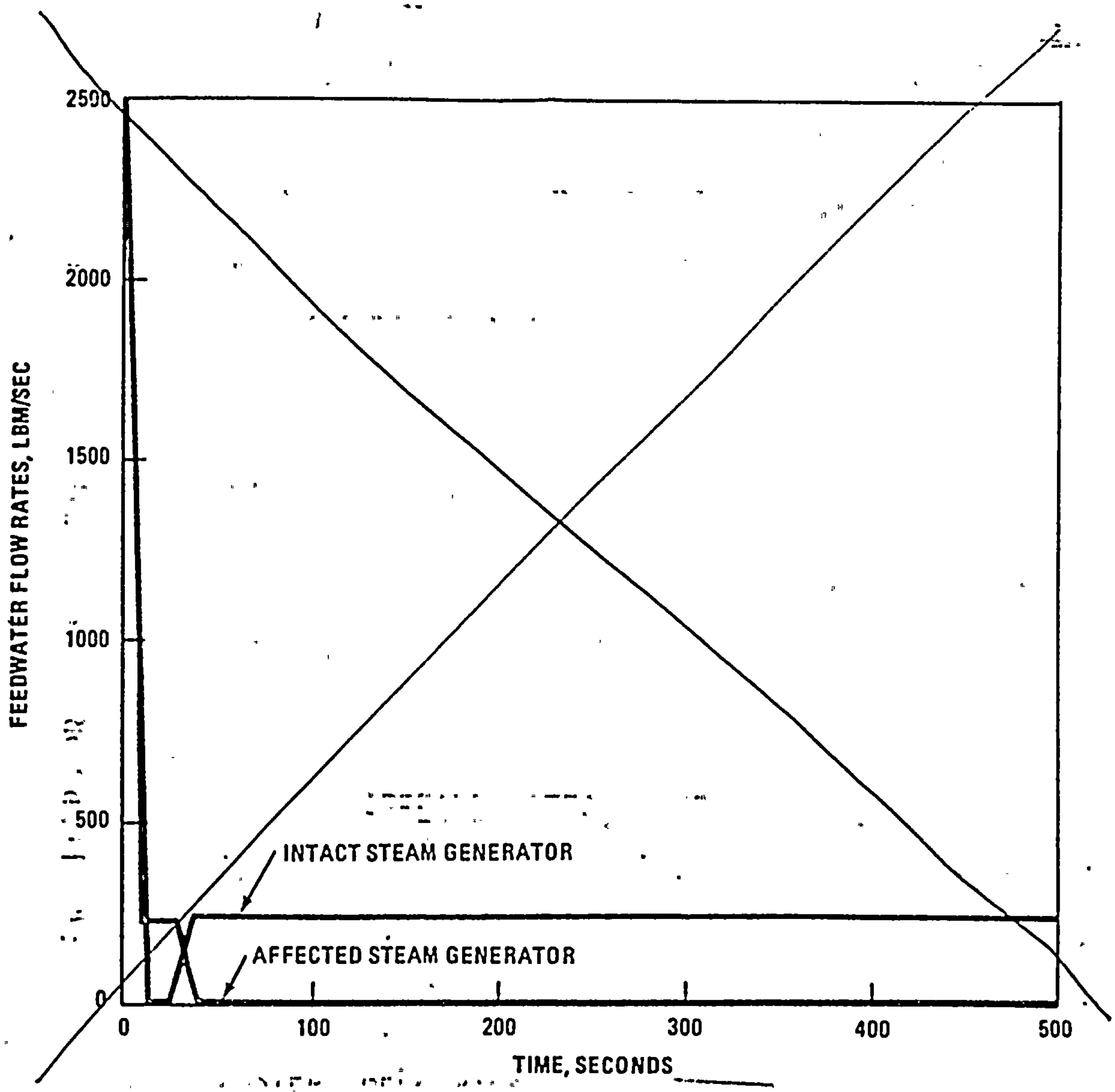
1



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR BLOWDOWN RATES VS TIME

Figure 15.1.1-10



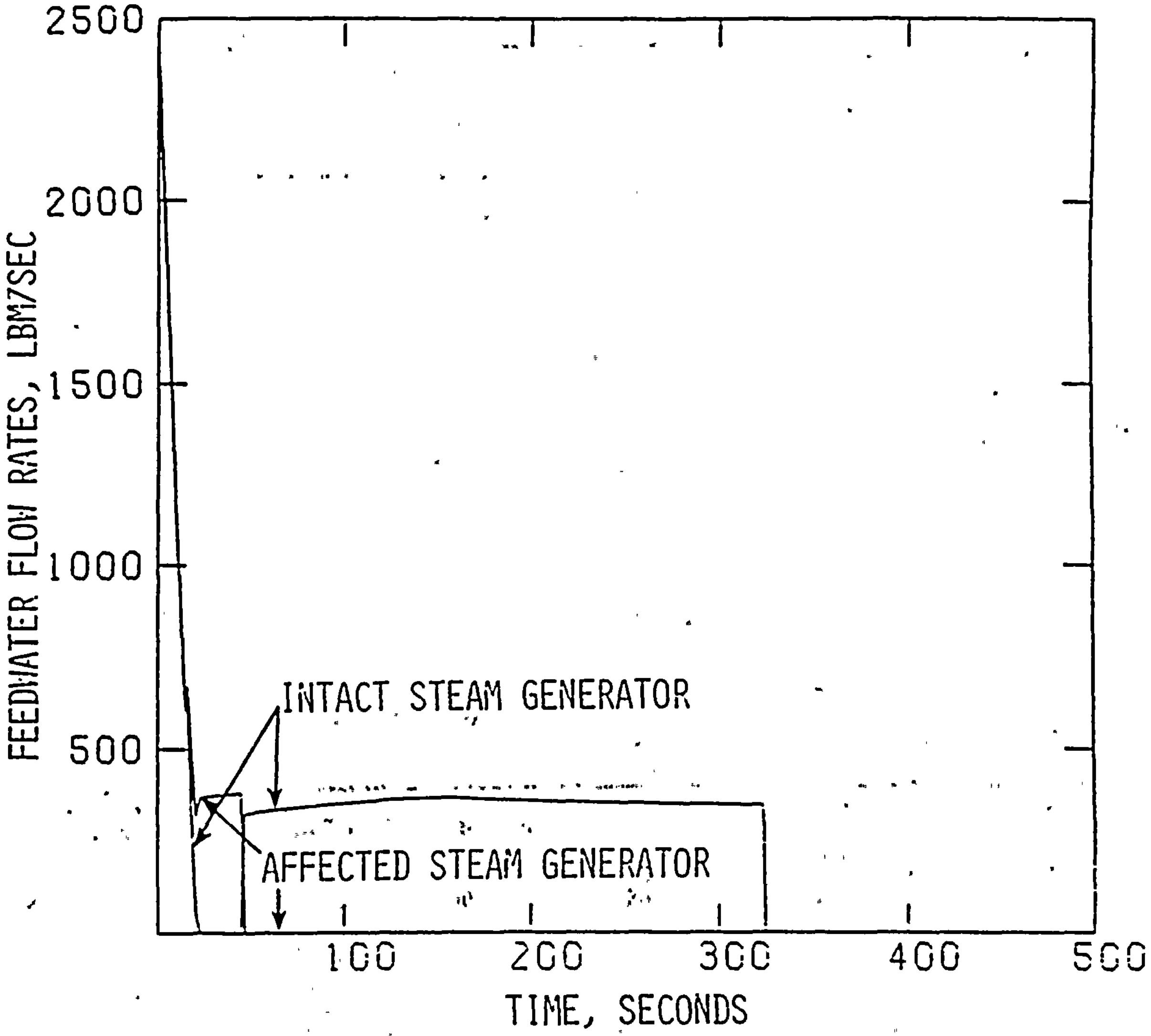
Replace with Figure 15.1.1-11

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER FEEDWATER FLOW RATES VS TIME

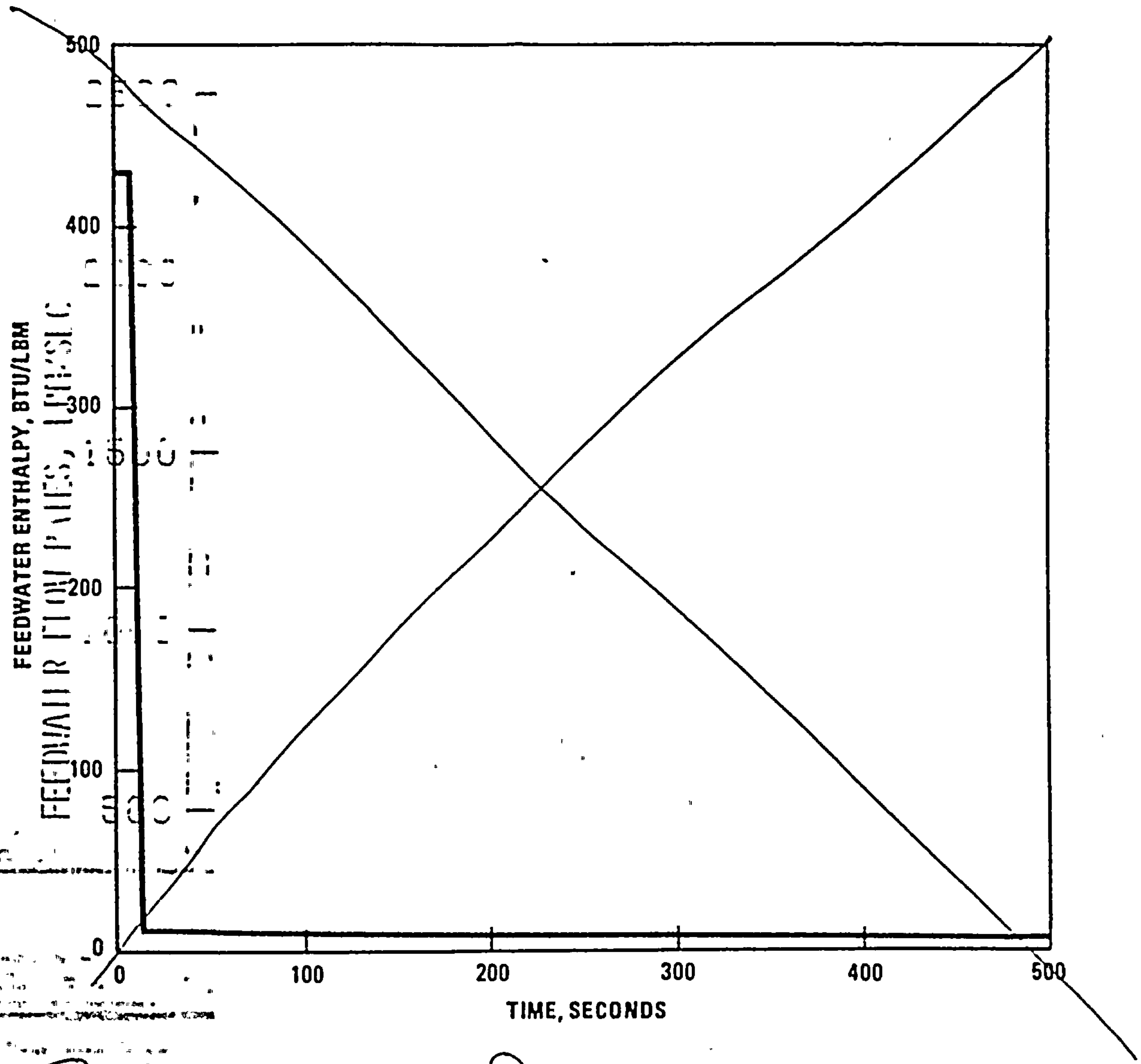
Figure 15.1-10



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER FEEDWATER FLOW RATES VS TIME

Figure 15.1.1-11



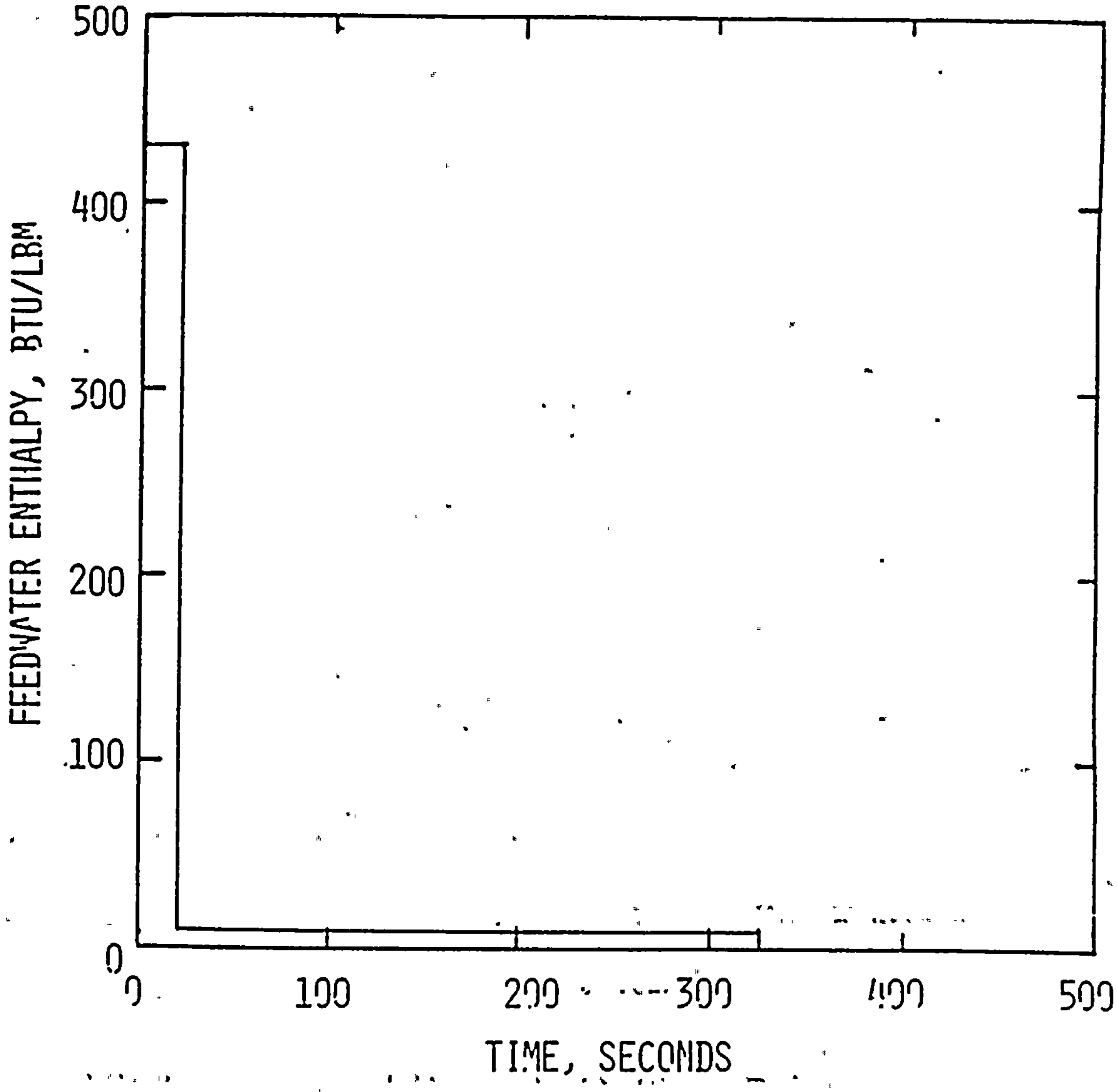
Replace WITH figure 15.1.1-12

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE BREAK FEEDWATER ENTHALPY VS TIME

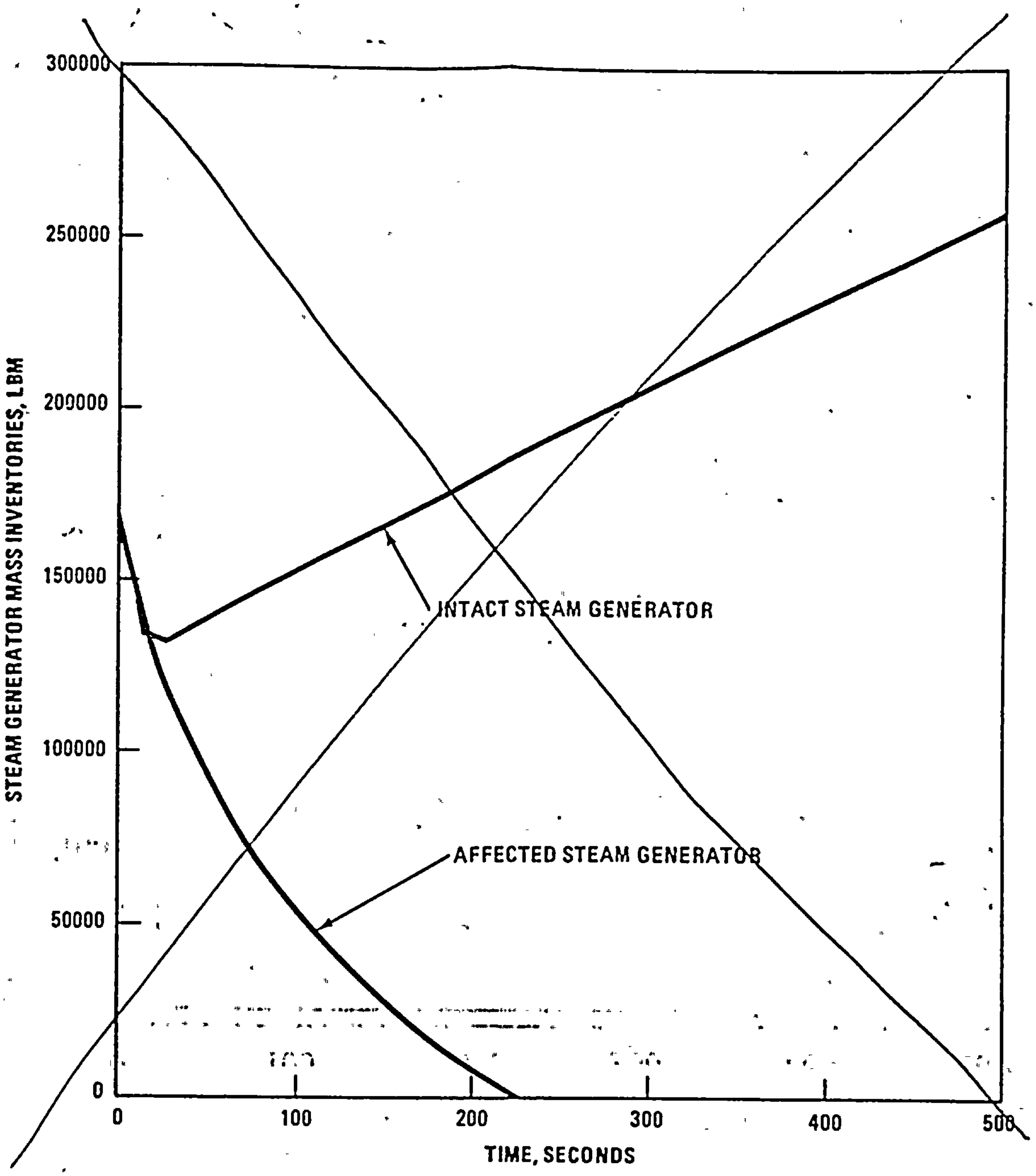
Figure 15.1-11



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER FEEDWATER ENTHALPY VS TIME

Figure 15.1.1-12



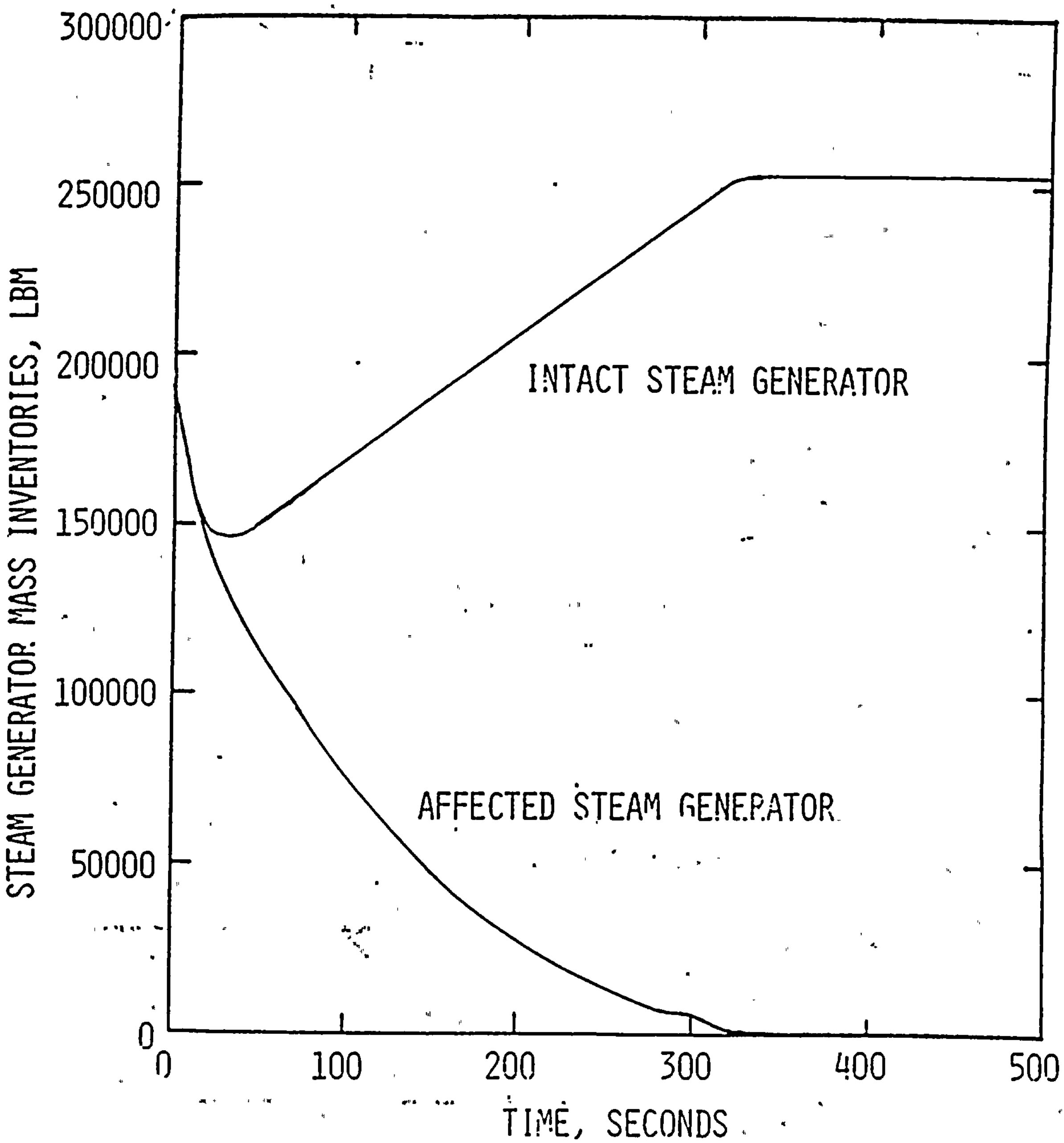
Replace WITH Figure 15.1.1-13

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR MASS INVENTORIES VS TIME

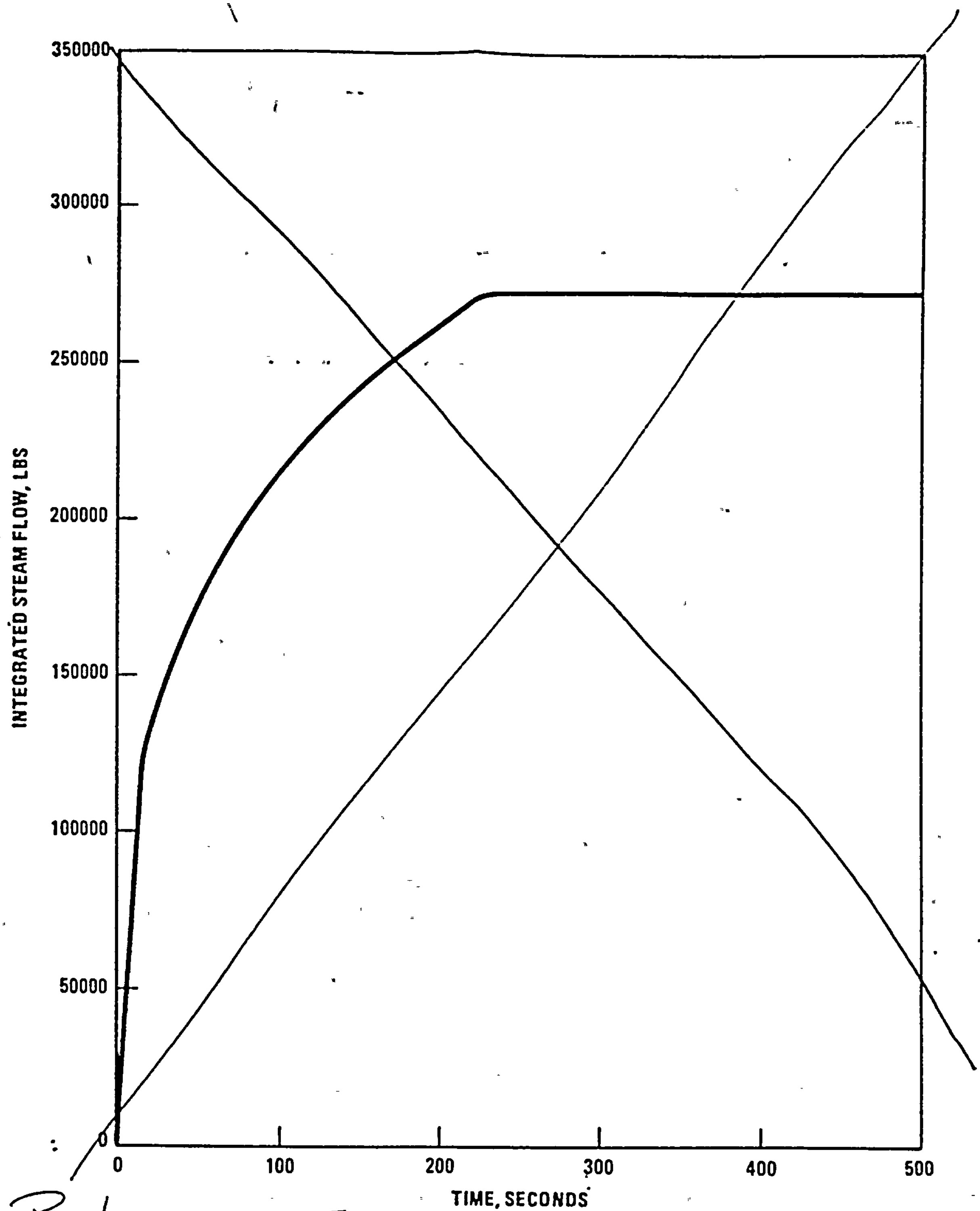
Figure 15.1-12



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR MASS INVENTORIES VS TIME

Figure 15.1.1-13



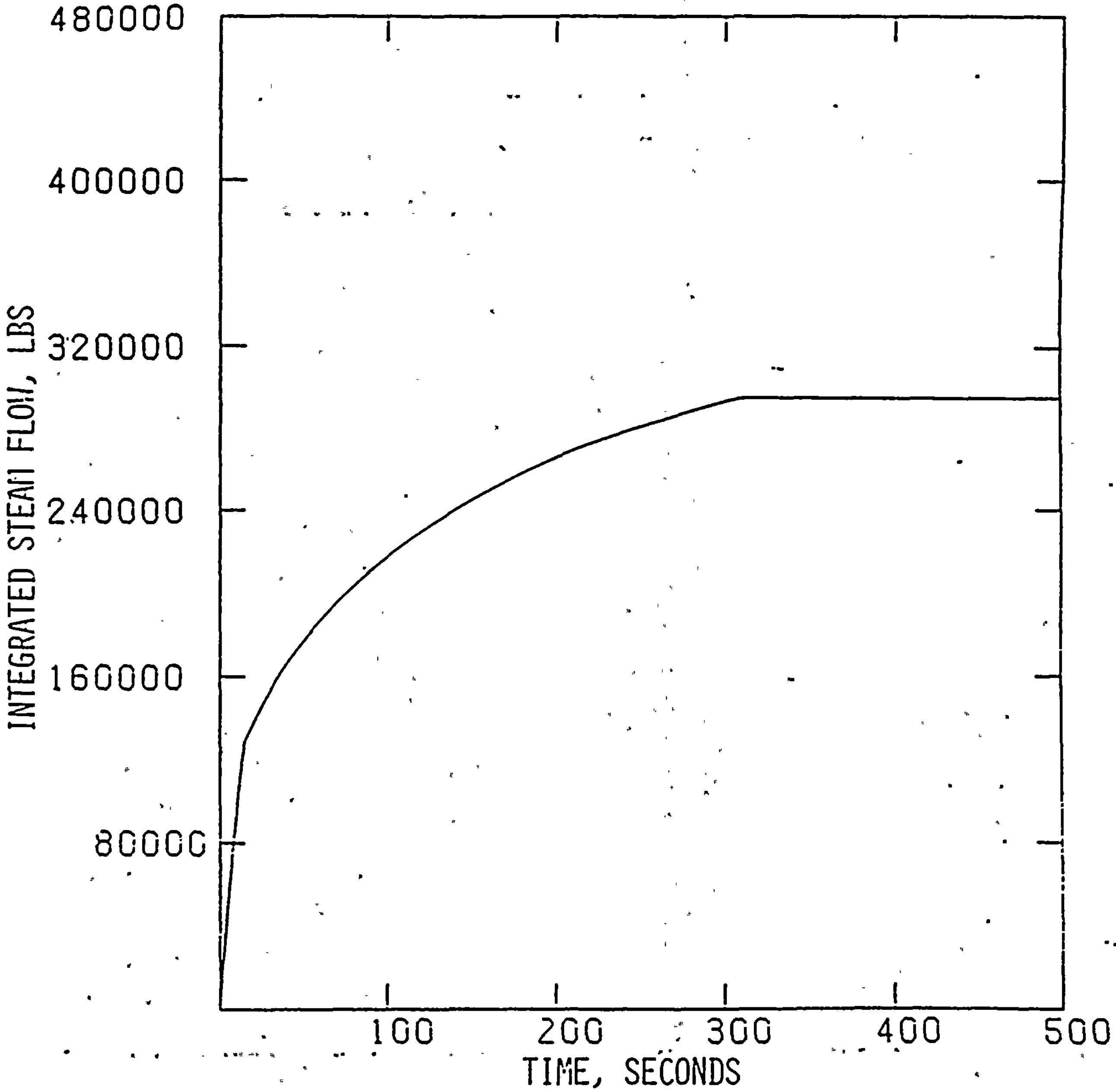
14

Replace WITH Figure 15.1.1-14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER INTEGRATED STM MASS RELEASE THRU BREAK VS TIME

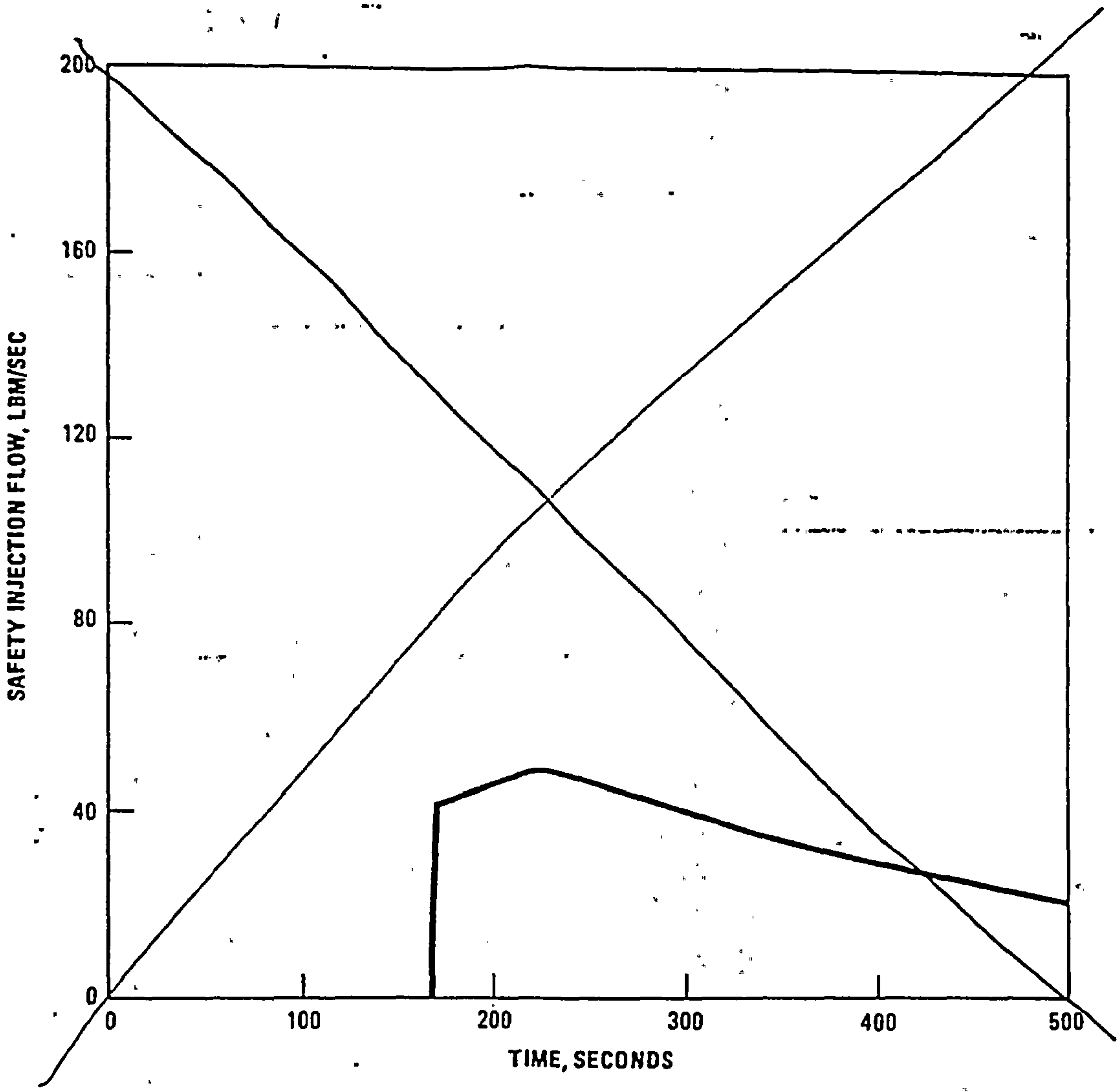
Figure 15.1-13



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER INTEGRATED STM MASS RELEASE THRU BREAK VS TIME

Figure 15.1.1-14

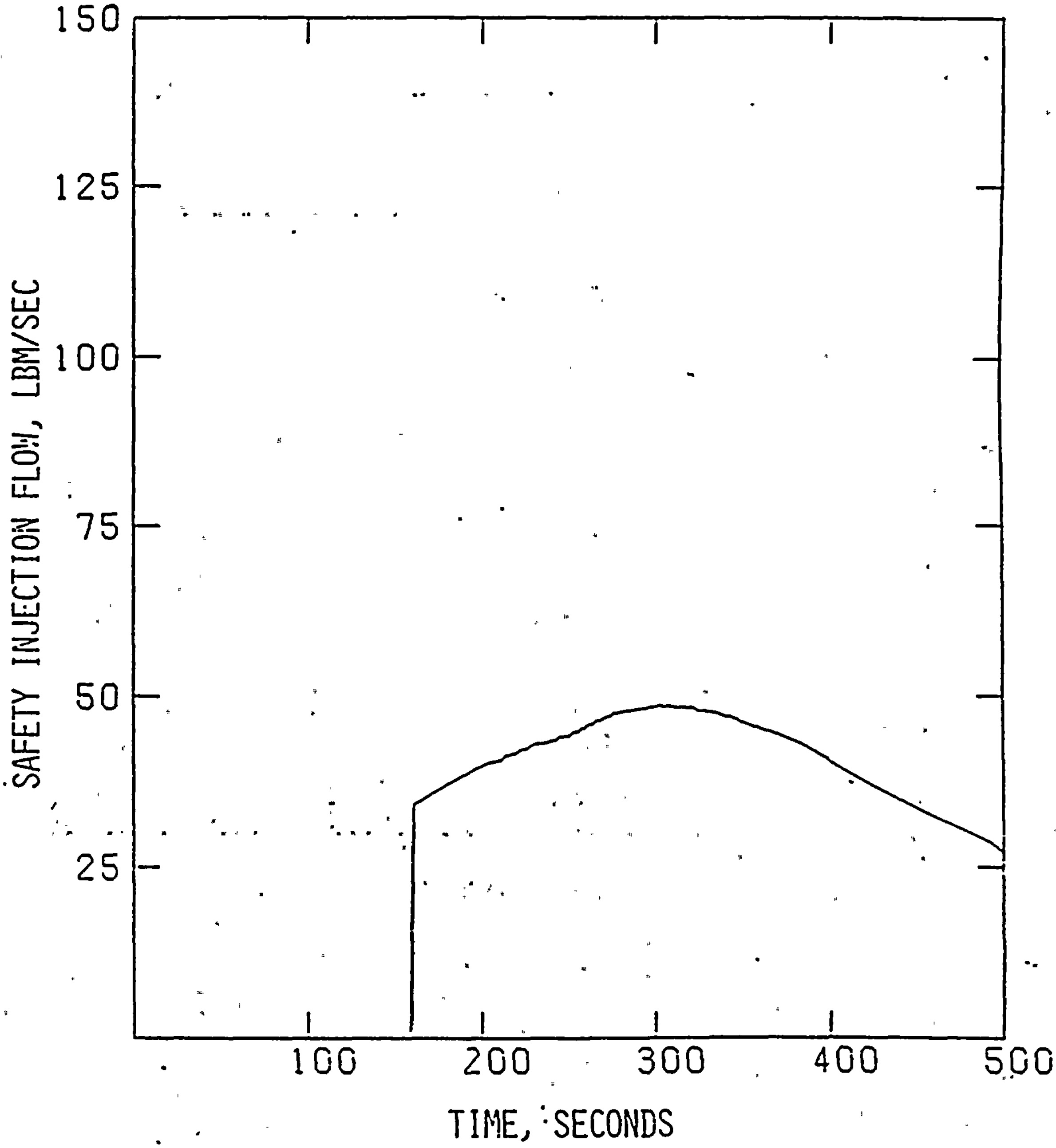


Replace WITH Figure 15.1.1-15

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH
CONCURRENT LOSS OF OFFSITE POWER
SAFETY INJECTION FLOW VS TIME

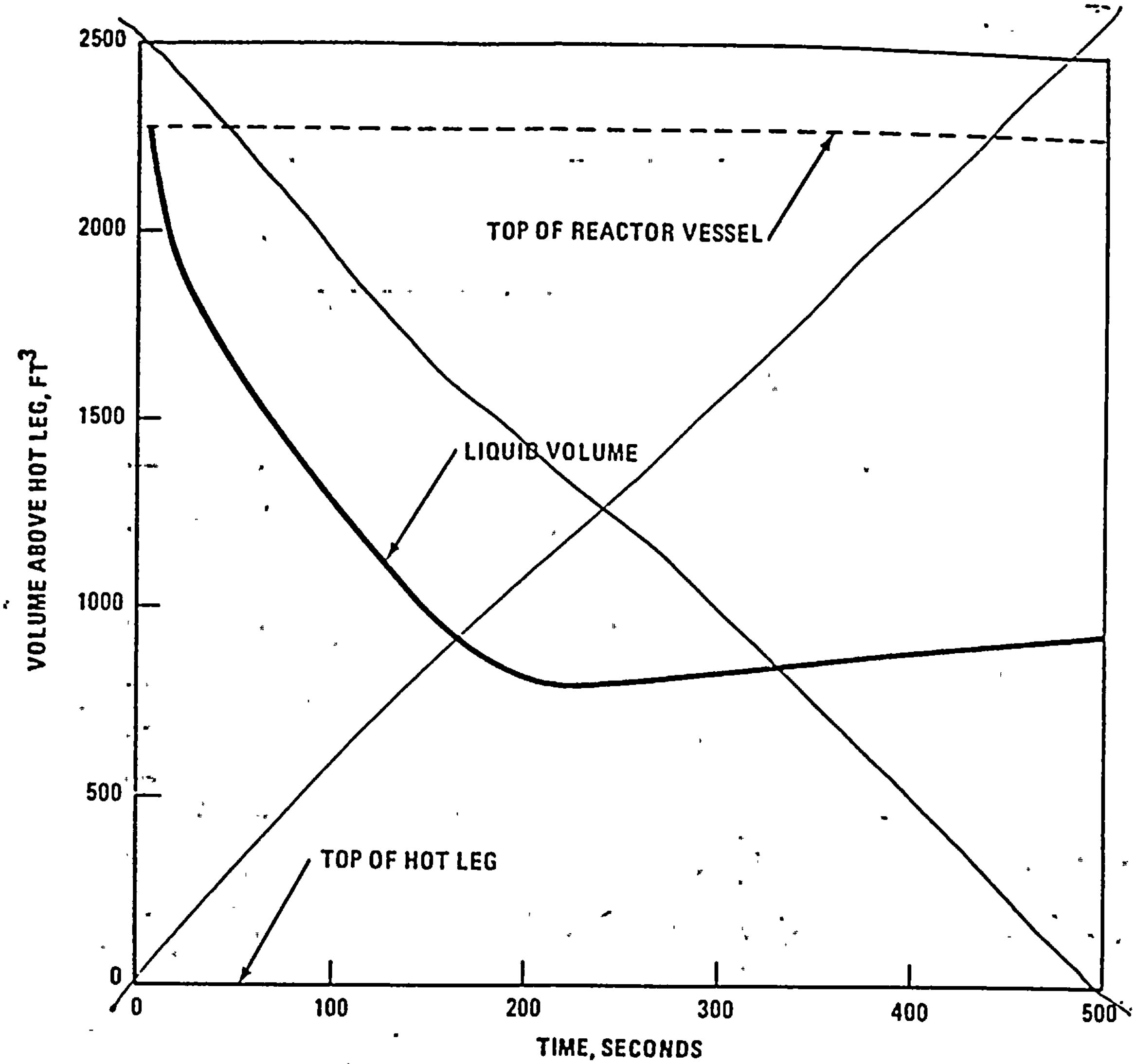
Figure 15.1-14



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER SAFETY INJECTION FLOW VS TIME

Figure 15.1.1-15



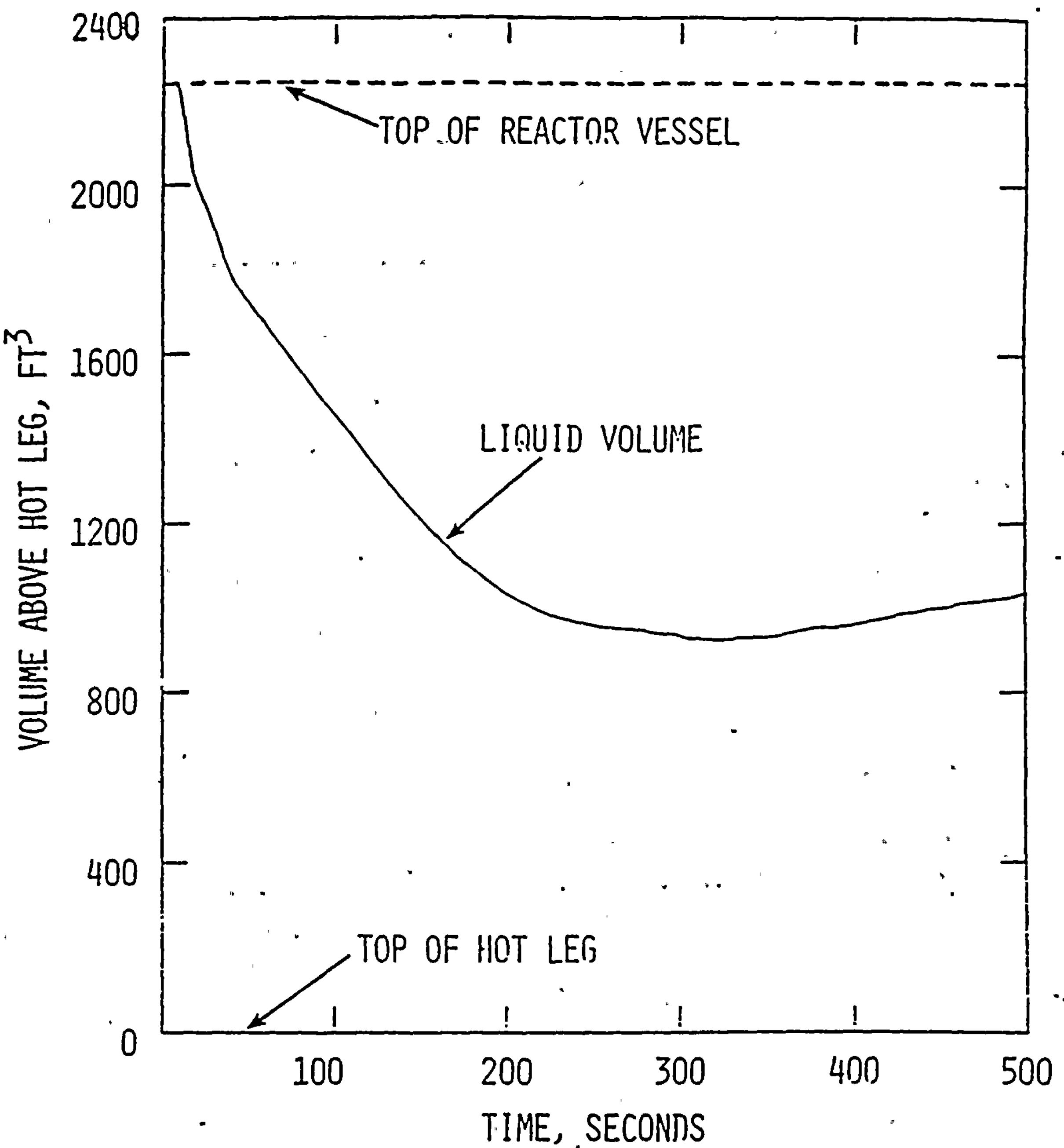
Replace with figure 15.1.1-16

14

Palo Verde Nuclear Generating Station
FSAR

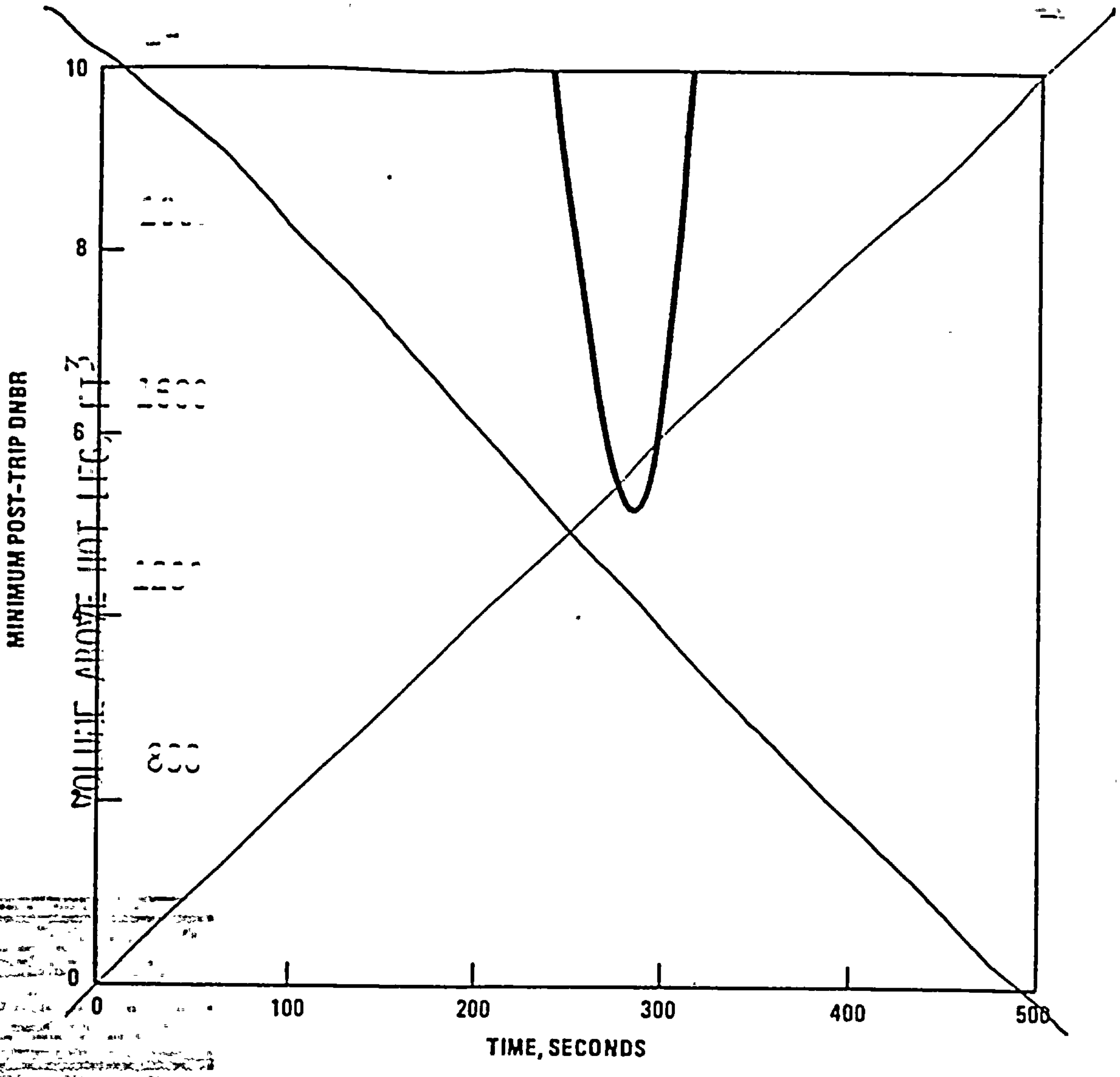
FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR VESSEL LIQUID VOLUME VS TIME

Figure 15.1-15



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR VESSEL LIQUID VOLUME VS TIME
Figure 15.1.1-16



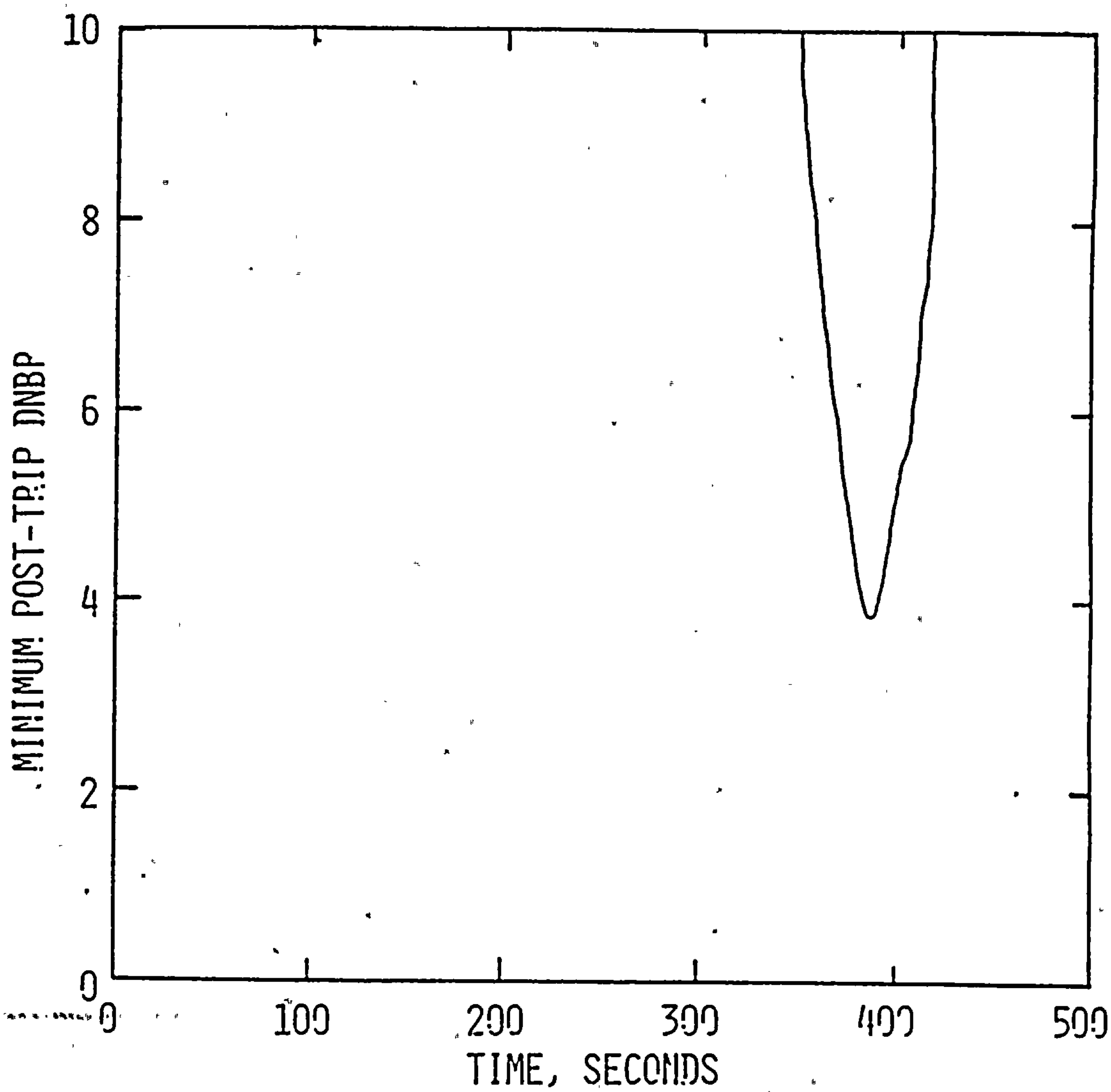
Replace WITH figure 15.1.1-17

14

Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER MINIMUM POST-TRIP DNBR VS TIME

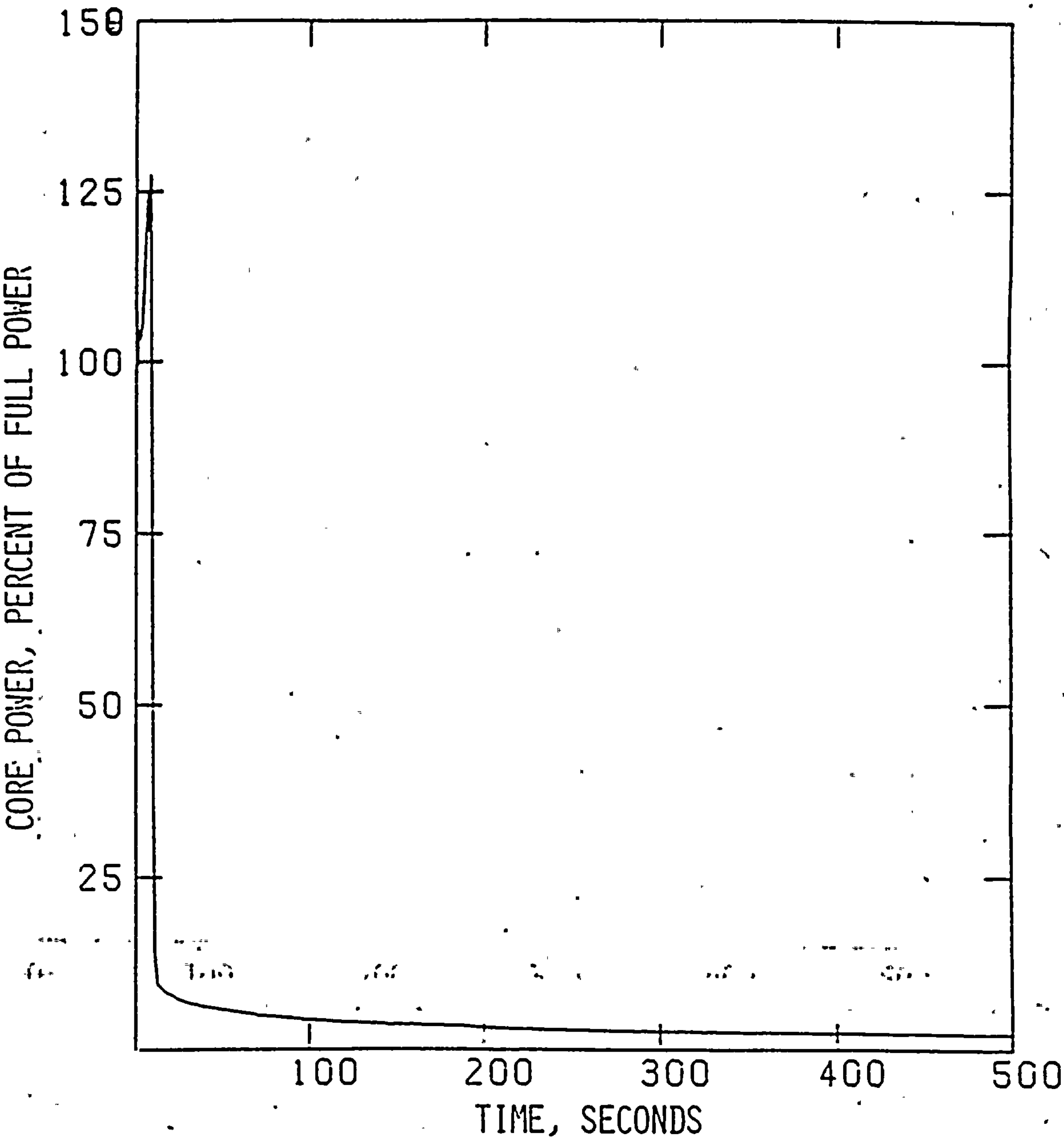
Figure 15.1-16



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER MINIMUM POST-TRIP DNBR VS TIME

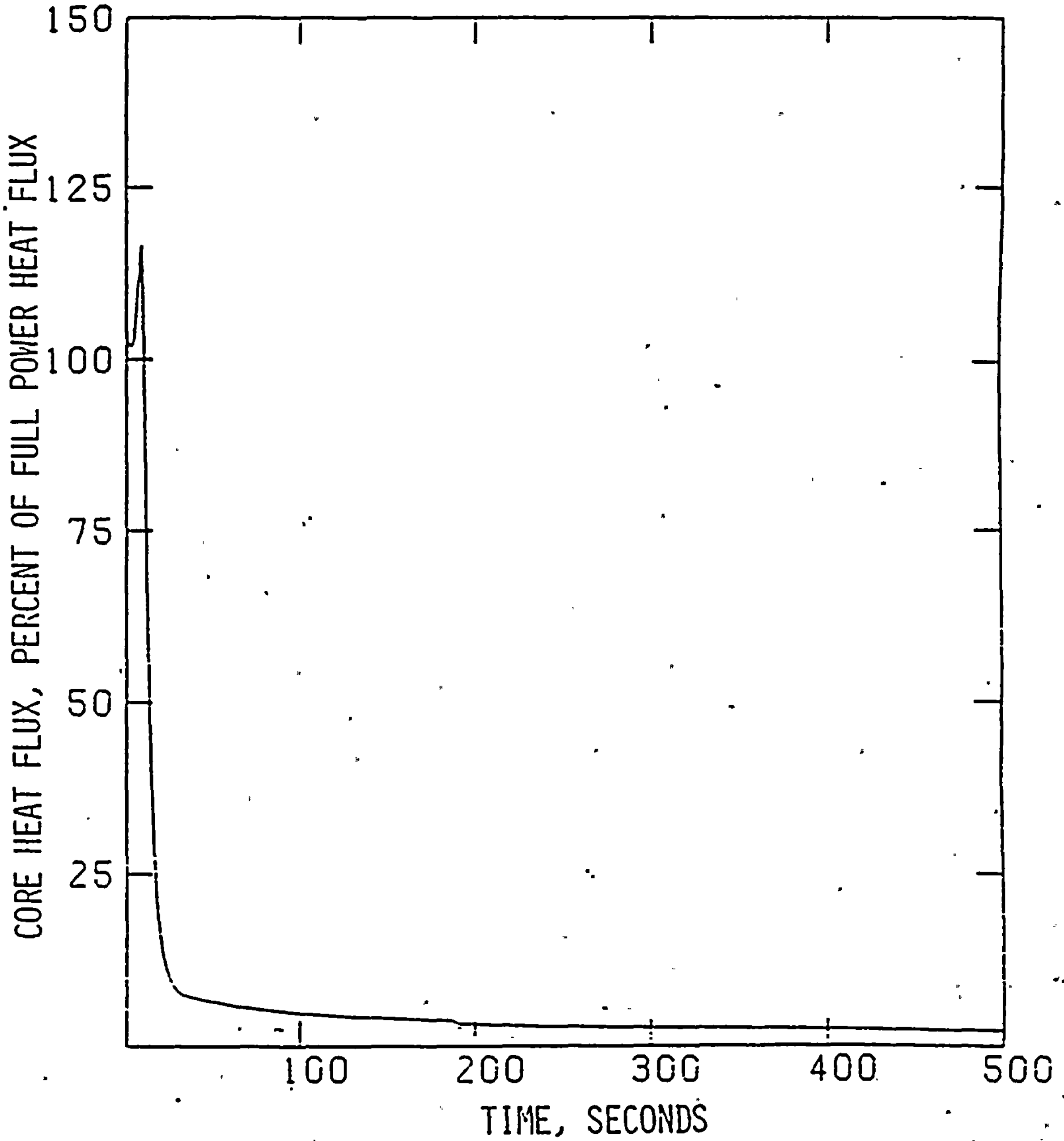
Figure 15.1.1-17



Palo Verde Nuclear Generating Station
FSAR

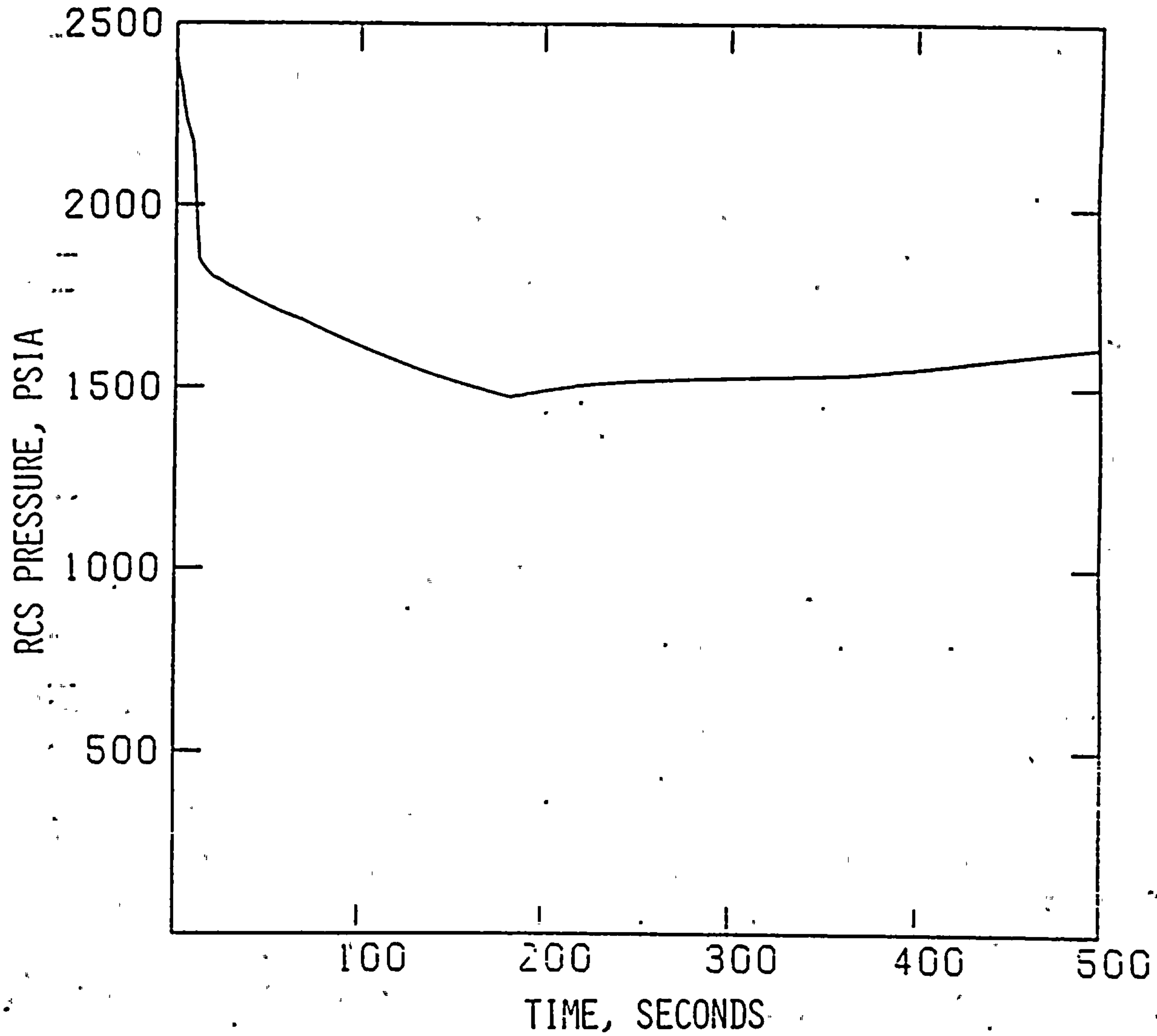
FULL POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
CORE POWER VS TIME

Figure 15.1.2-1



Palo Verde Nuclear Generating Station
FSAR

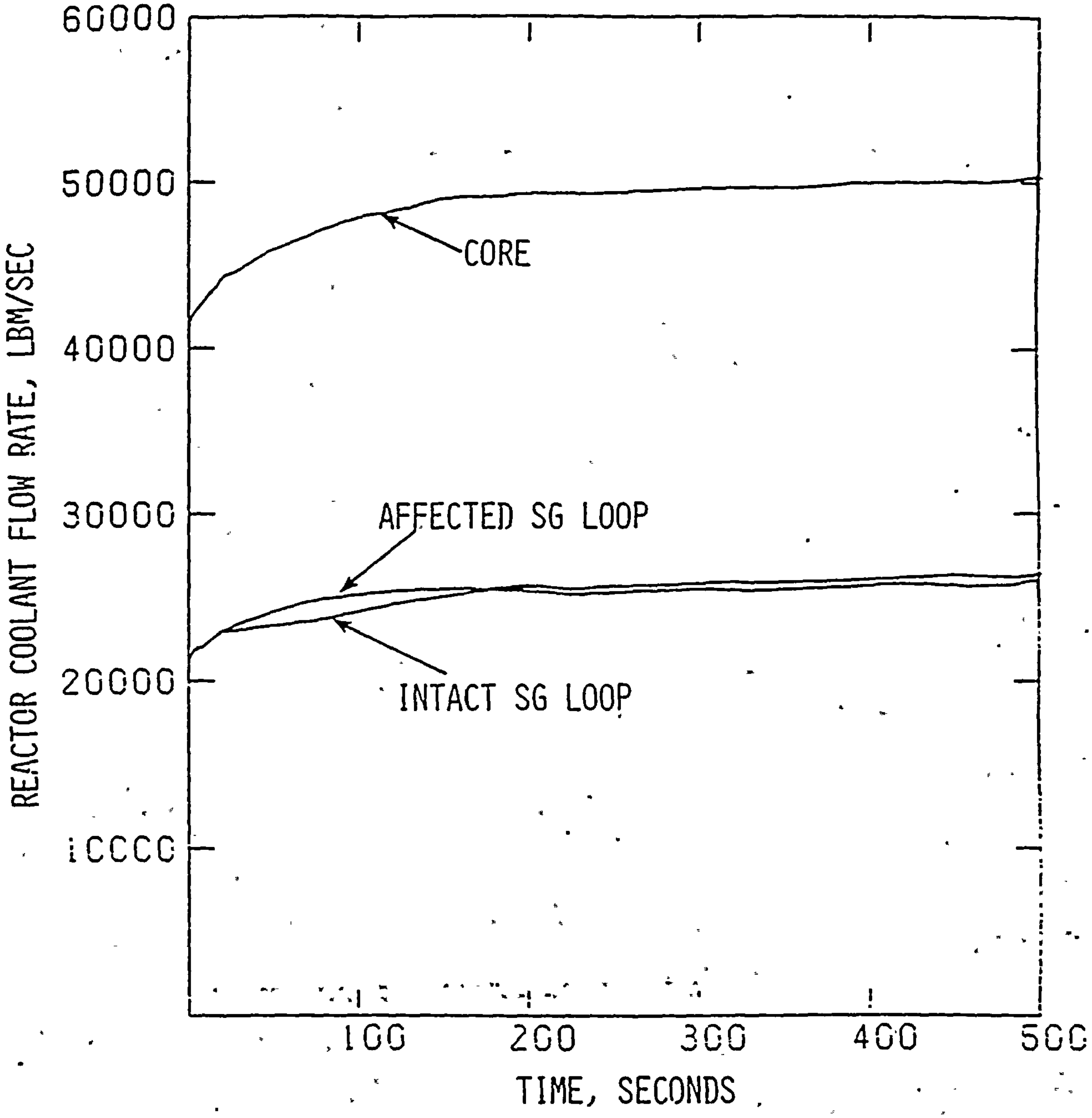
FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
CORE HEAT FLUX VS TIME
Figure 15.1.2-2



Palo Verde Nuclear Generating Station
FSAR

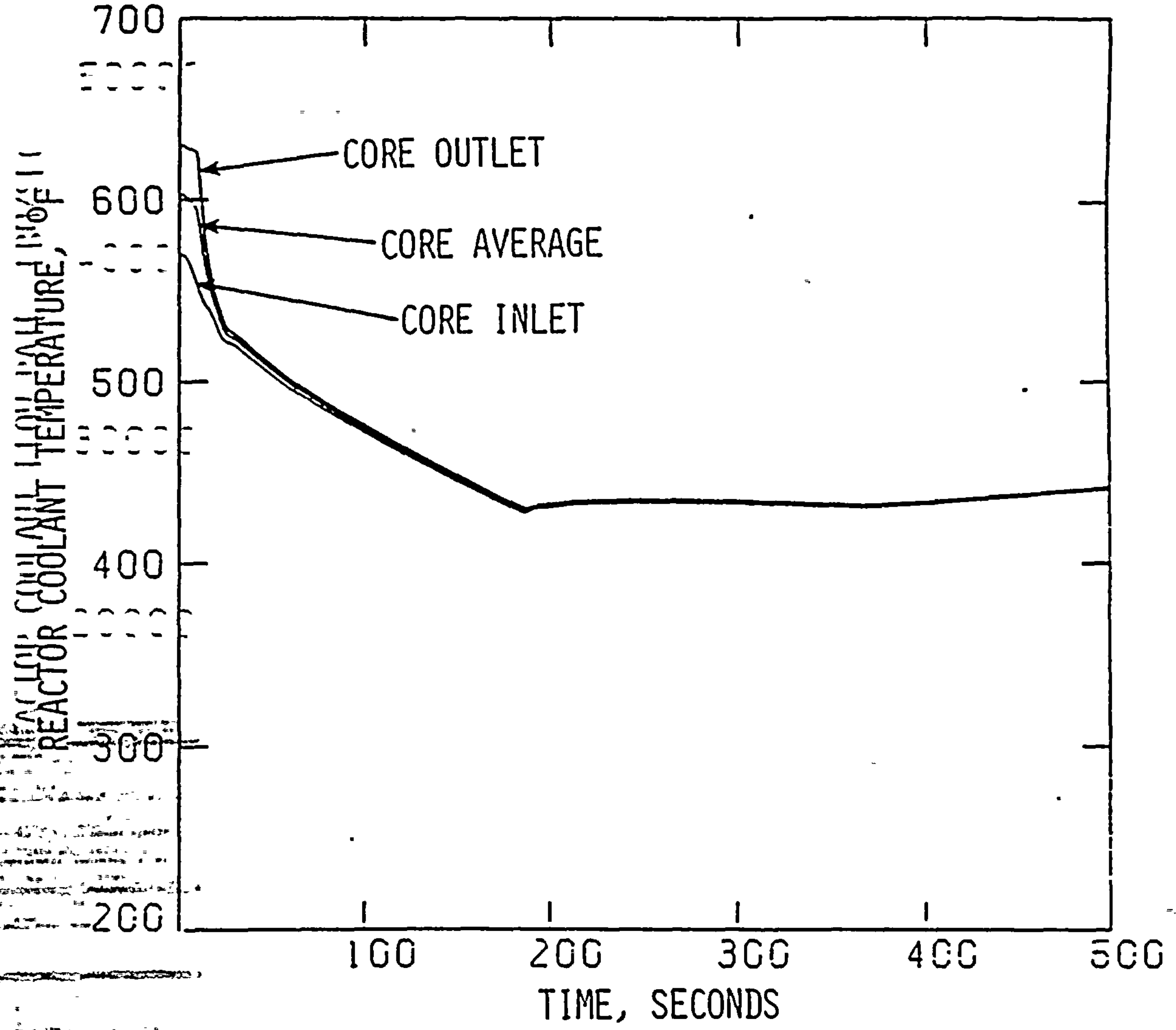
FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
RCS PRESSURE VS TIME

Figure 15.1.2-3



Palo Verde Nuclear Generating Station
FSAR

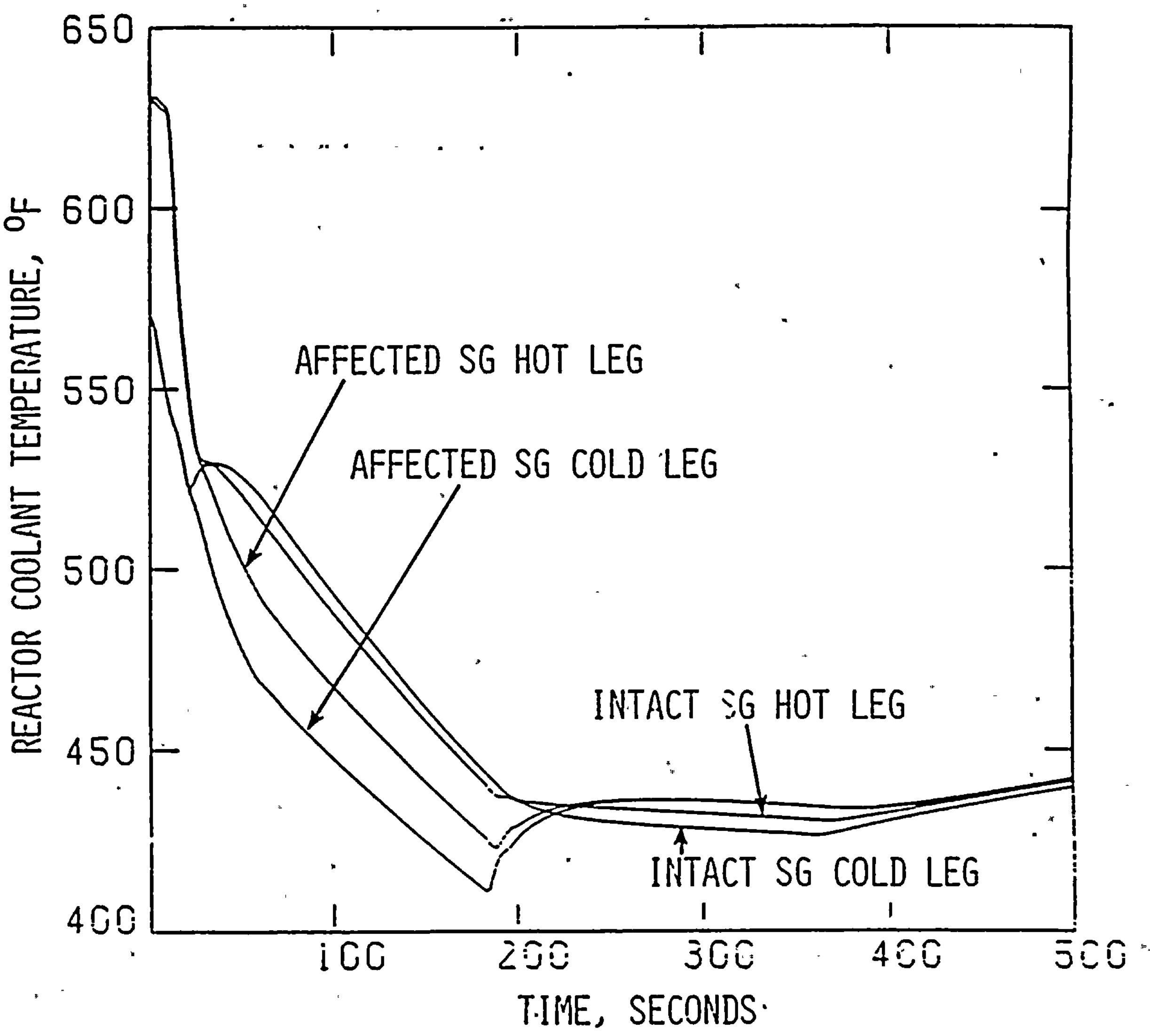
FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTOR COOLANT FLOW RATE VS TIME
Figure 15.1.2-4



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
REACTOR COOLANT TEMPERATURES VS TIME

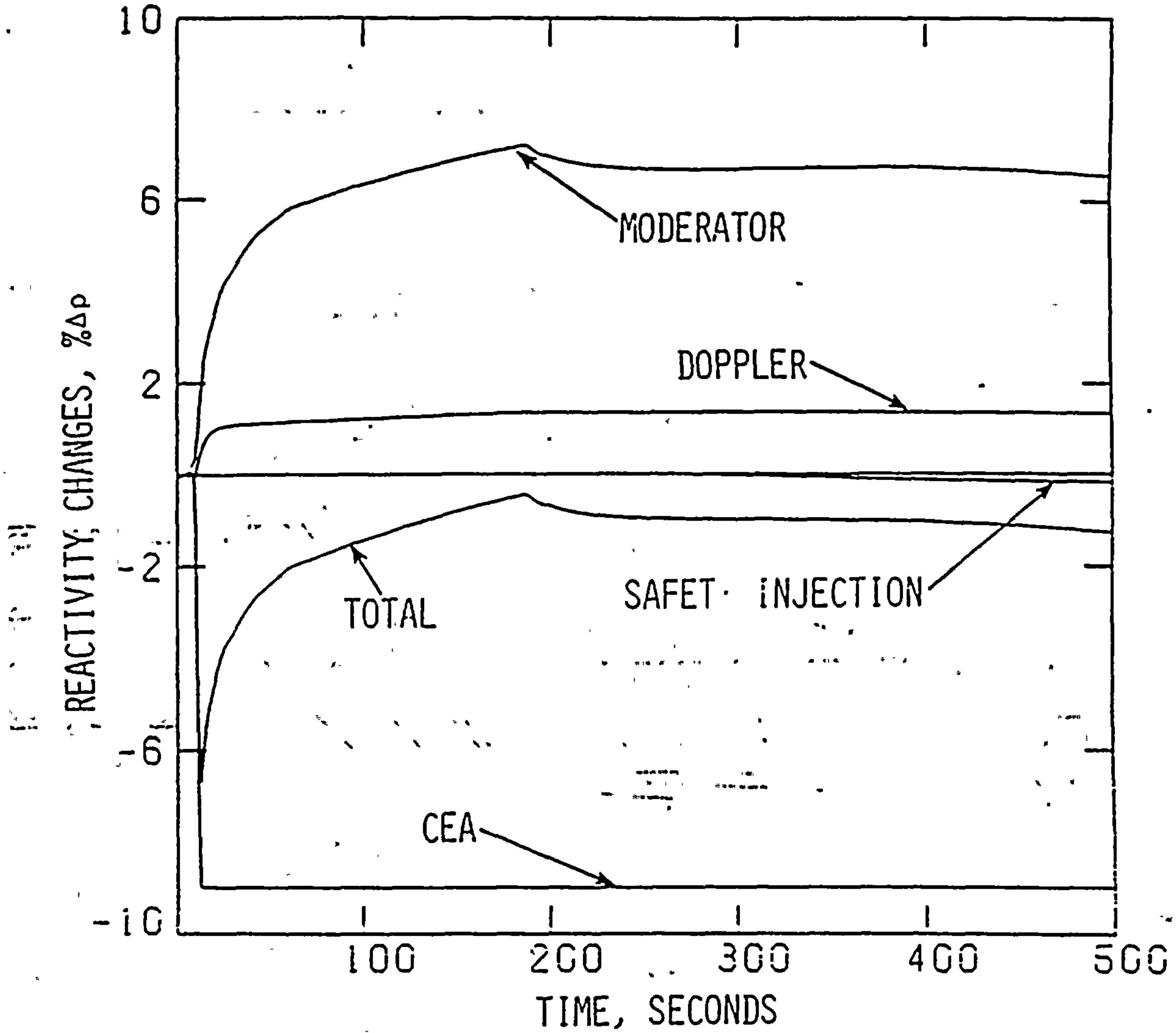
Figure 15.1.2-5



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTOR COOLANT TEMPERATURES VS TIME

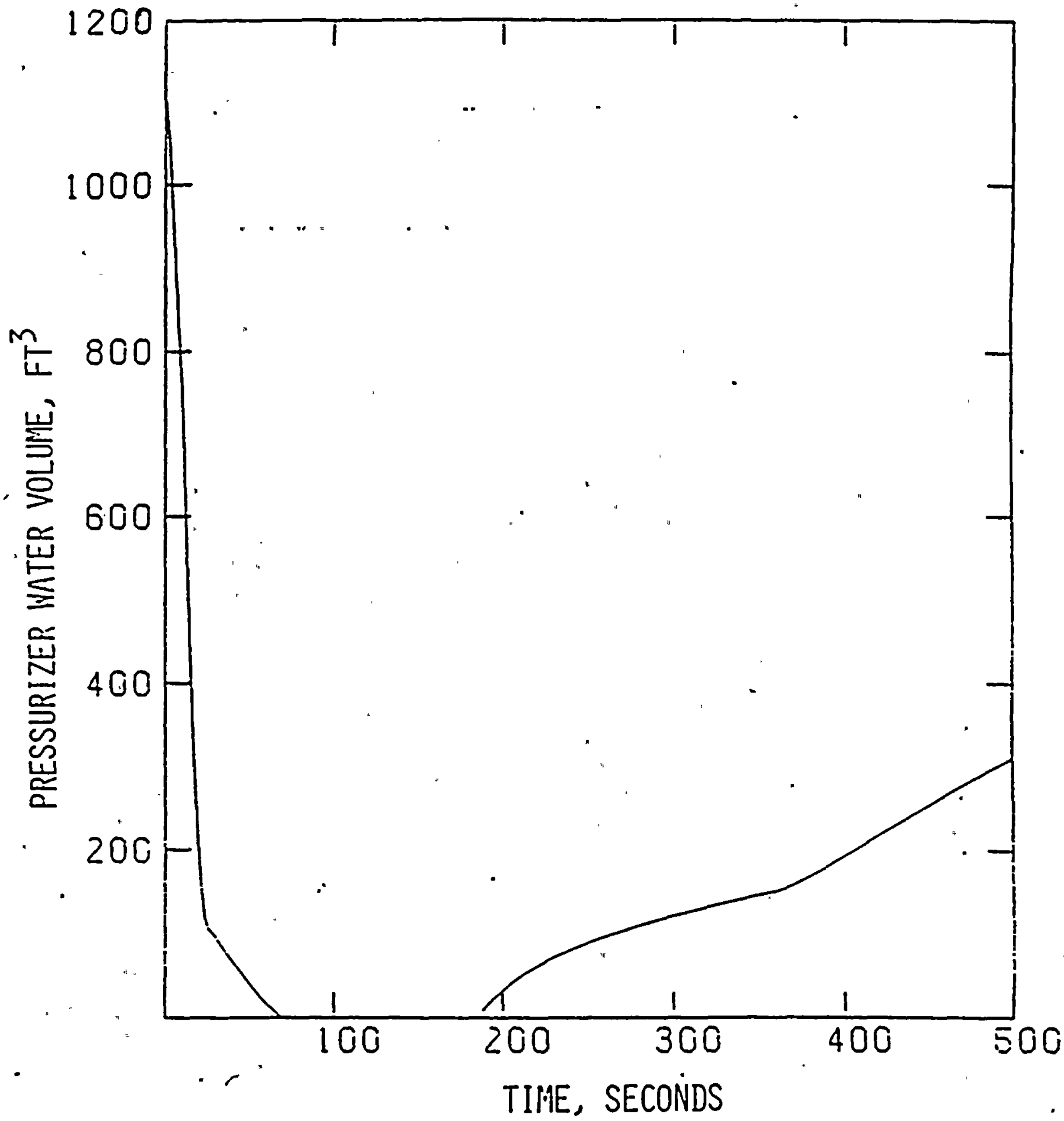
Figure 15.1.2-6



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE REACTIVITY CHANGES VS TIME

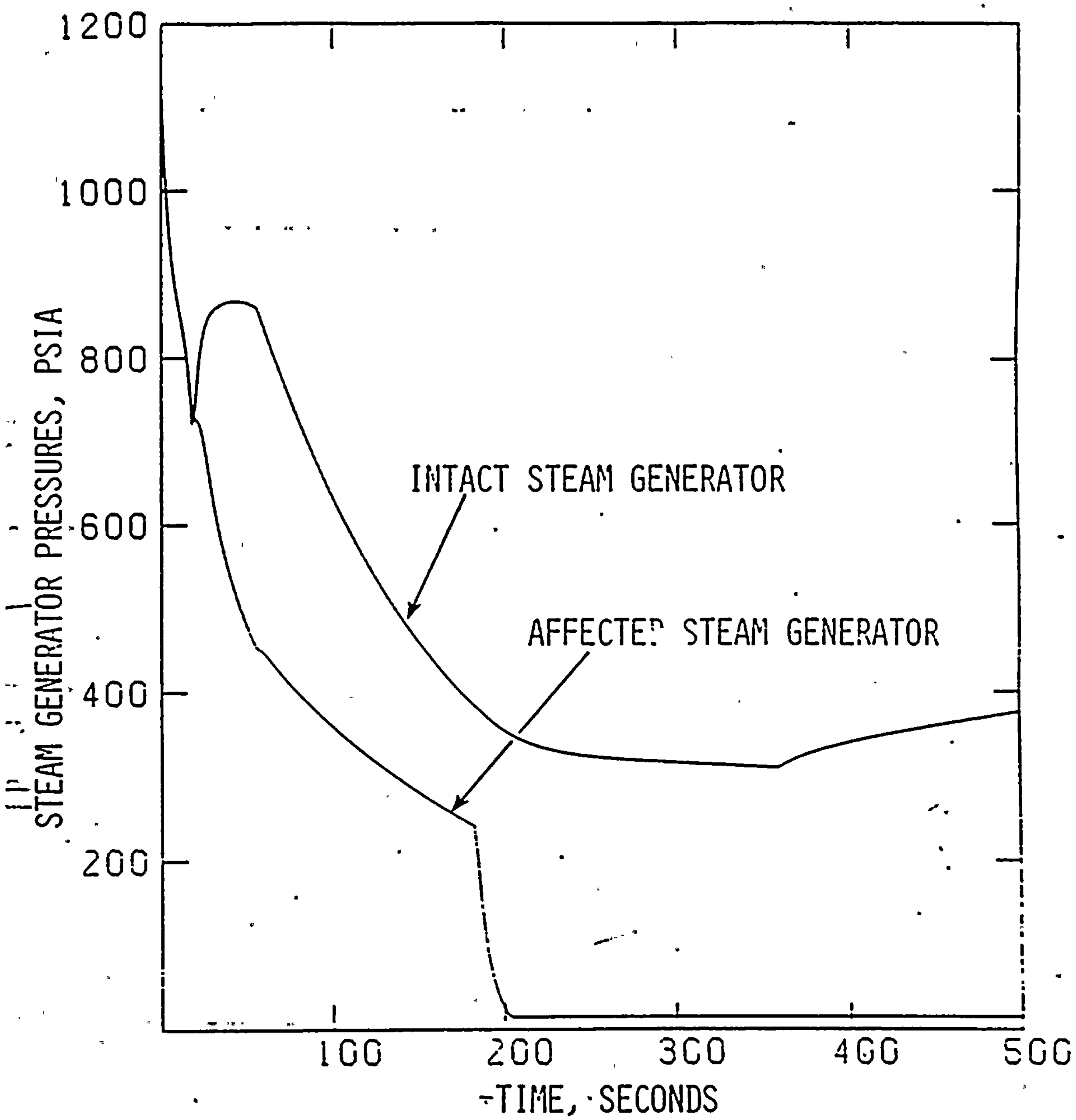
Figure 15.1.2-7



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
PRESSURIZER WATER VOLUME VS TIME

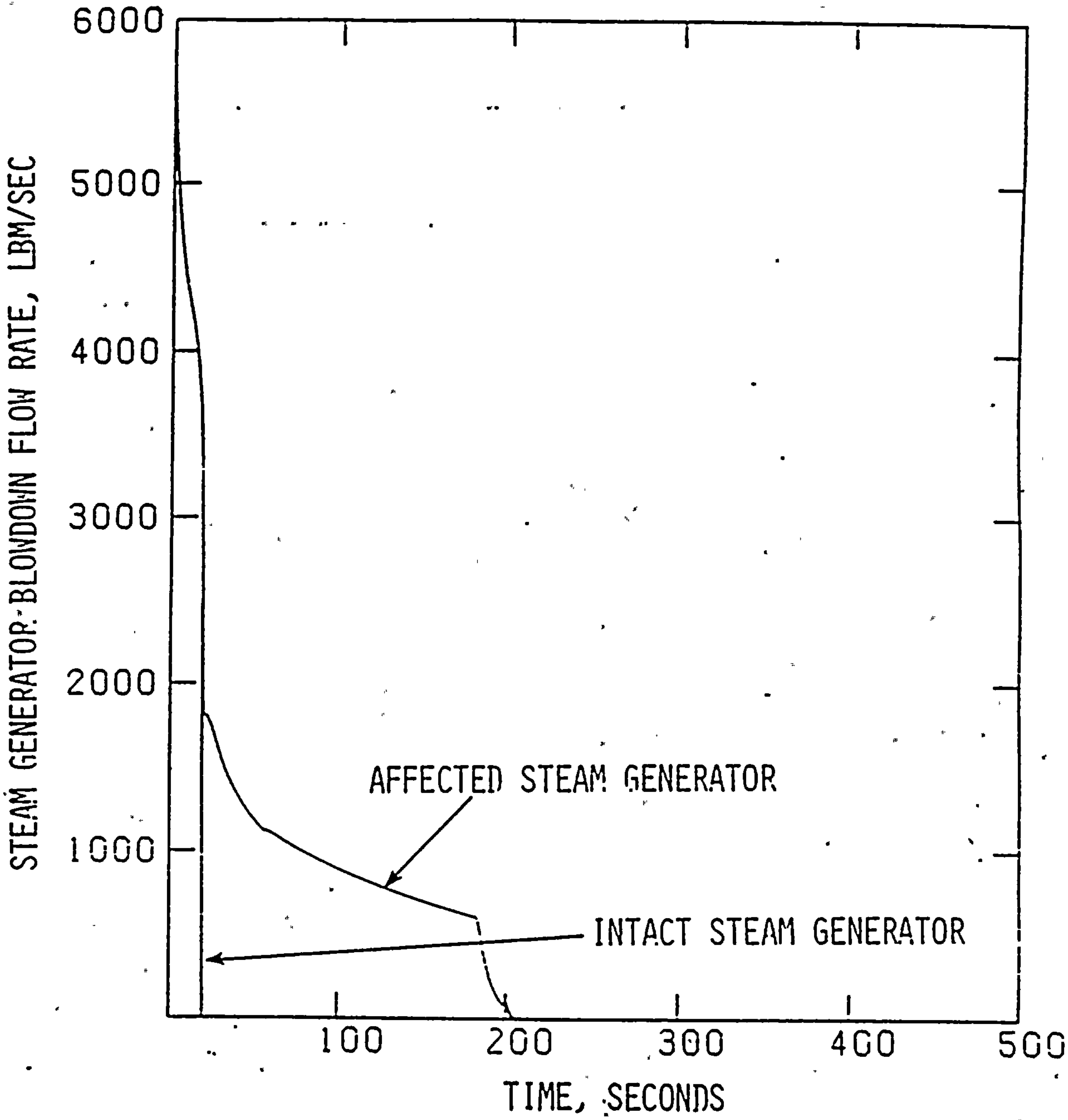
Figure 15.1.2-8



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
STEAM GENERATOR PRESSURES VS TIME

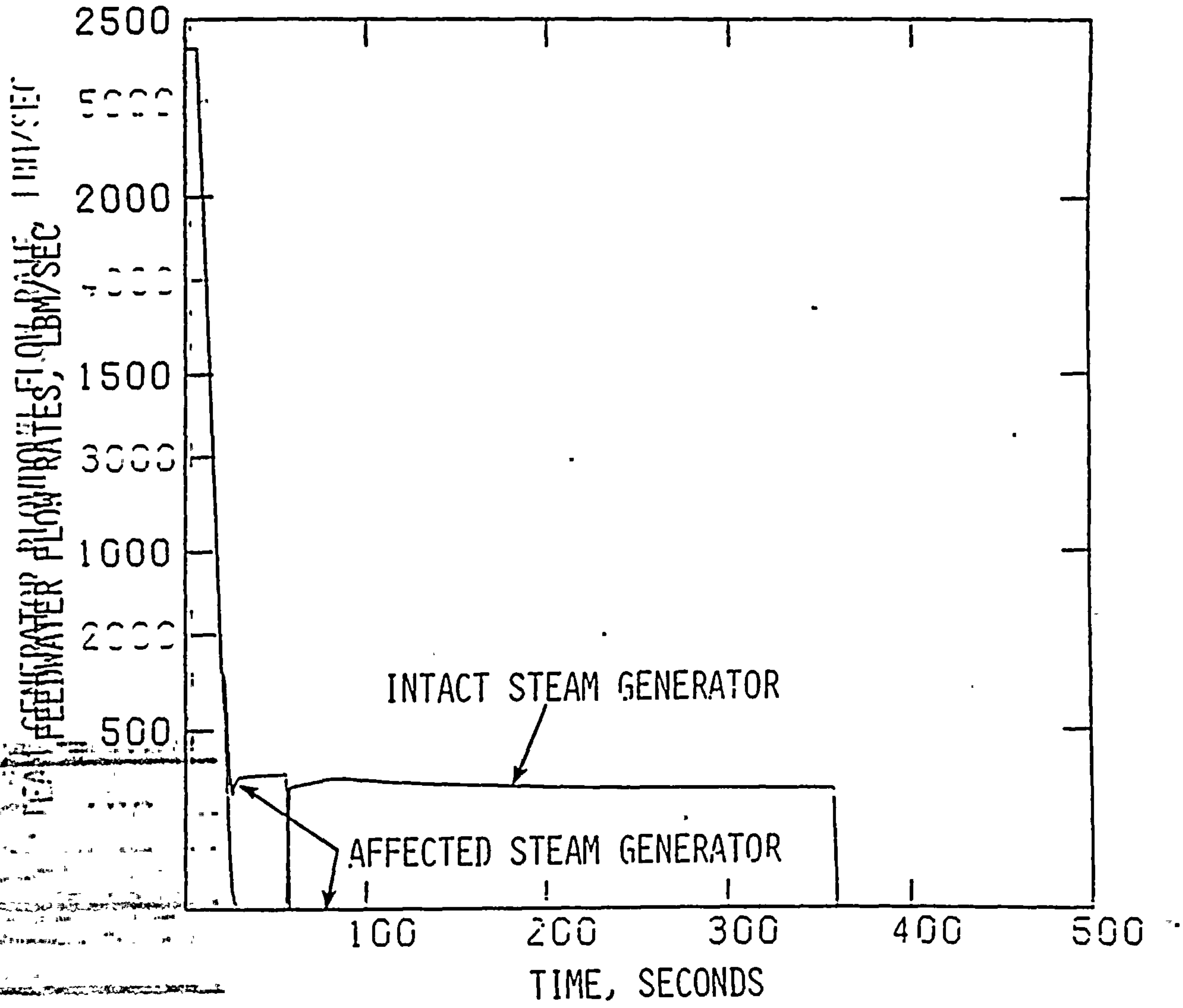
Figure 15.1.2-9



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
STEAM GENERATOR BLOWDOWN RATES VS TIME

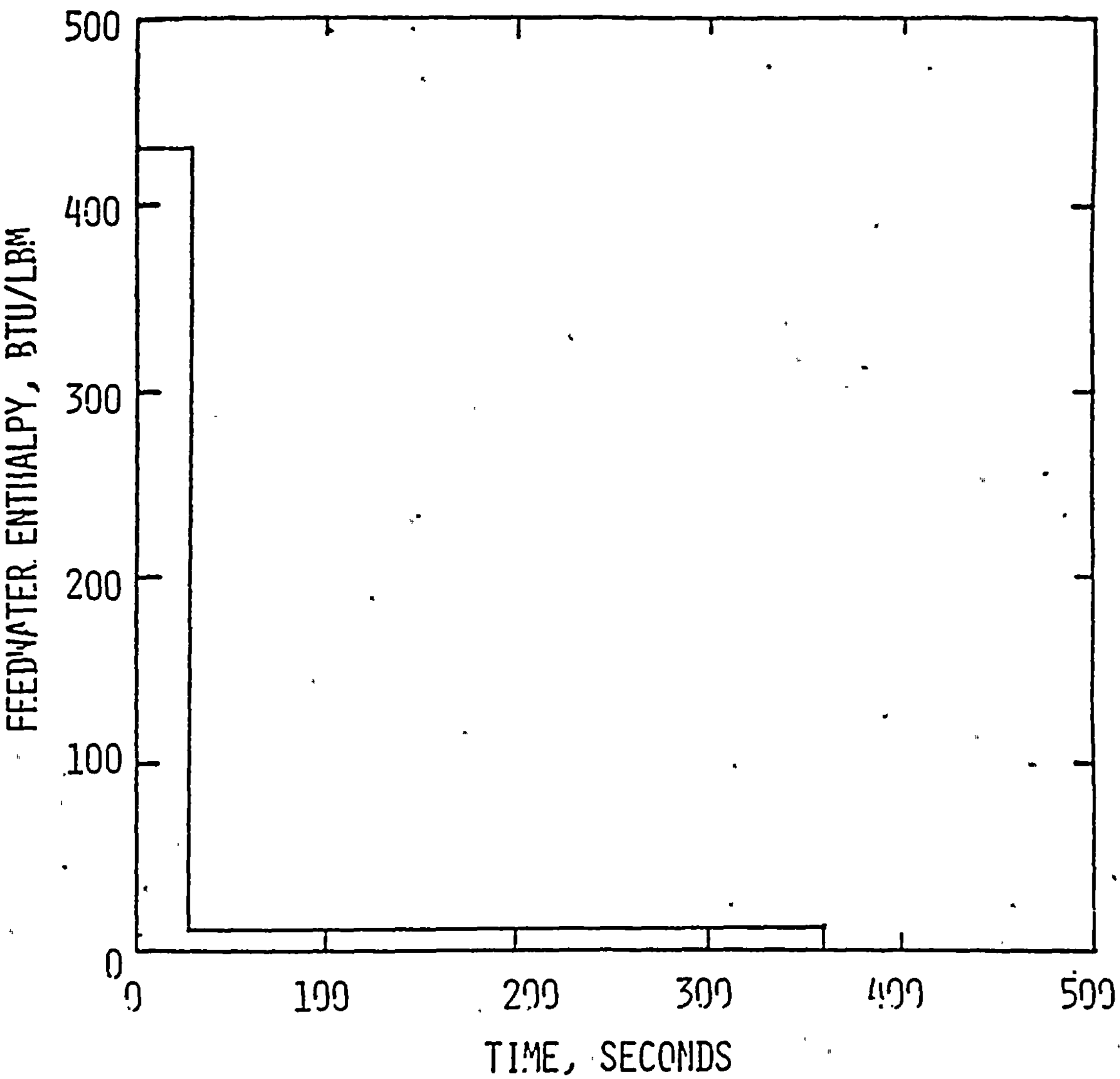
Figure 15.1.2-10



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE FEEDWATER FLOW RATES VS TIME

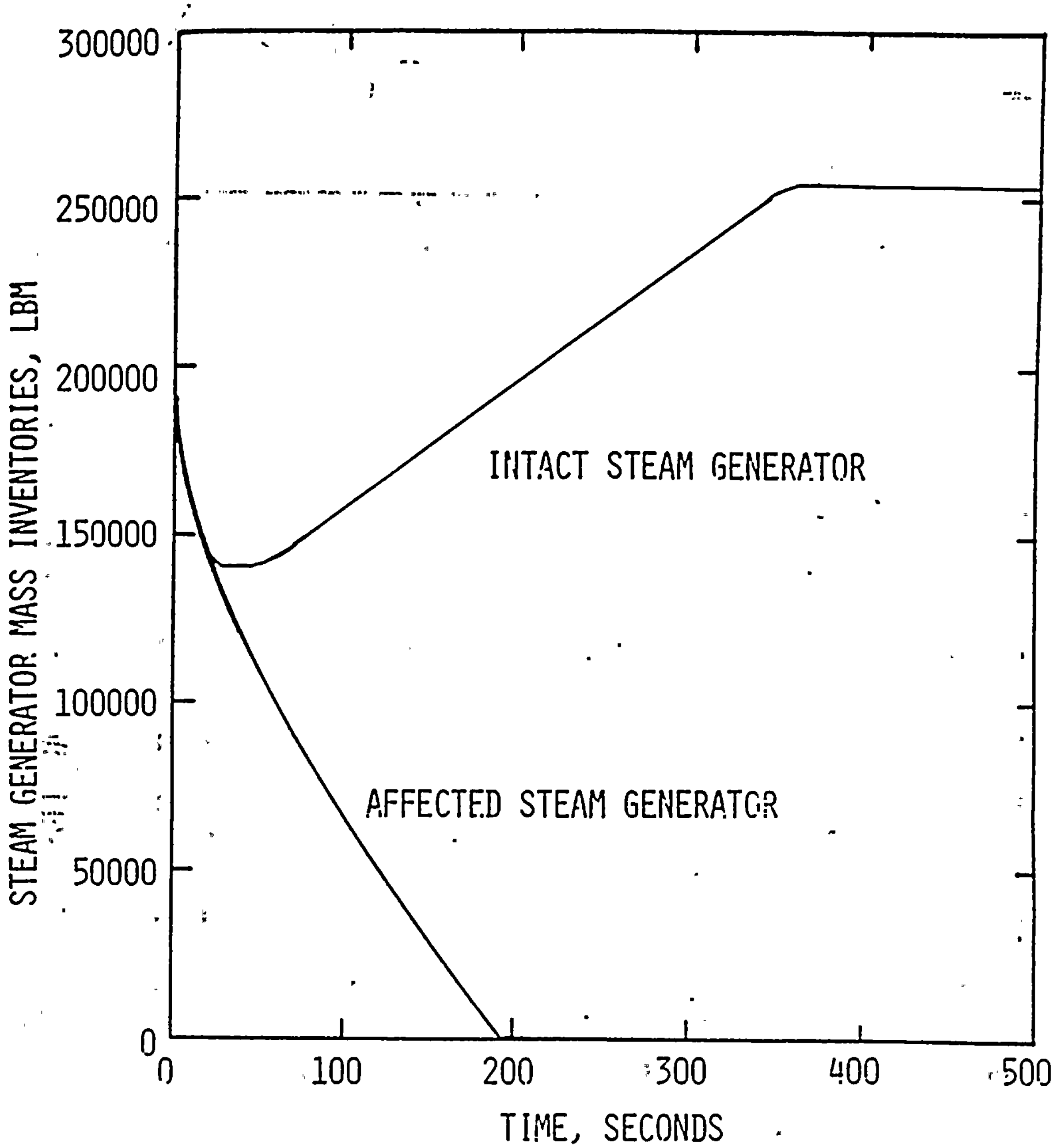
Figure 15.1.2-11



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
FEEDWATER ENTHALPY VS TIME

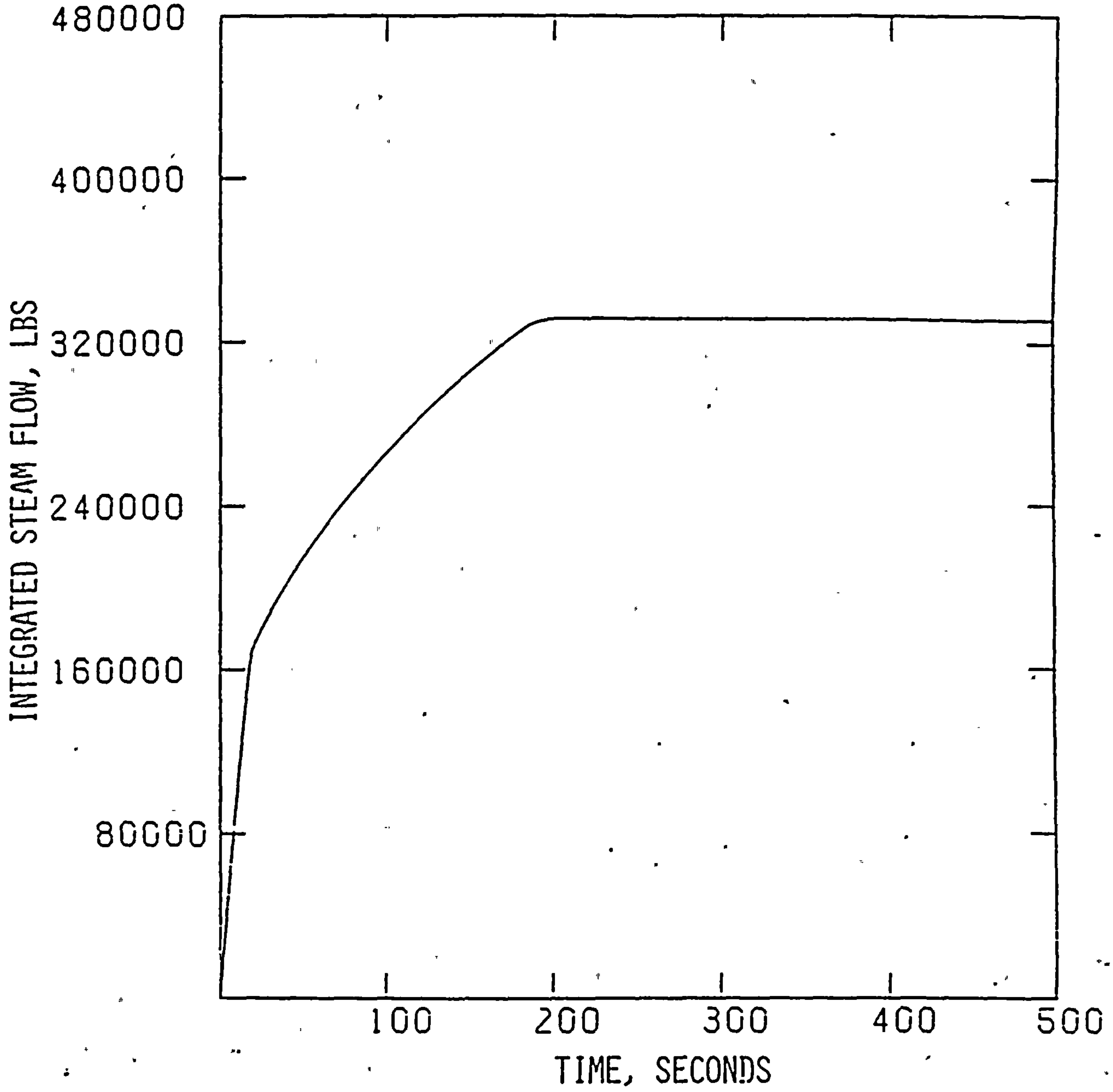
Figure 15.1.2-12



Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
STEAM GENERATOR MASS INVENTORIES VS TIME

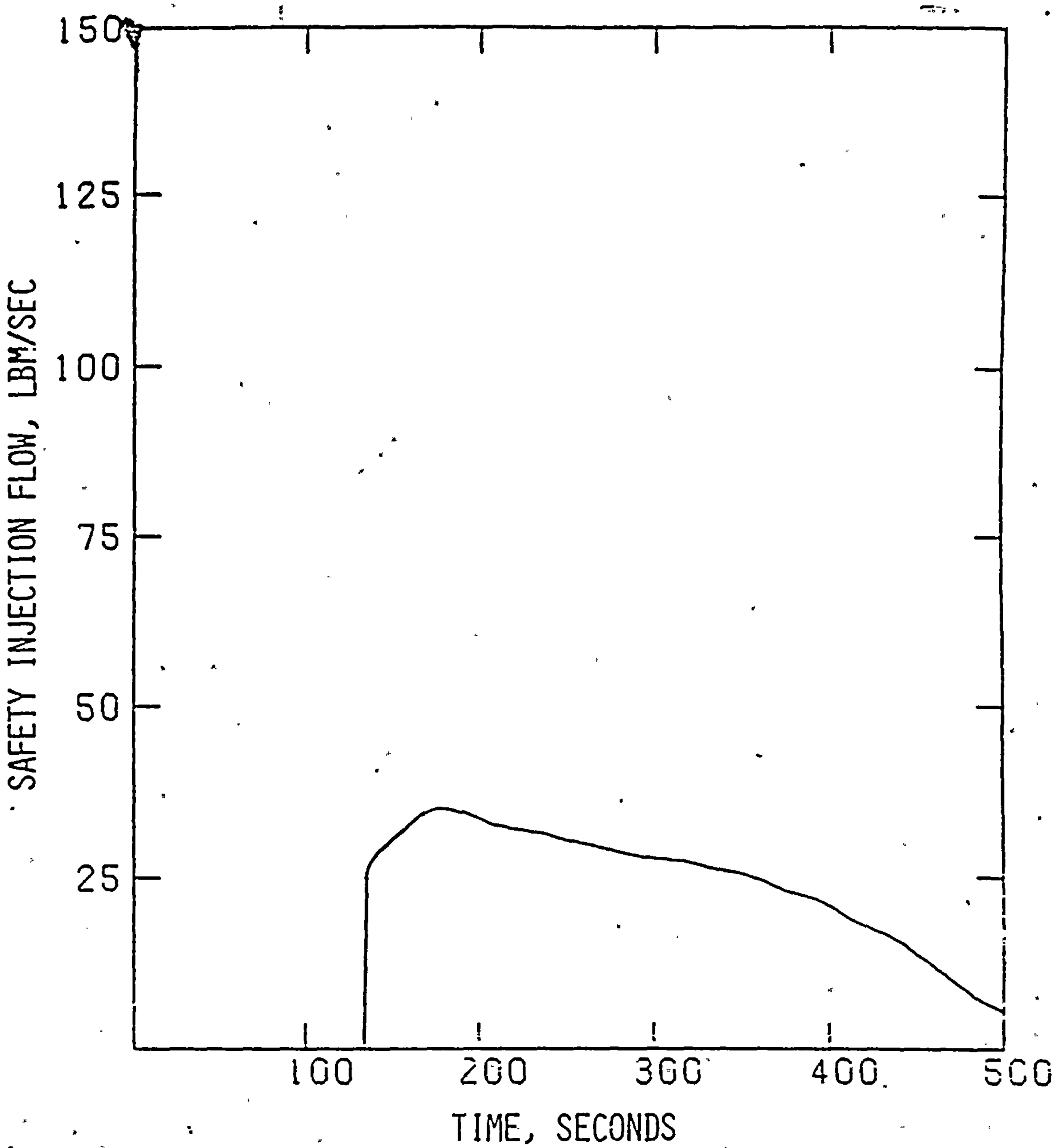
Figure 15.1.2-13



Palo Verde Nuclear Generating Station
FSAR

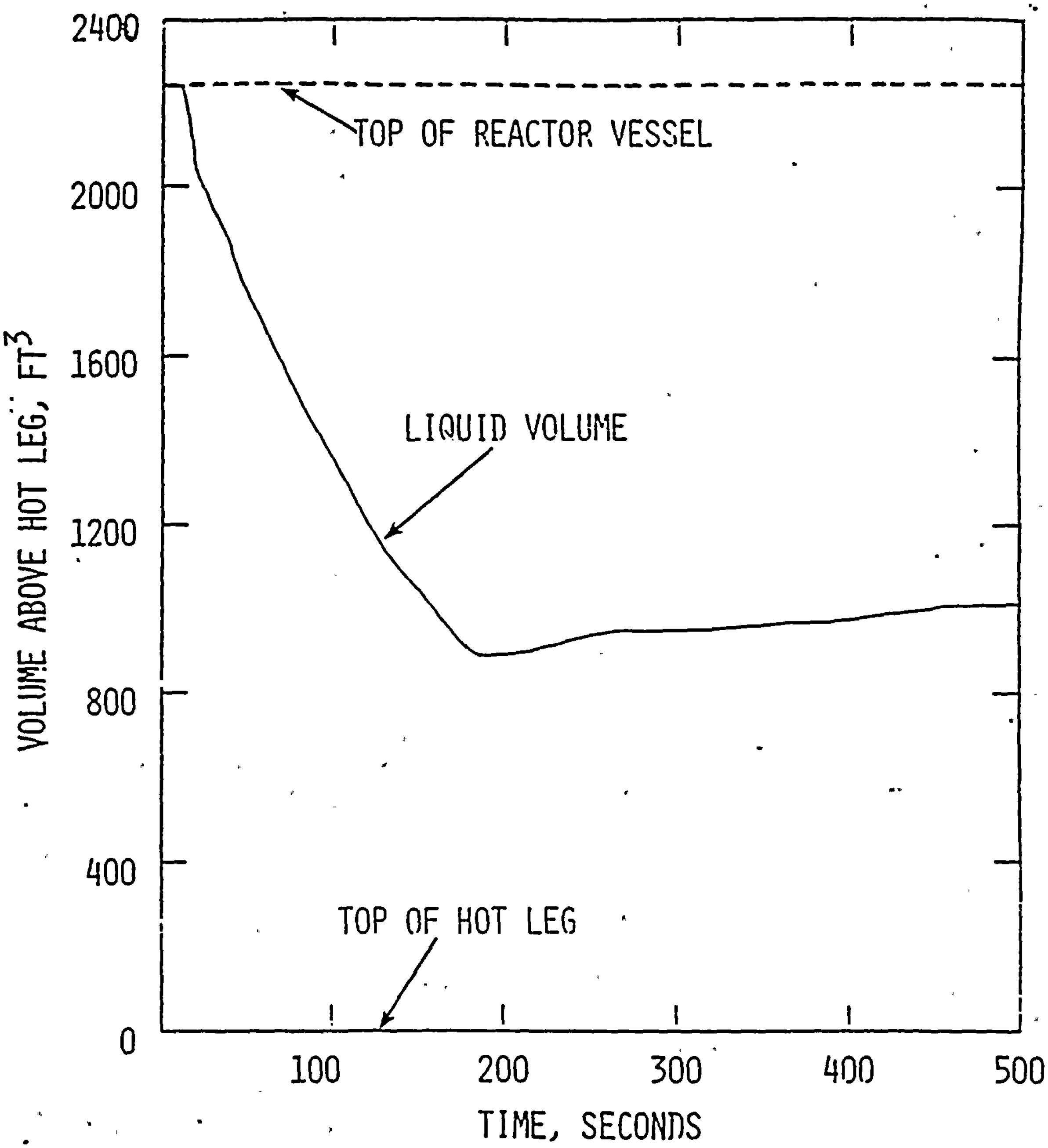
FULL POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
INTEGRATED STEAM RELEASE VS TIME

Figure 15.1.2-14



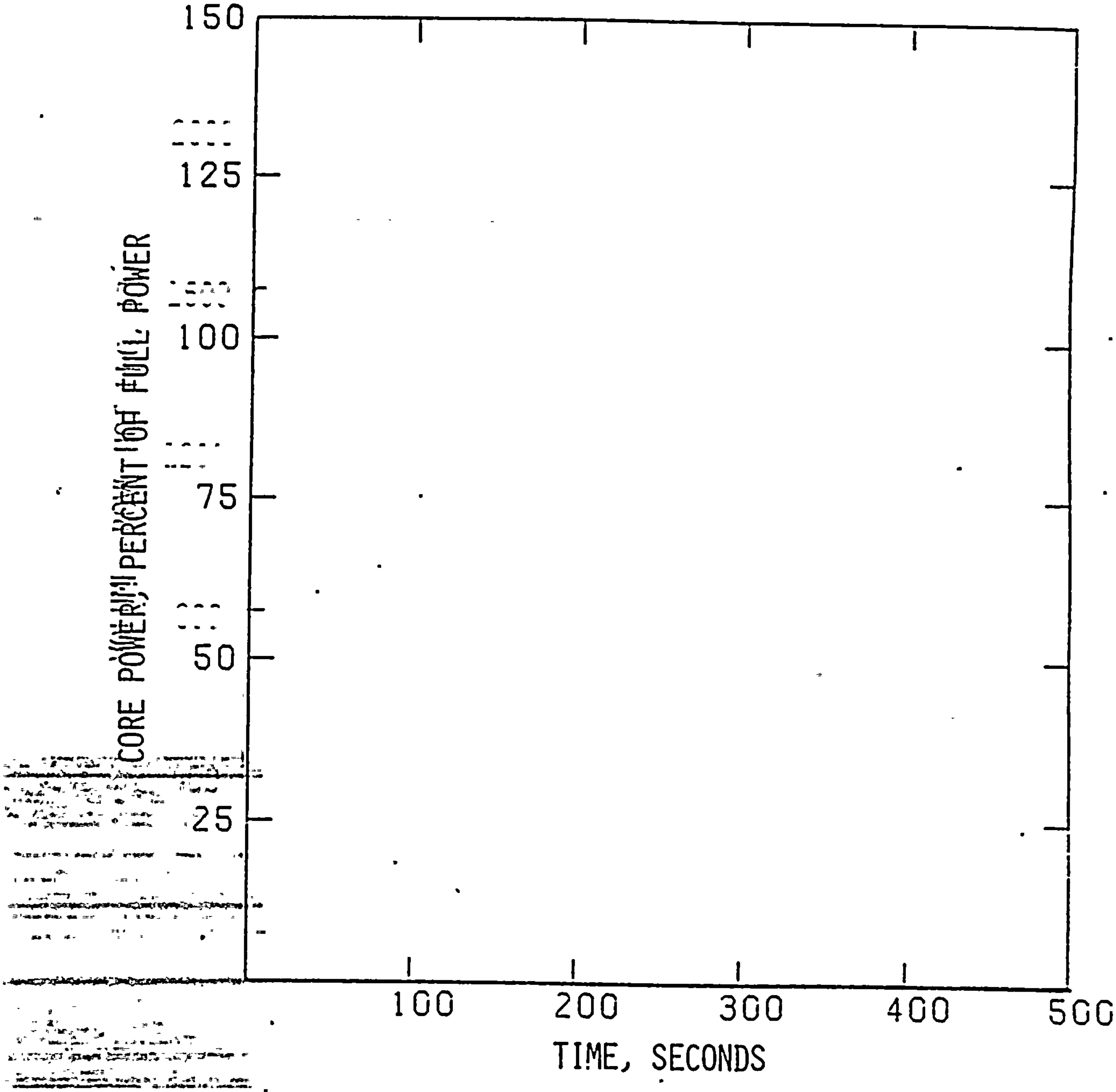
Palo Verde Nuclear Generating Station
FSAR

FULL POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
SAFETY INJECTION FLOW VS TIME
Figure 15.1.2-15



Palo Verde Nuclear Generating Station
FSAR

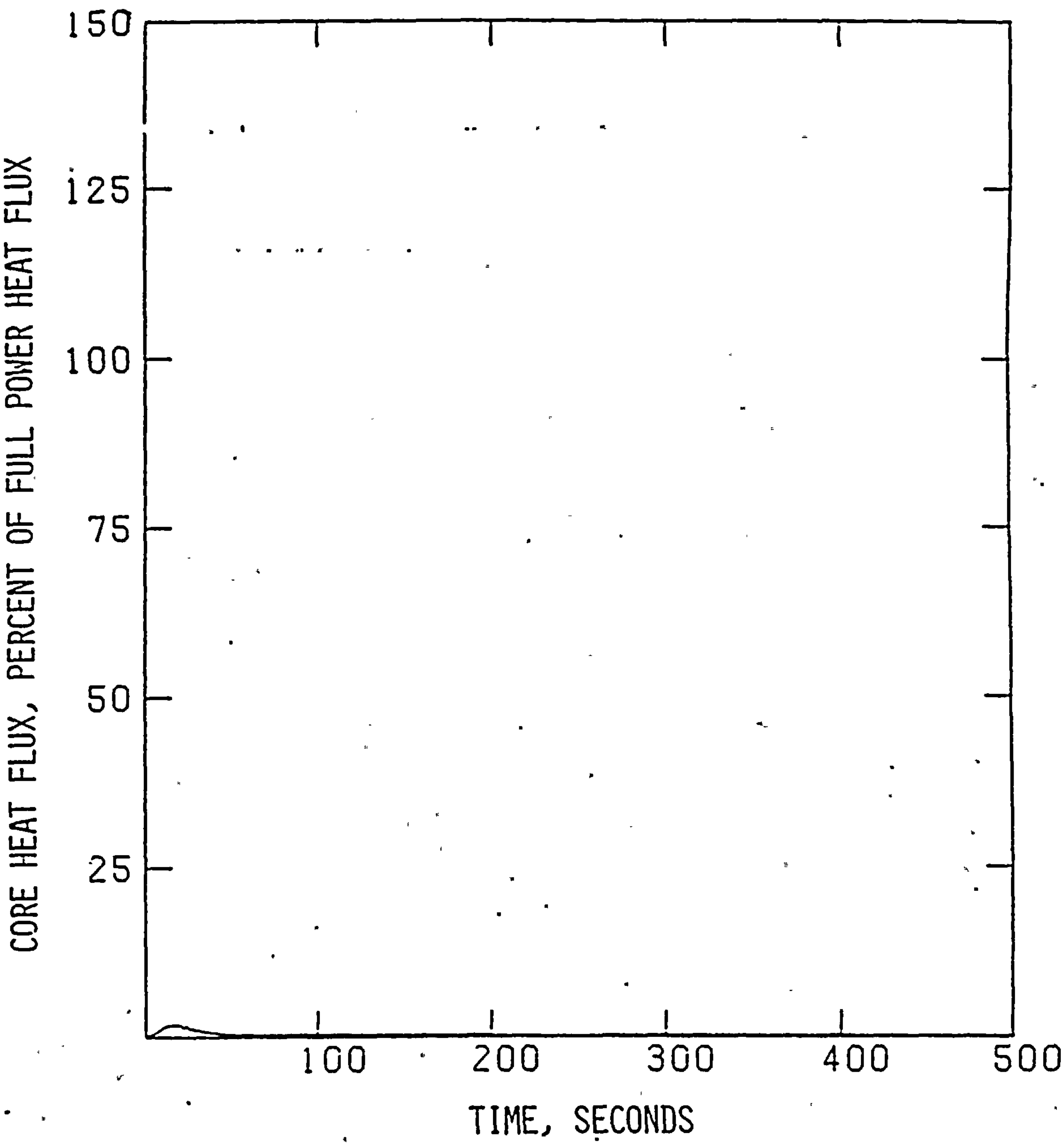
FULL POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
REACTOR VESSEL LIQUID VOLUME VS TIME
Figure 15.1.2-16



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER
CORE POWER VS TIME

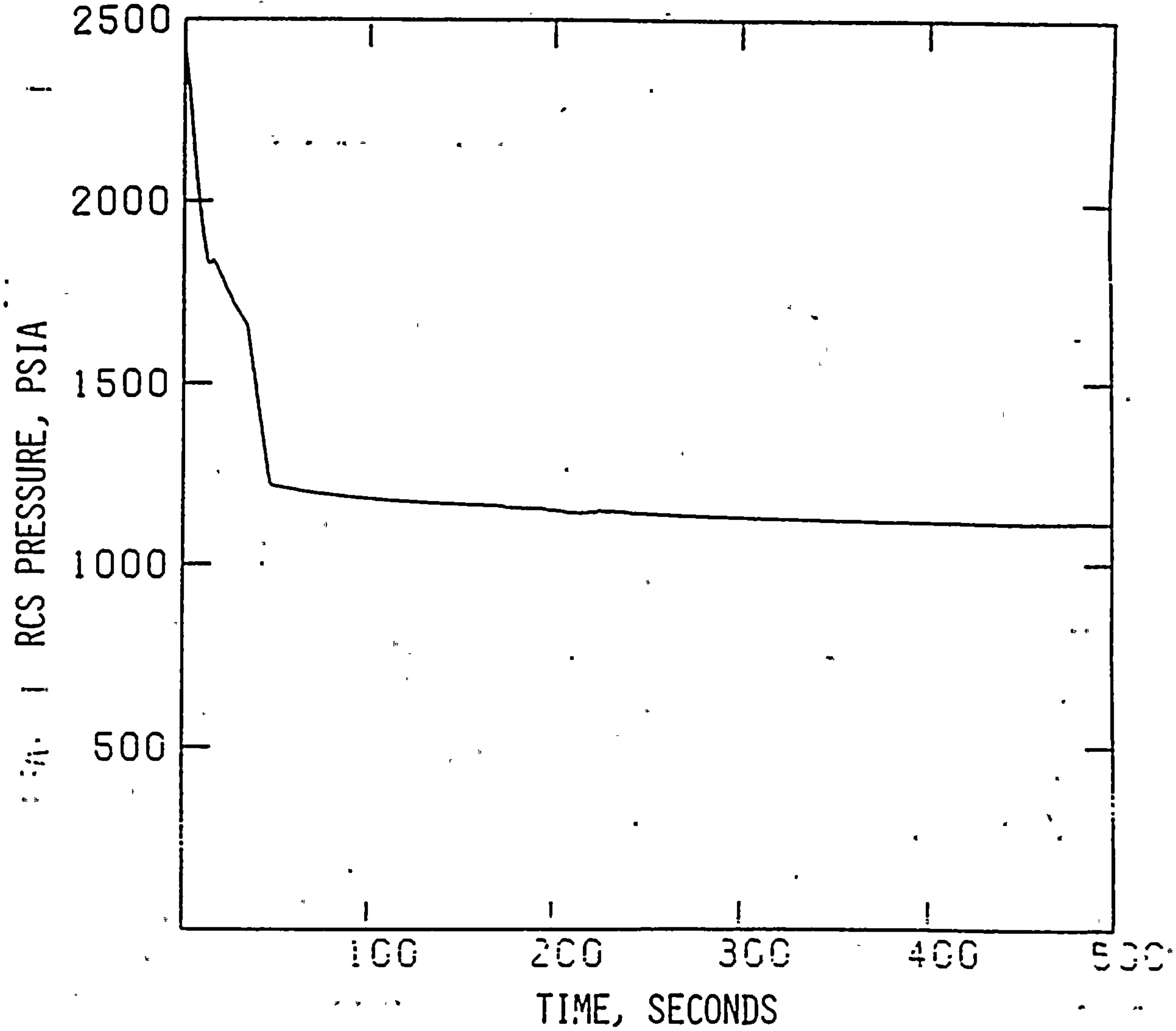
Figure 15.1.3-1



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER CORE HEAT FLUX VS TIME

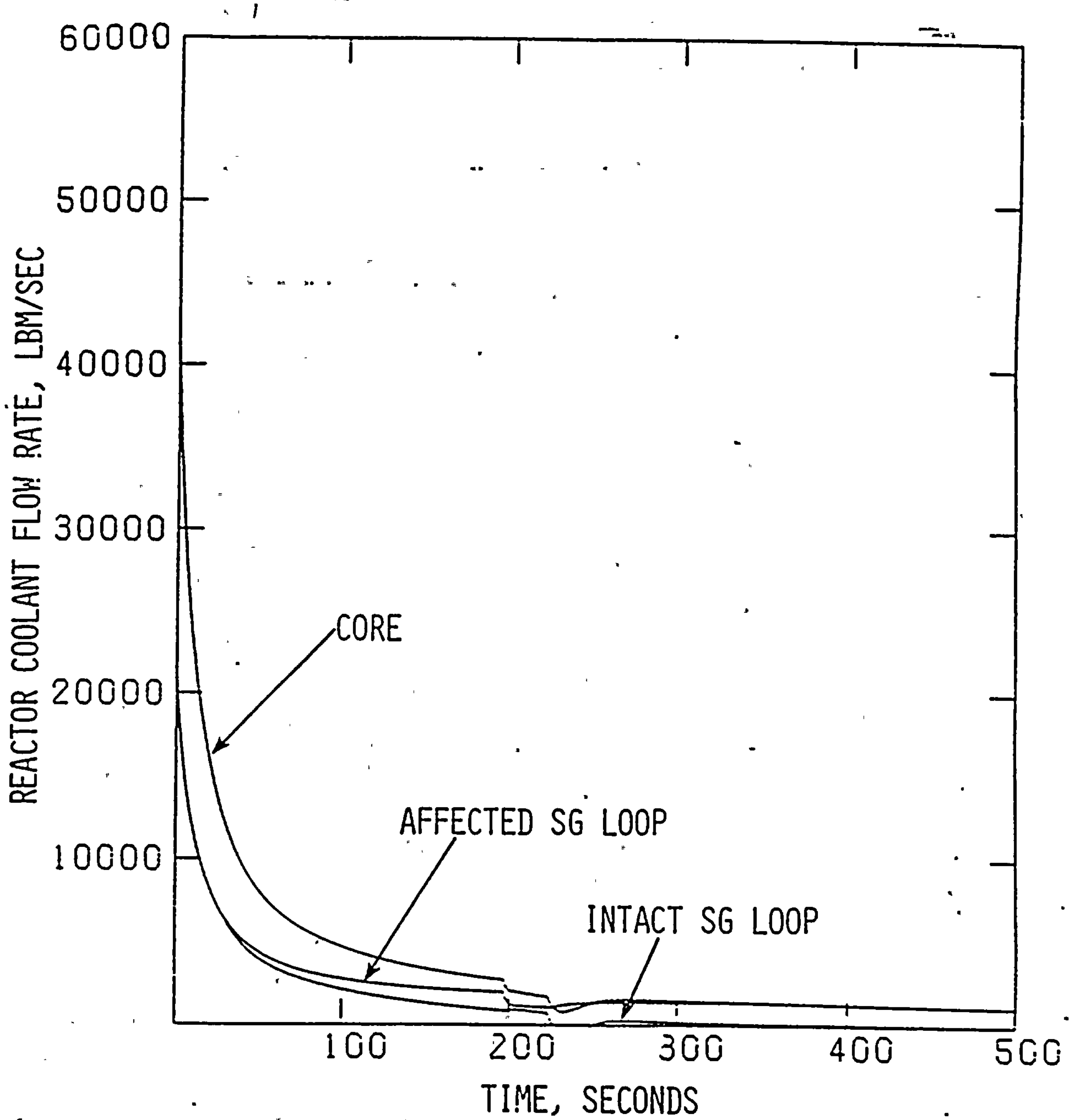
Figure 15.1.3-2



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER
RCS PRESSURE VS TIME

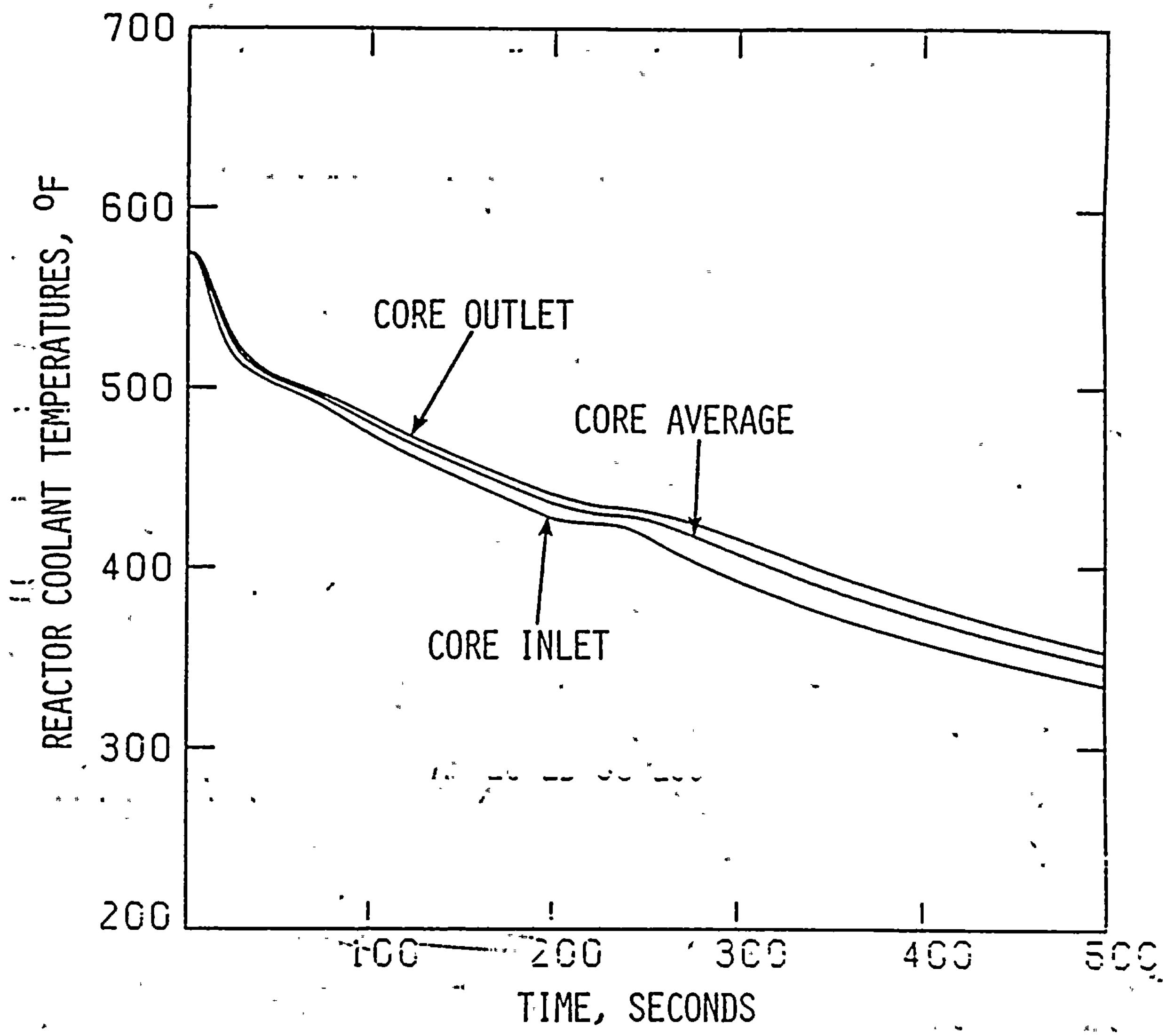
Figure 15.1.3-3



Palo Verde Nuclear Generating Station FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT FLOW RATE VS TIME

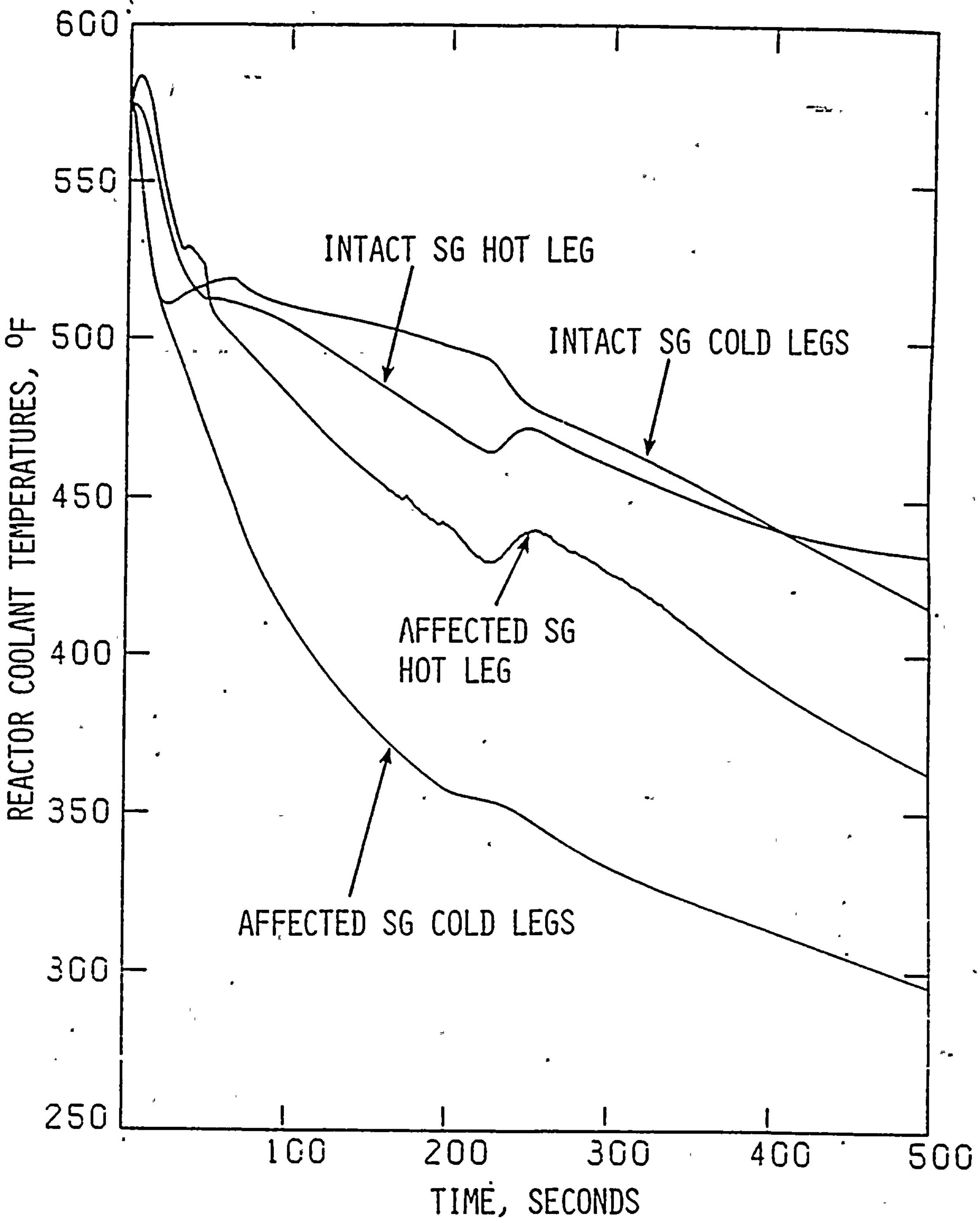
Figure 15.1.3-4



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT TEMPERATURES VS TIME

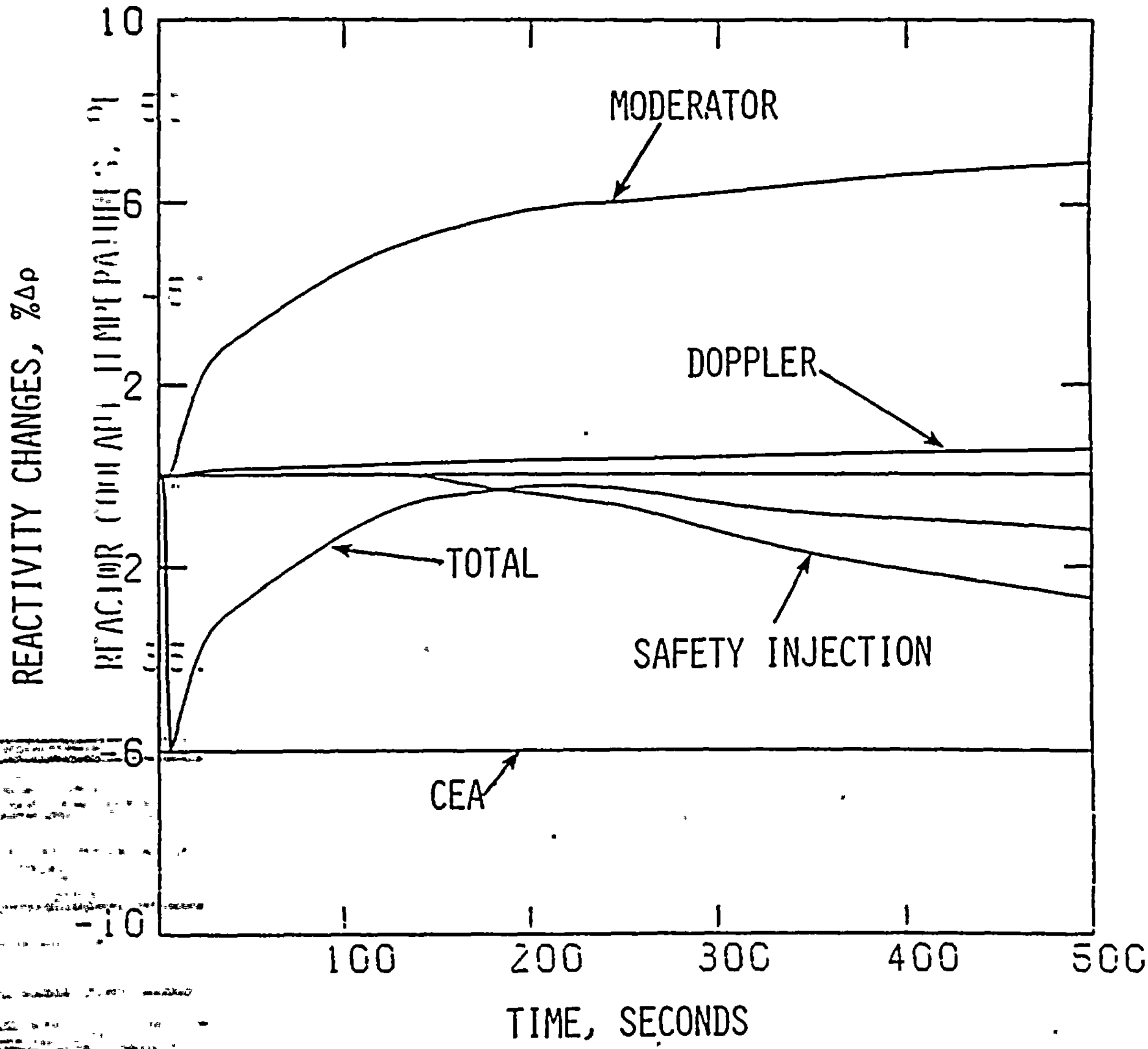
Figure 15.1.3-5



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR COOLANT TEMPERATURES VS TIME

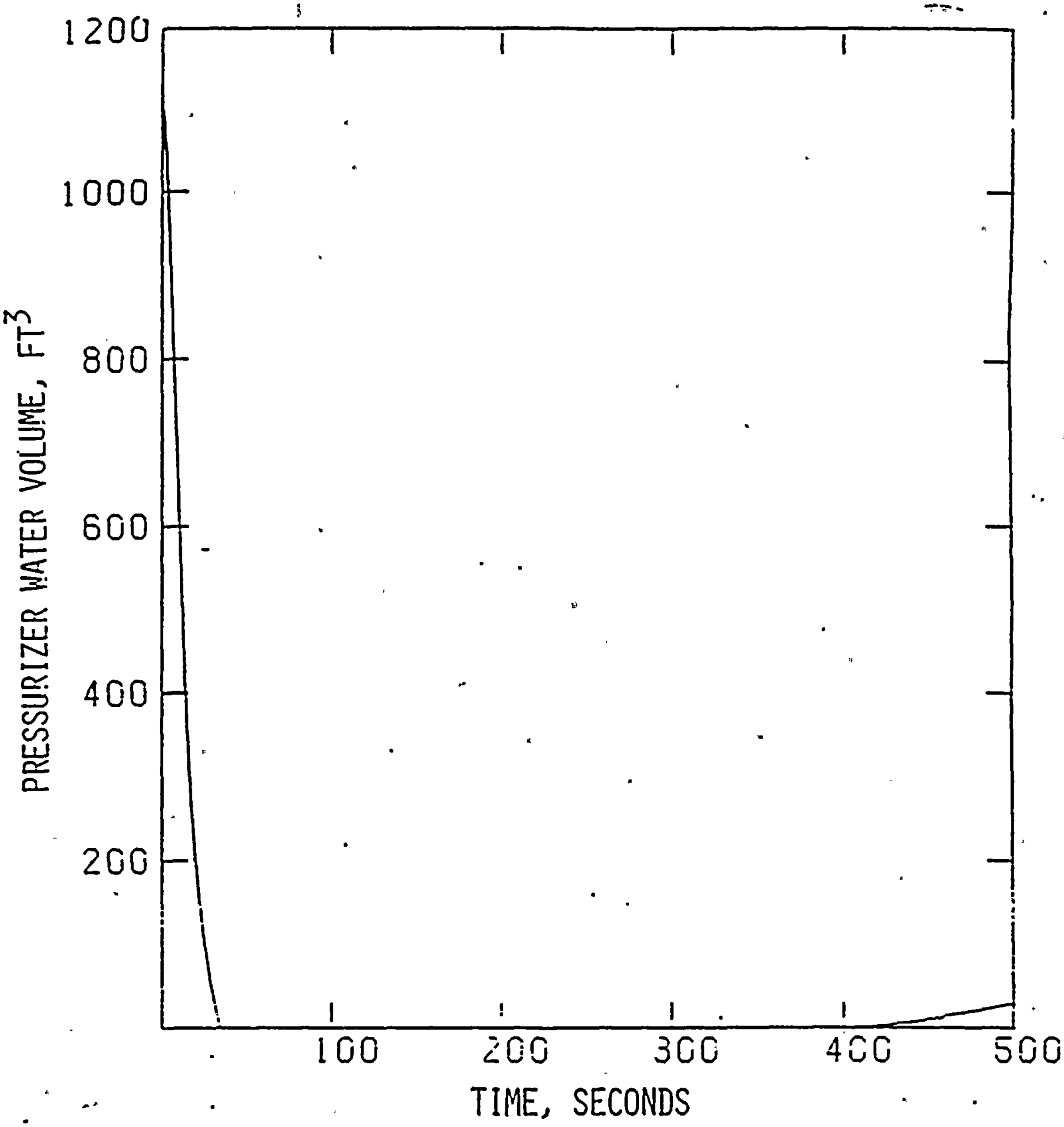
Figure 15.1.3-6



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARTE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTIVITY CHANGES VS TIME

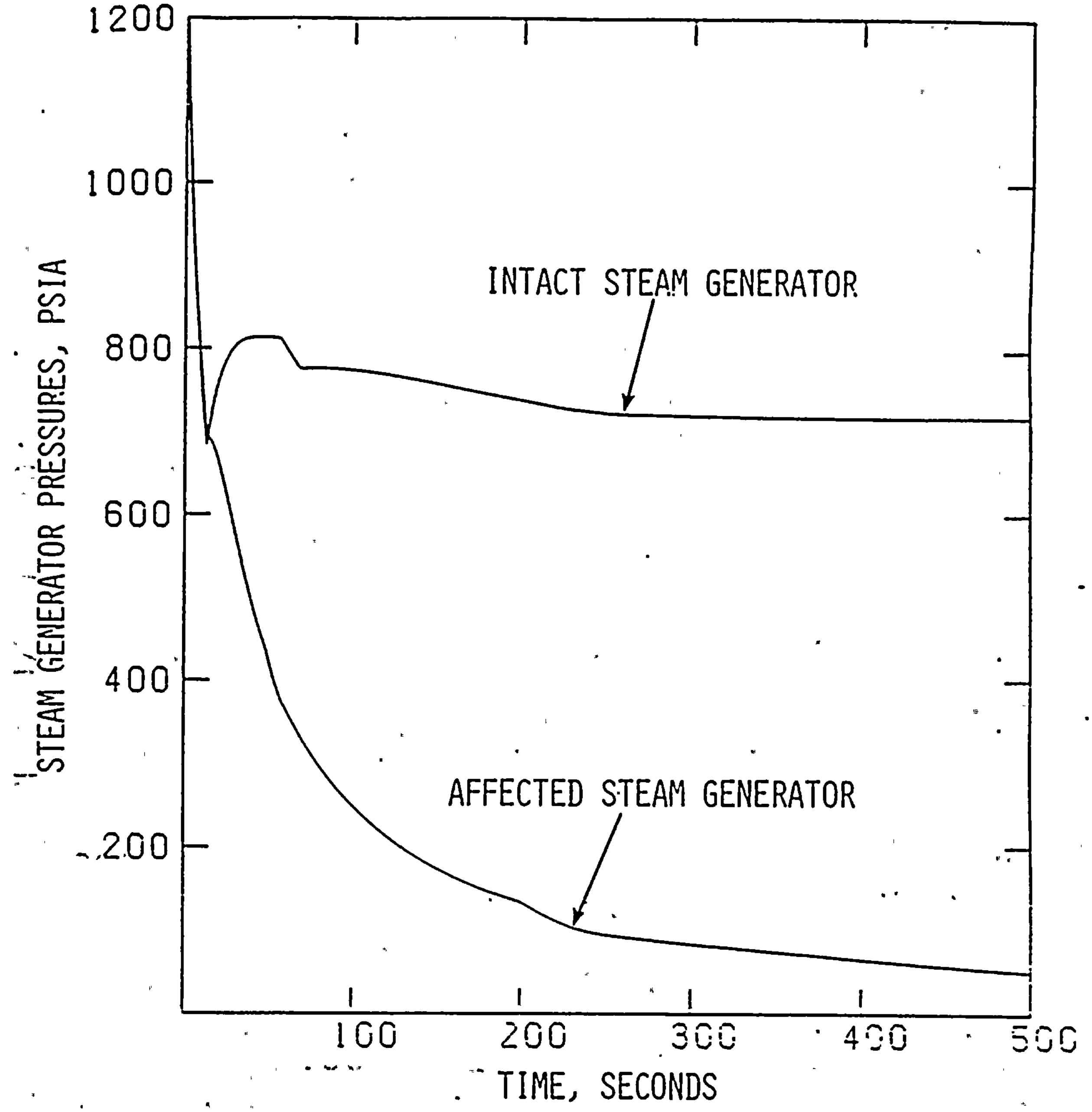
Figure 15.1.3-7



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER PRESSURIZER WATER VOLUME VS TIME

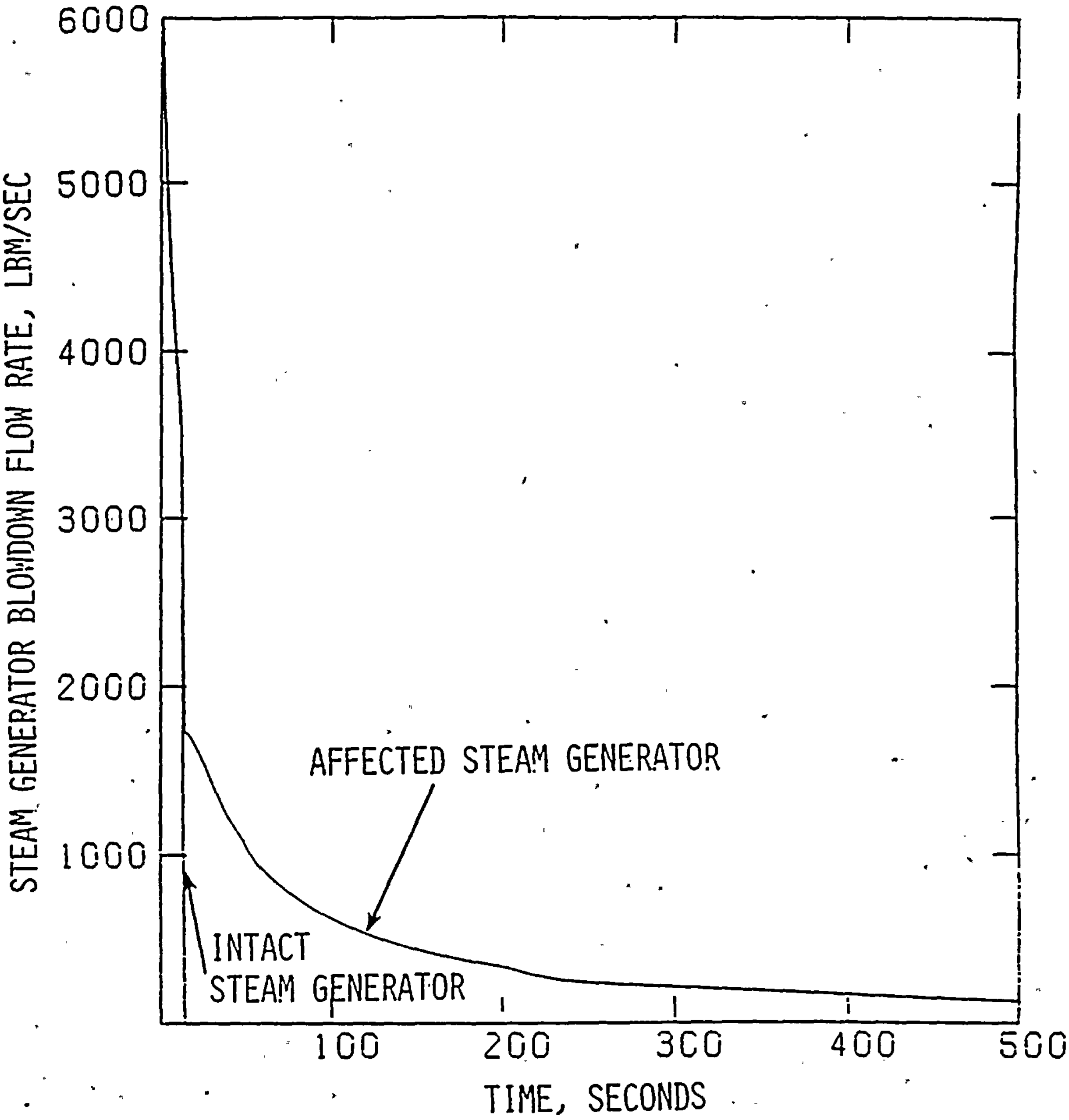
Figure 15.1.3-8



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR PRESSURES VS TIME

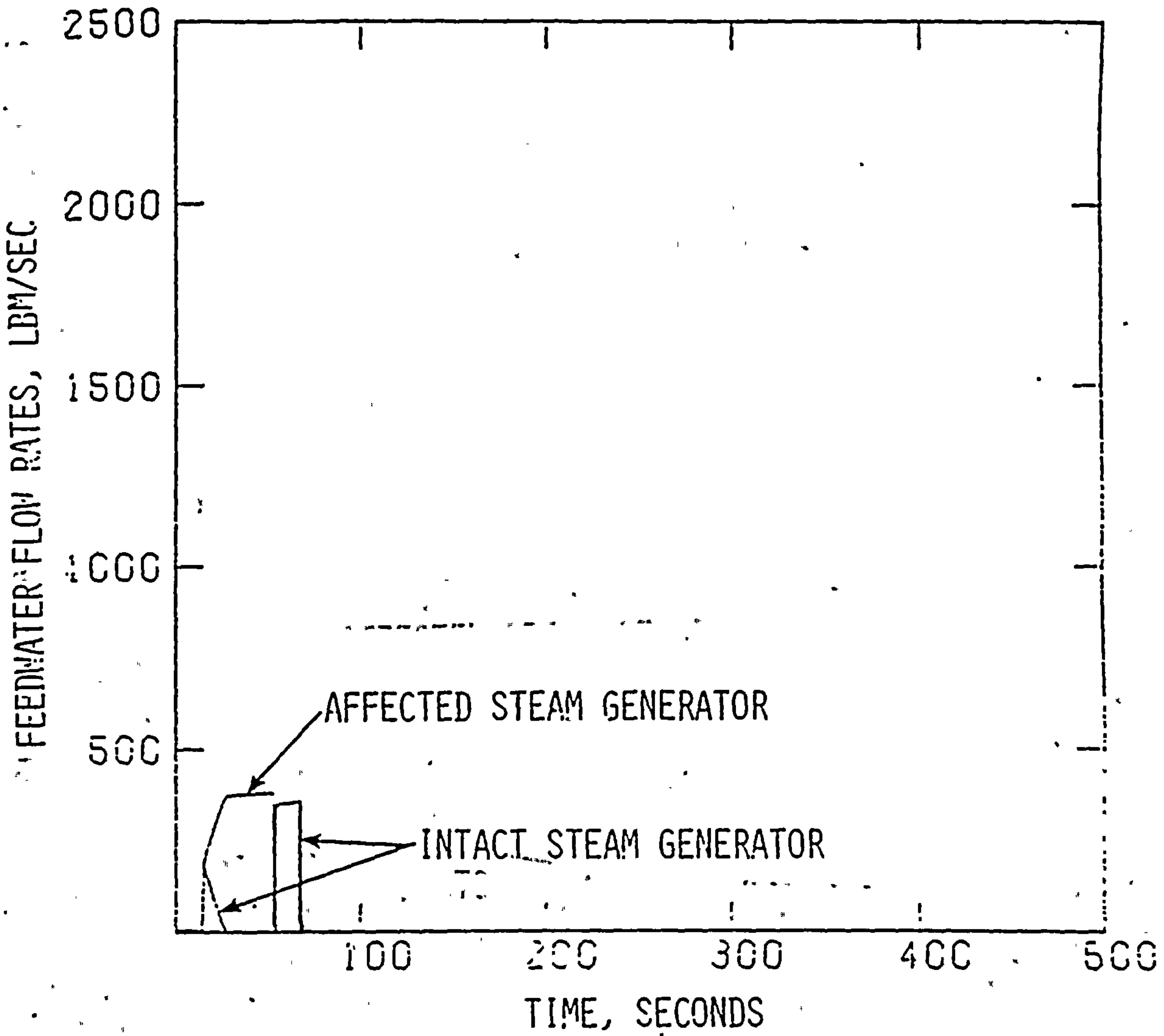
Figure 15.1.3-9



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR BLOWDOWN RATES VS TIME

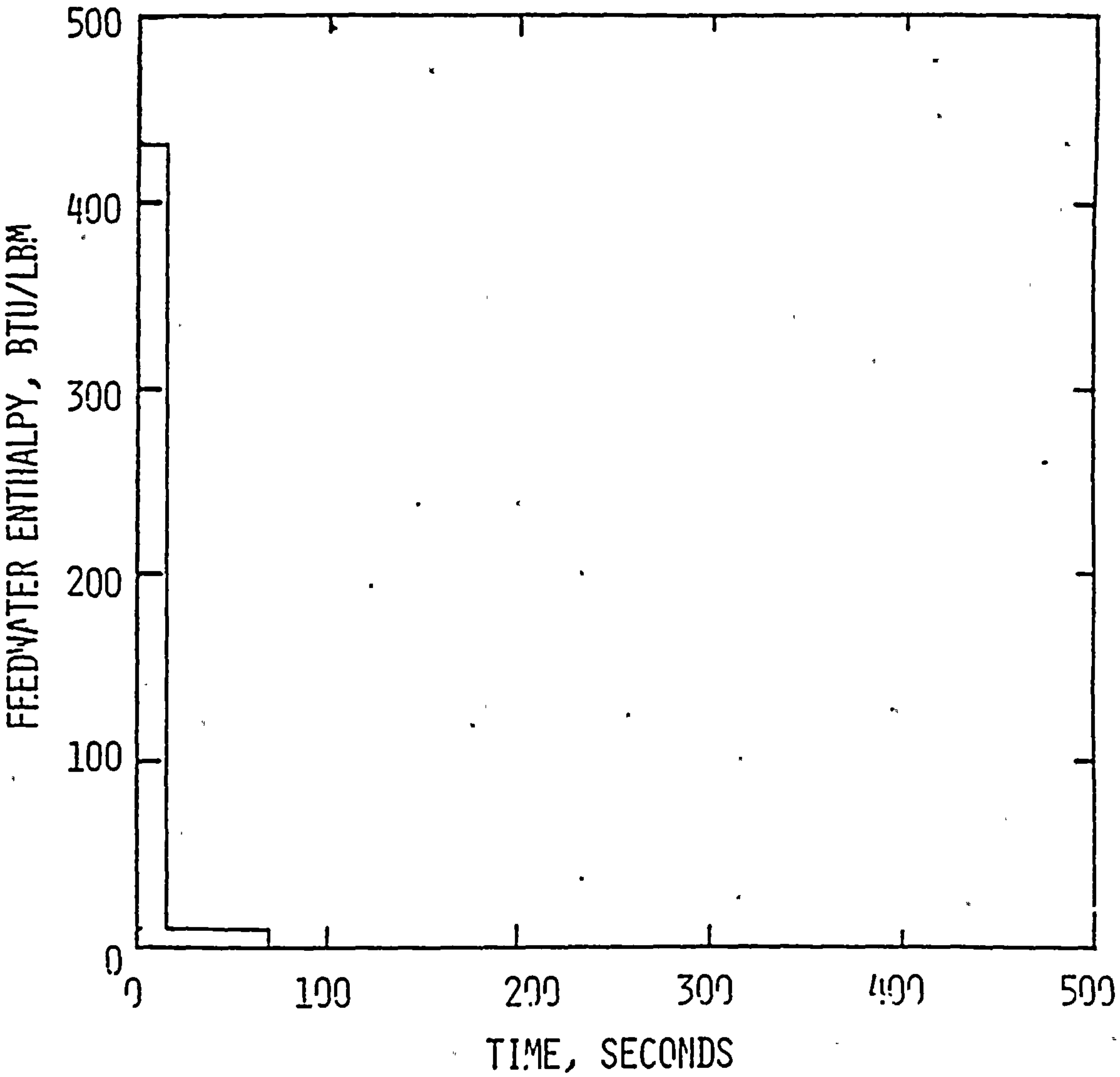
Figure 15.1.3-10



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER FEEDWATER FLOW RATES VS TIME

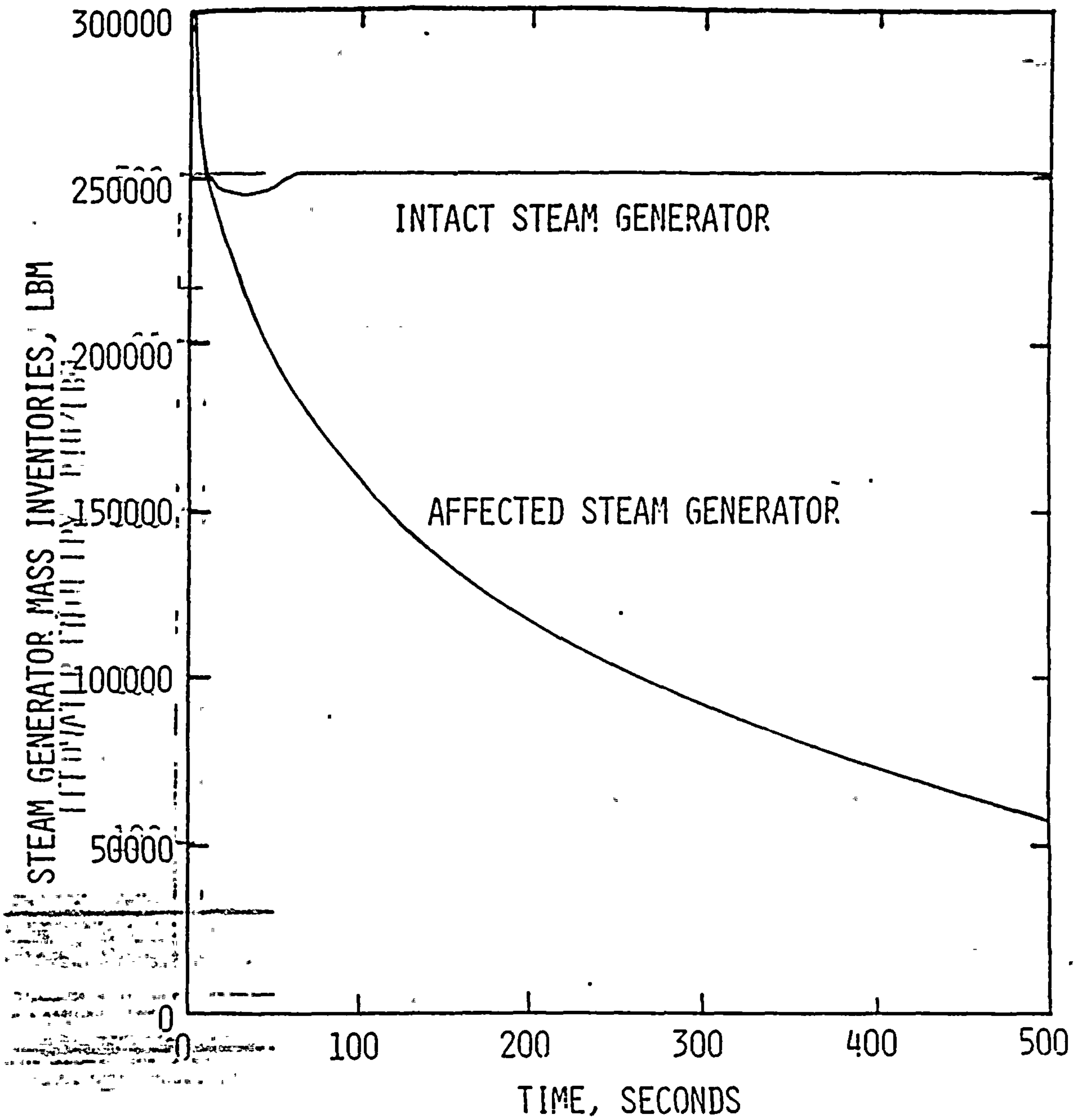
Figure 15.1.3-11



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER
FEEDWATER ENTHALPY VS TIME

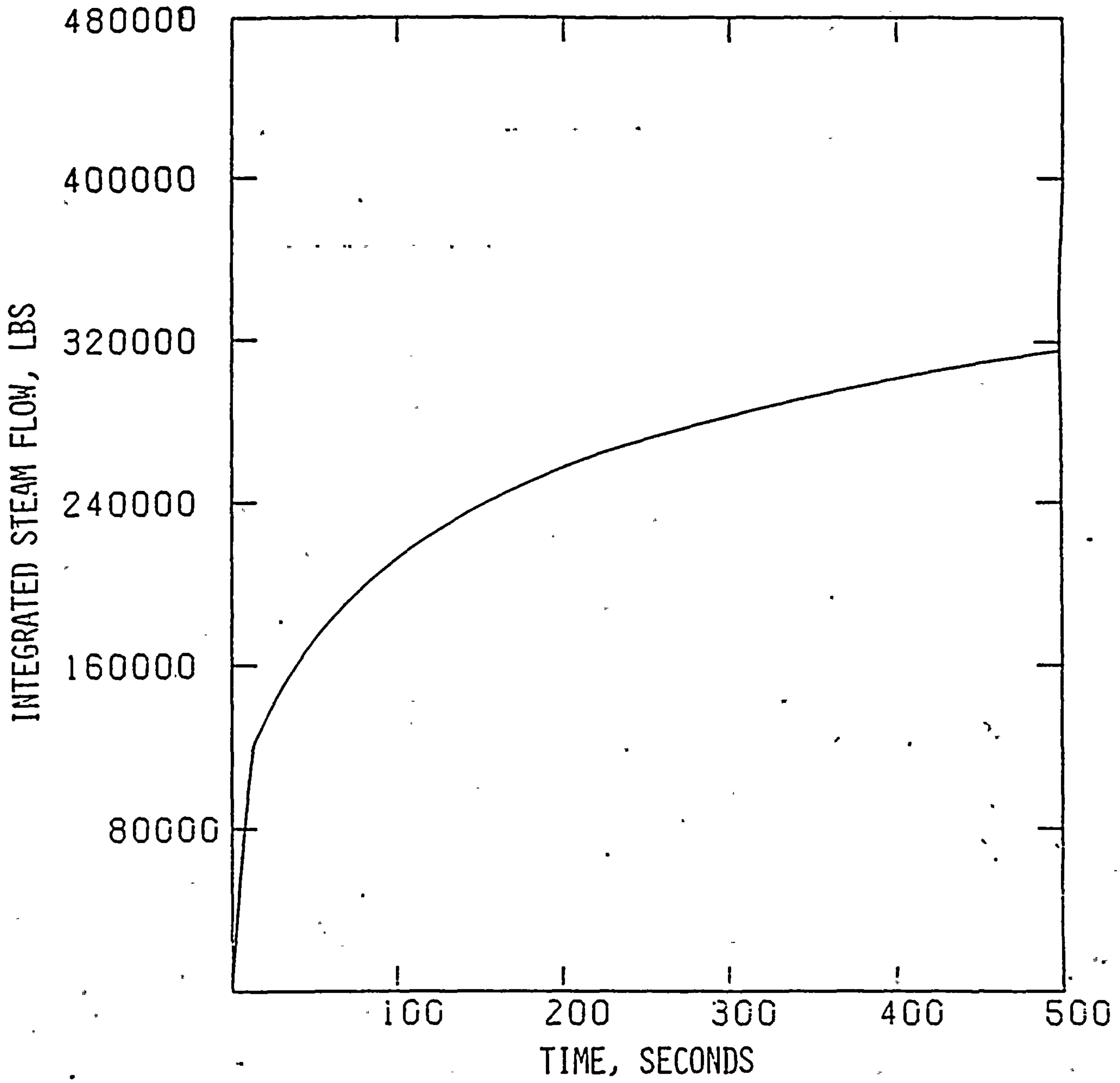
Figure 15.1.3-12



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER STEAM GENERATOR MASS INVENTORIES VS TIME

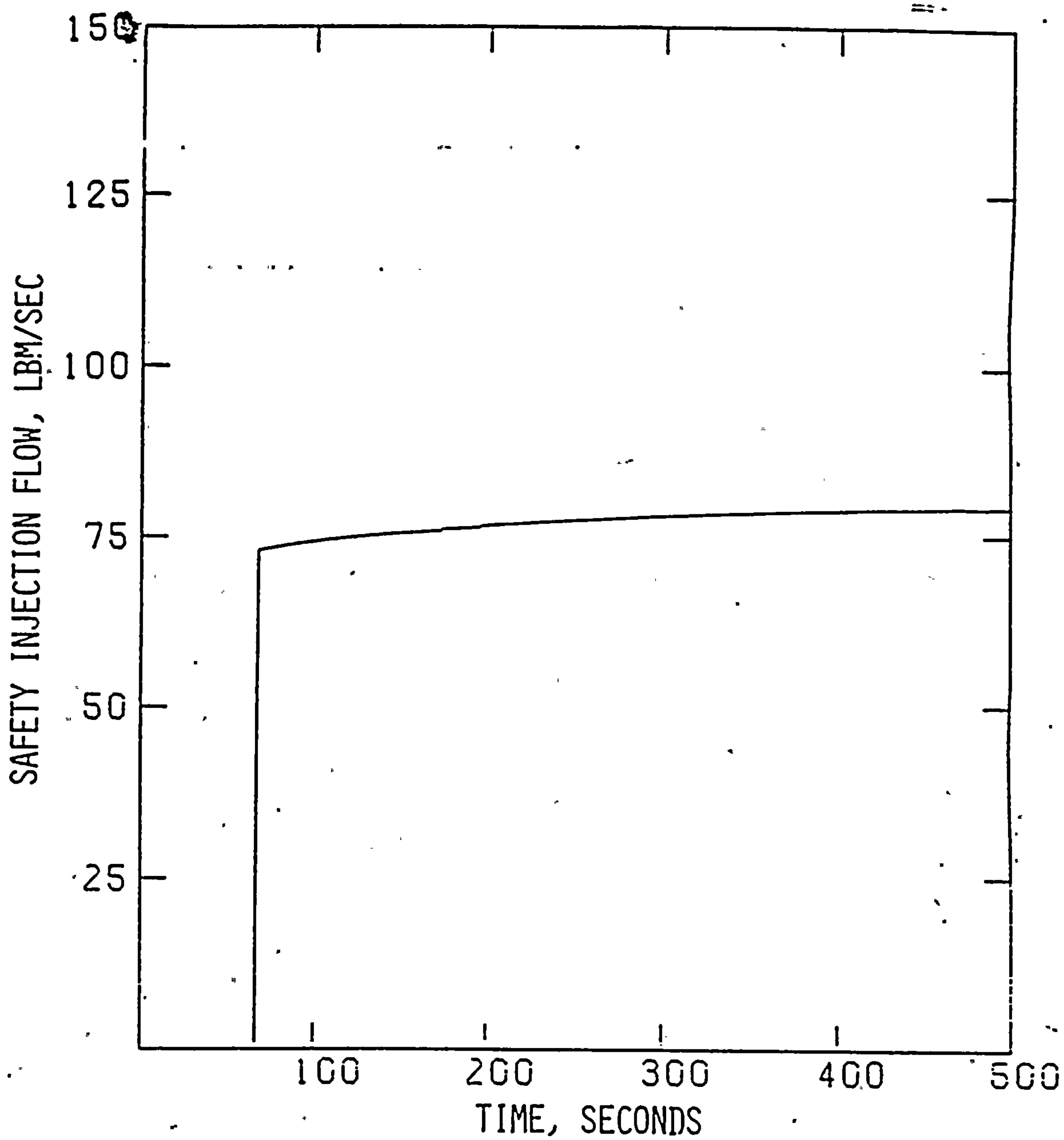
Figure 15.1.3-13



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER INTEGRATED STM MASS RELEASE THROUGH BREAK VS TIME

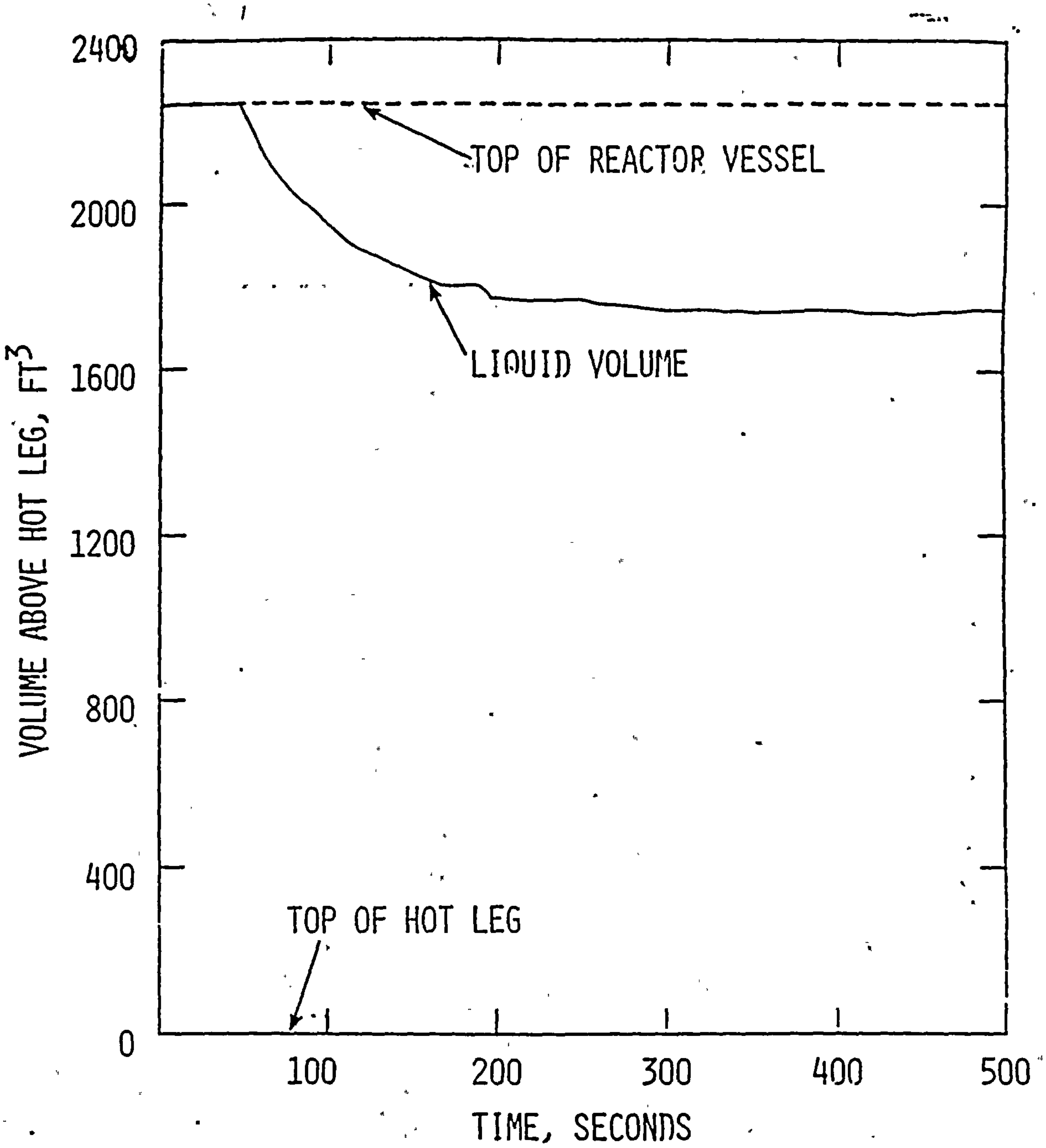
Figure 15.1.3-14



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER SAFETY INJECTION FLOW VS TIME

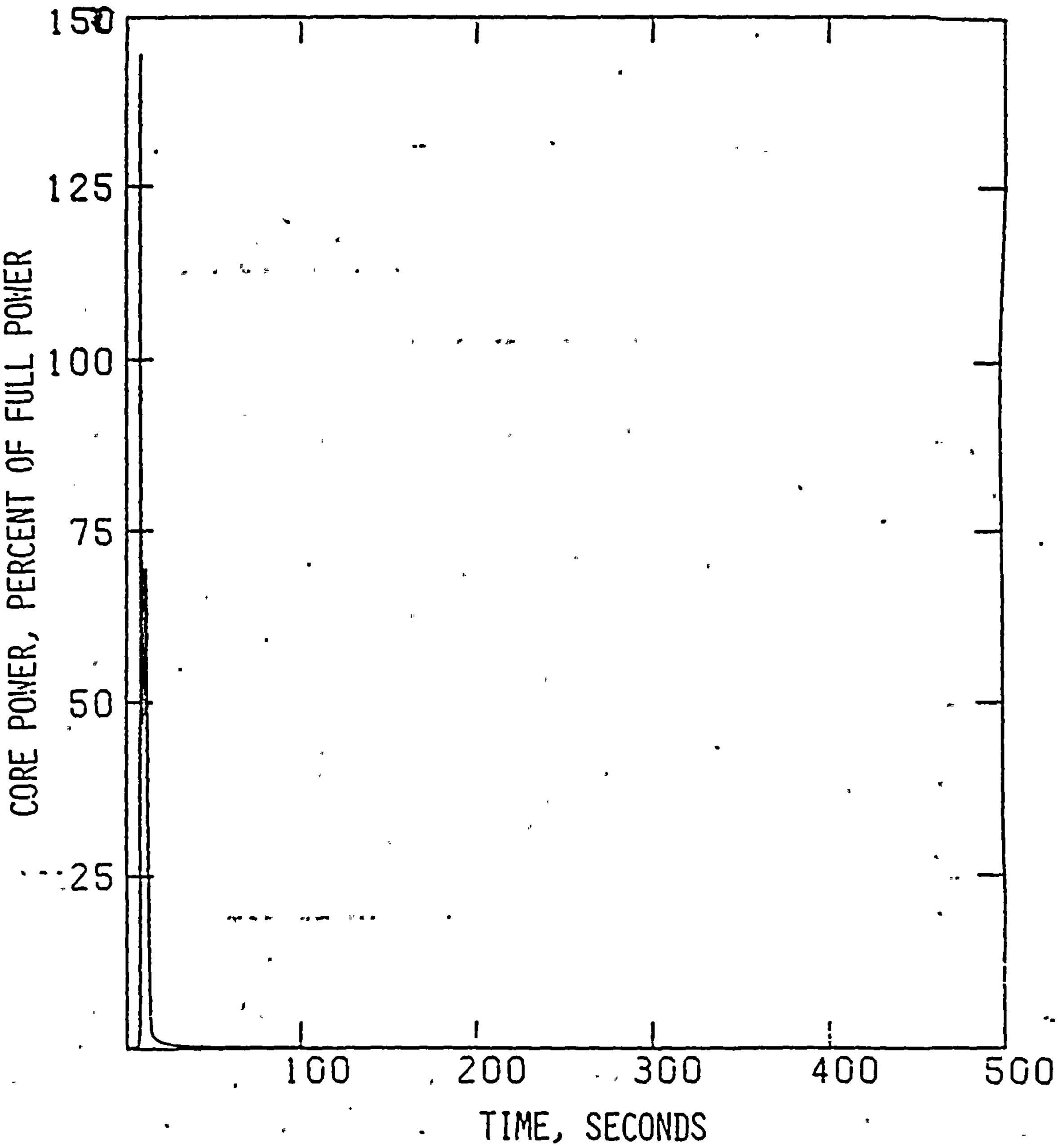
Figure 15.1.3-15



Palo Verde Nuclear Generating Station
FSAR

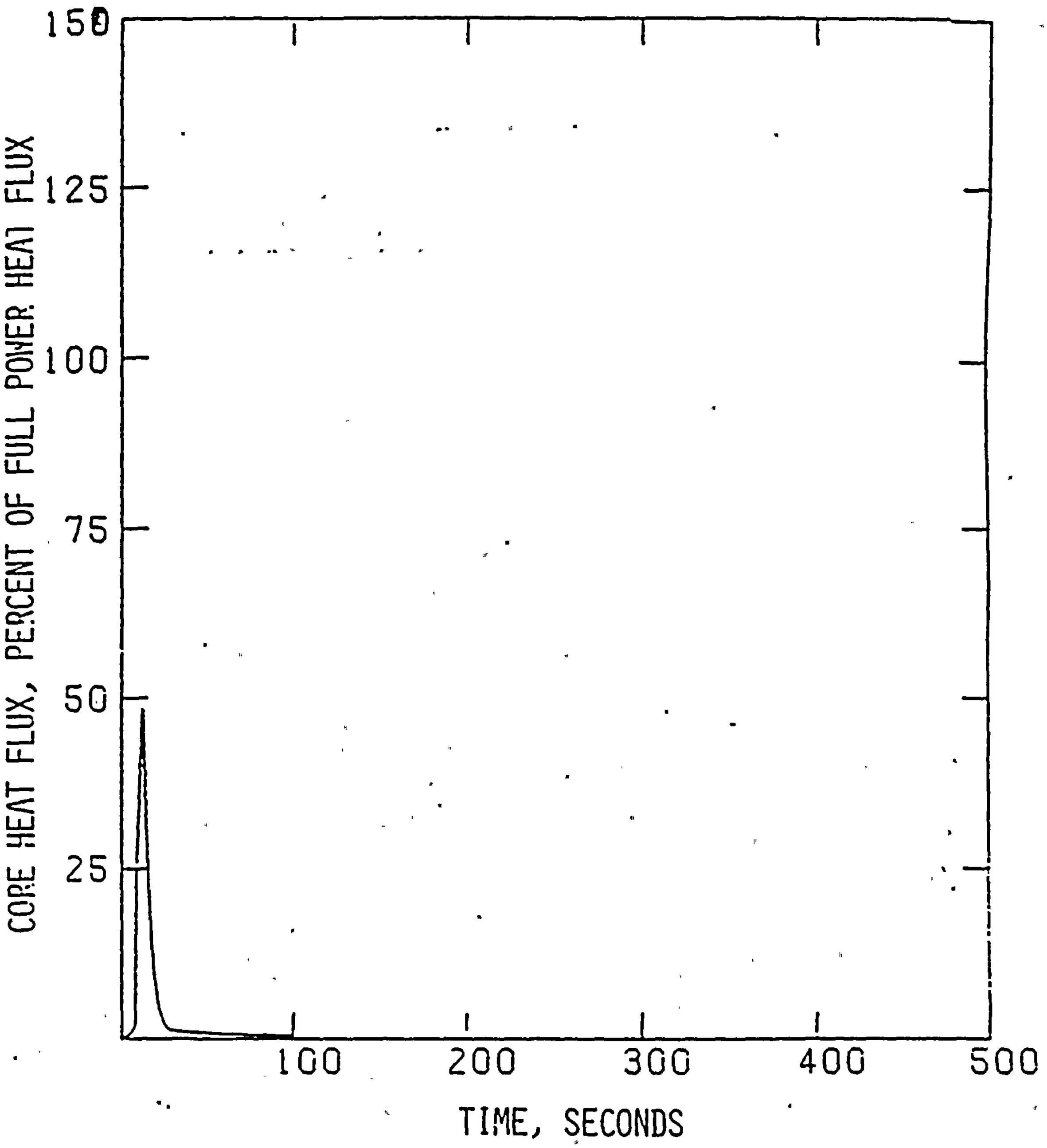
ZERO POWER LARGE STEAM LINE BREAK WITH CONCURRENT LOSS OF OFFSITE POWER REACTOR VESSEL LIQUID VOLUME VS TIME

Figure 15.1.3-16



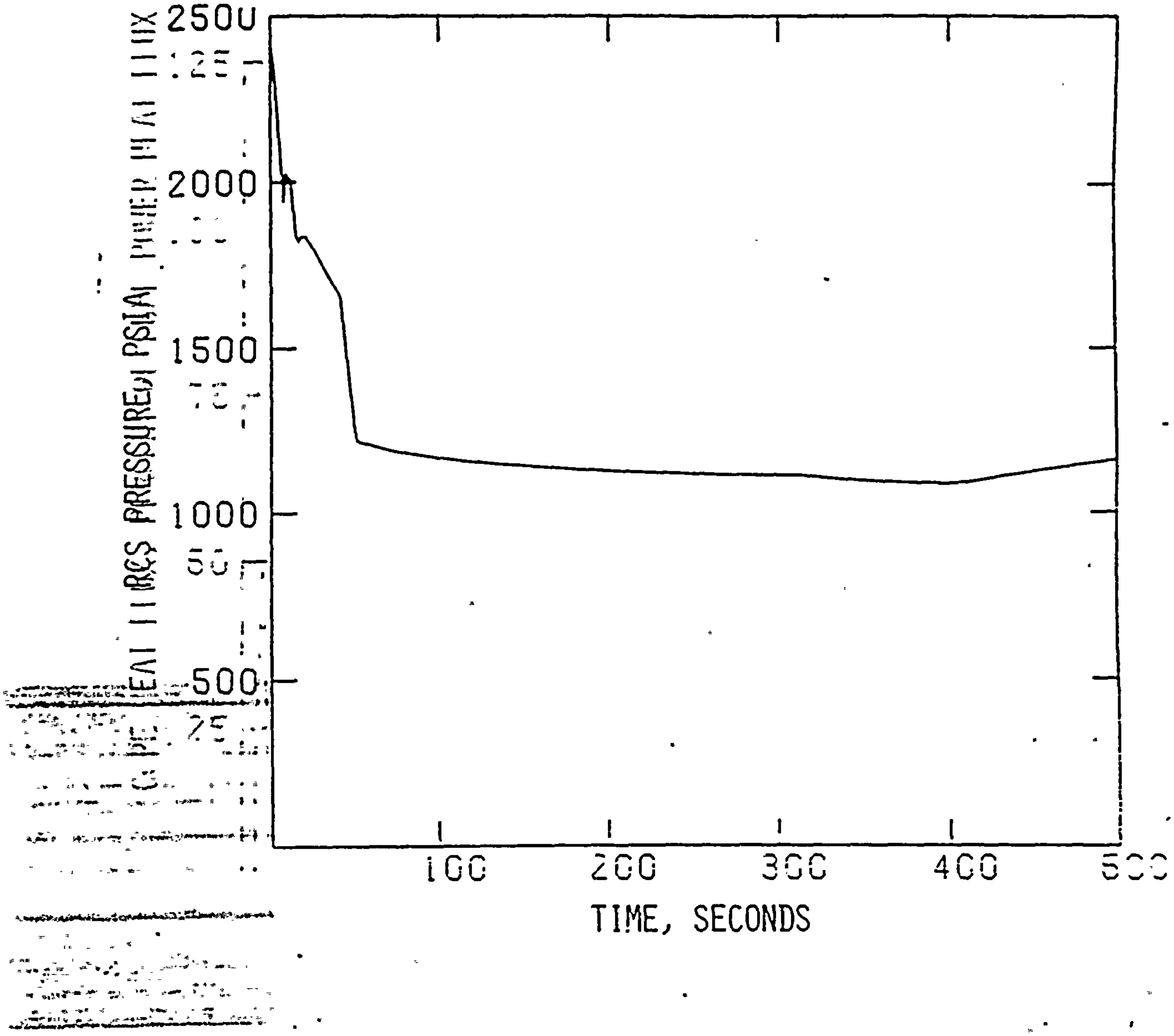
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
CORE POWER VS TIME
Figure 15.1.4-1



Palo Verde Nuclear Generating Station
FSAR

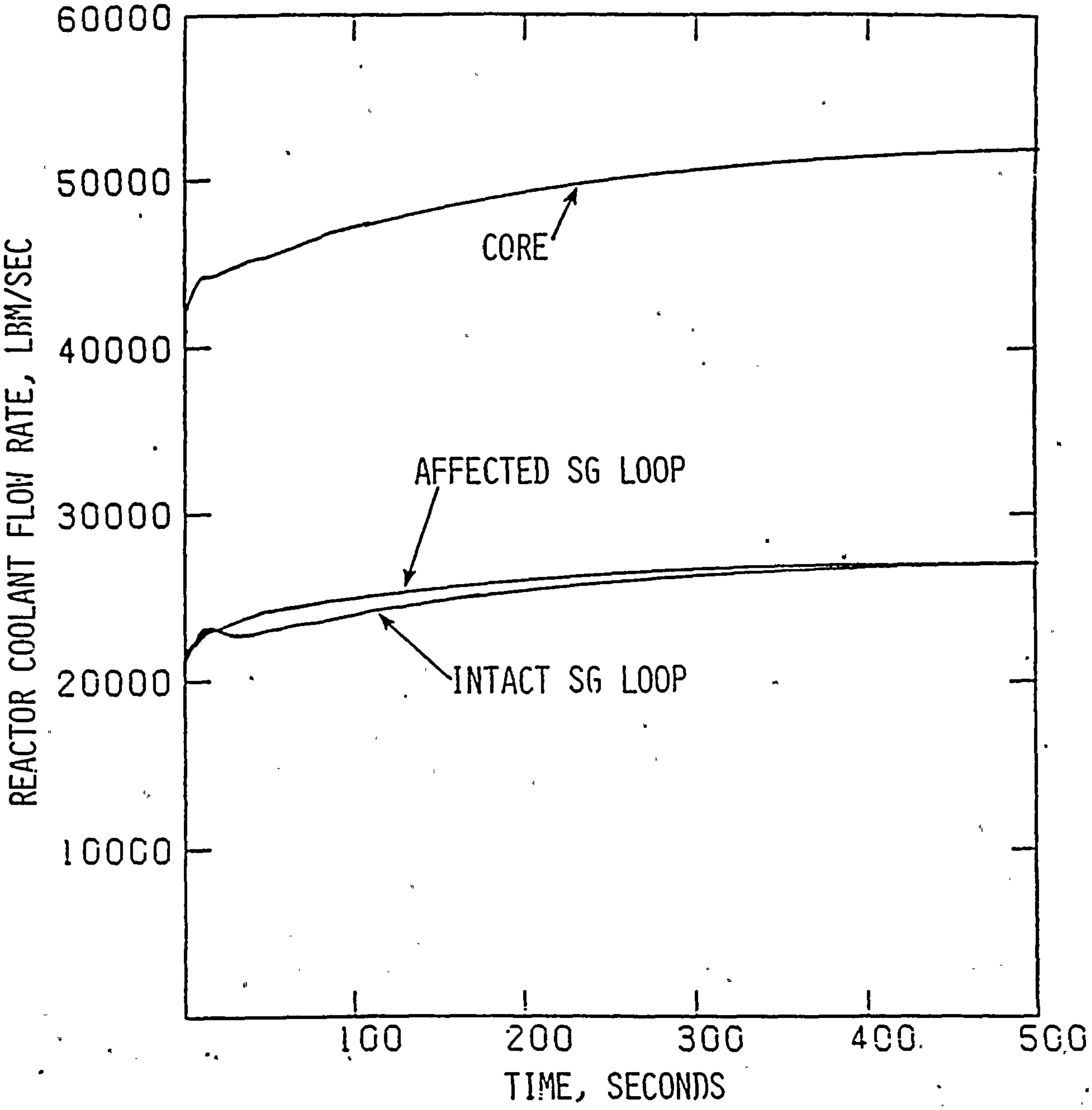
ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
CORE HEAT FLUX VS TIME
Figure 15.1.4-2



Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
RCS PRESSURE VS TIME

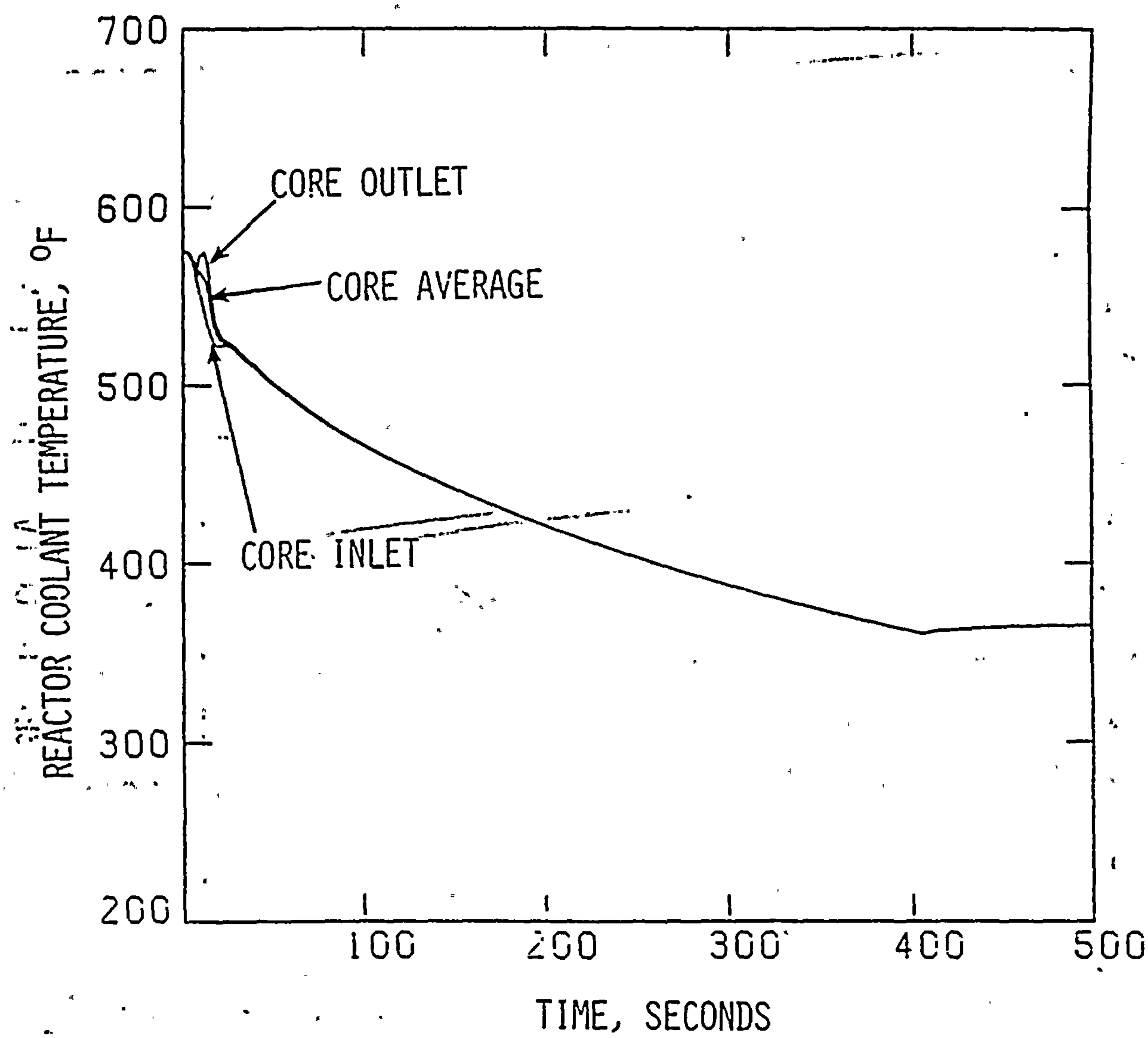
Figure 15.1.4-3



Palo Verde Nuclear Generating Station
FSAR

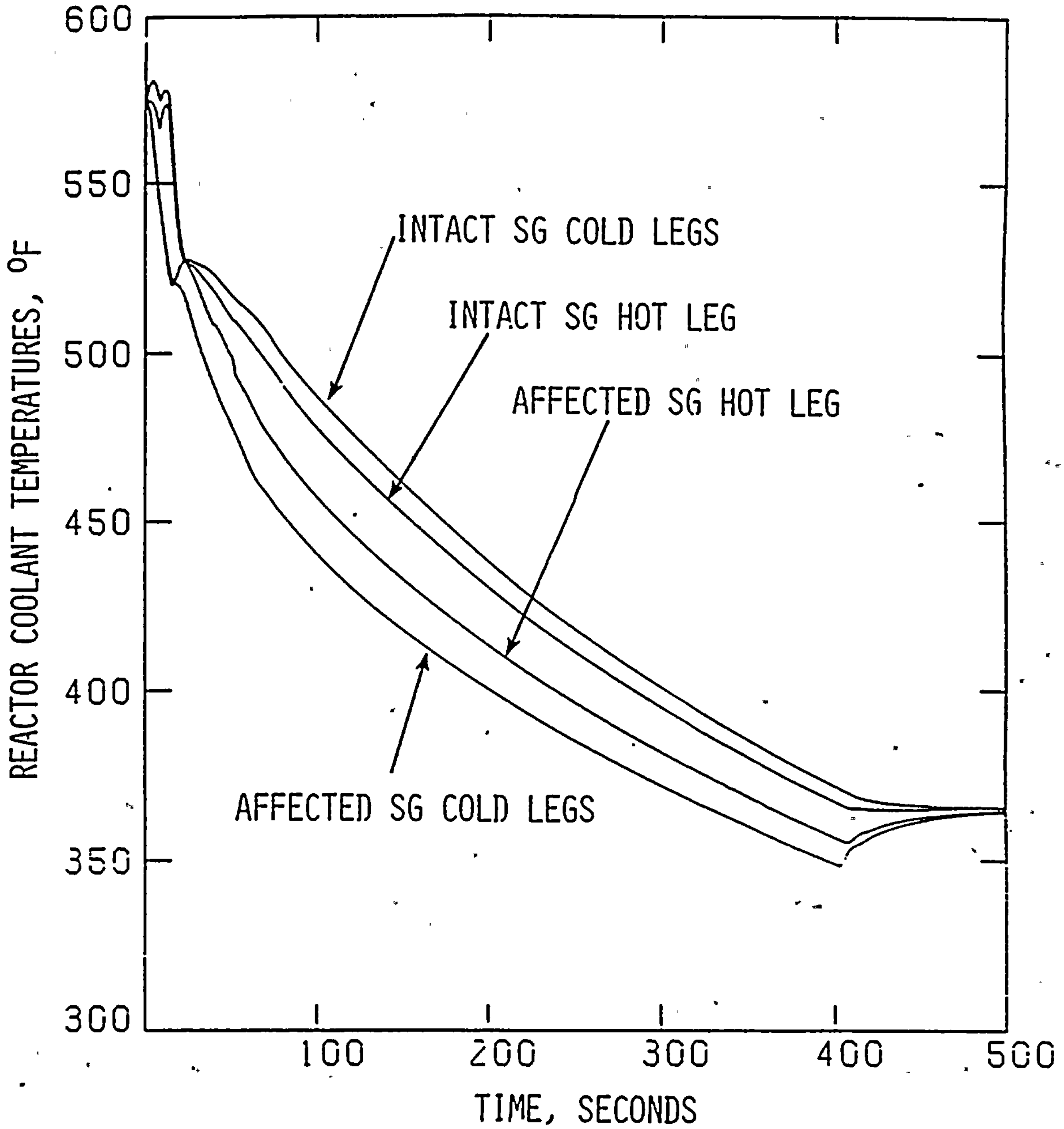
ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTOR COOLANT FLOW RATE VS TIME

Figure 15.1.4-4



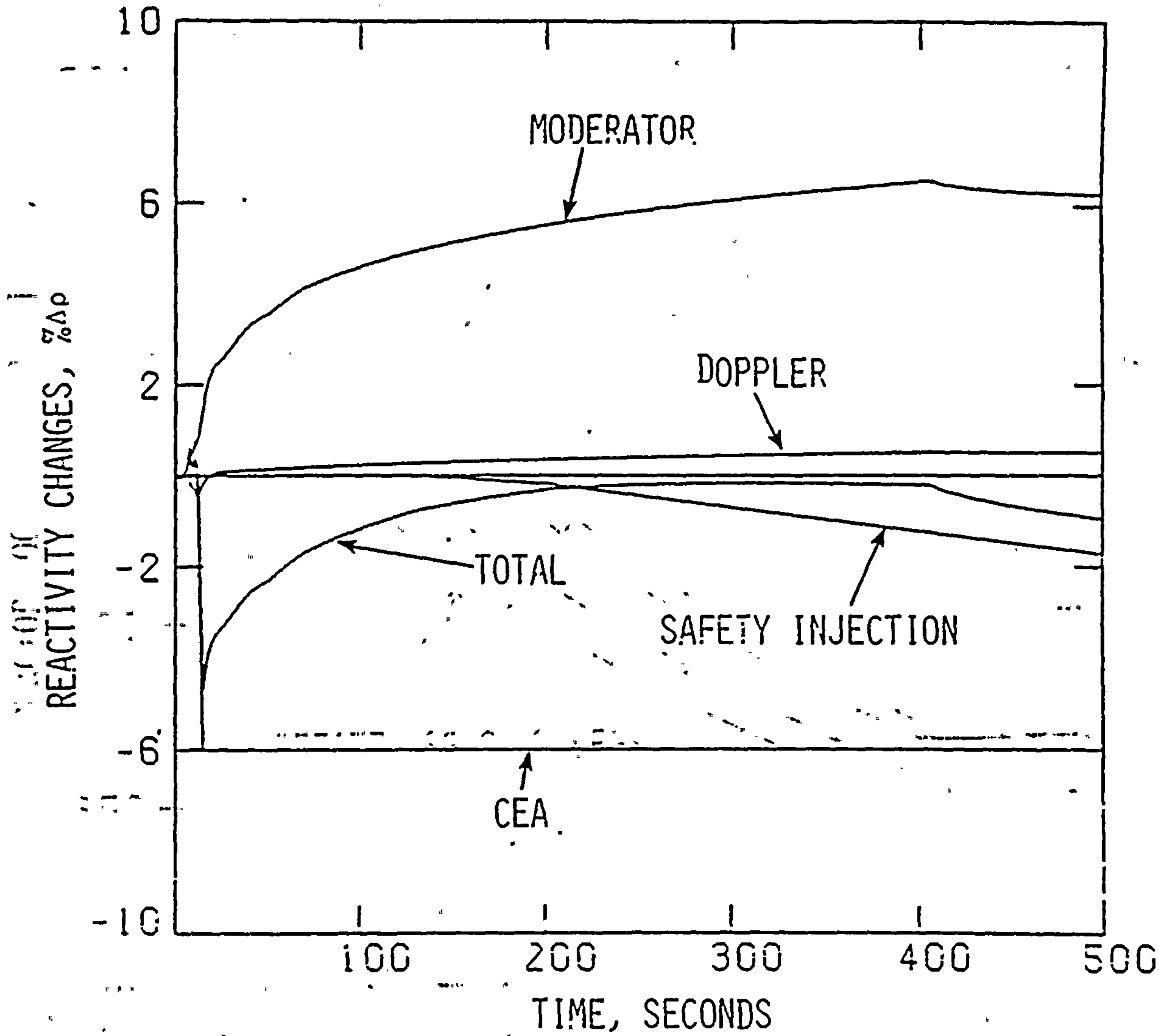
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTOR COOLANT TEMPERATURES VS TIME
Figure 15.1.4-5



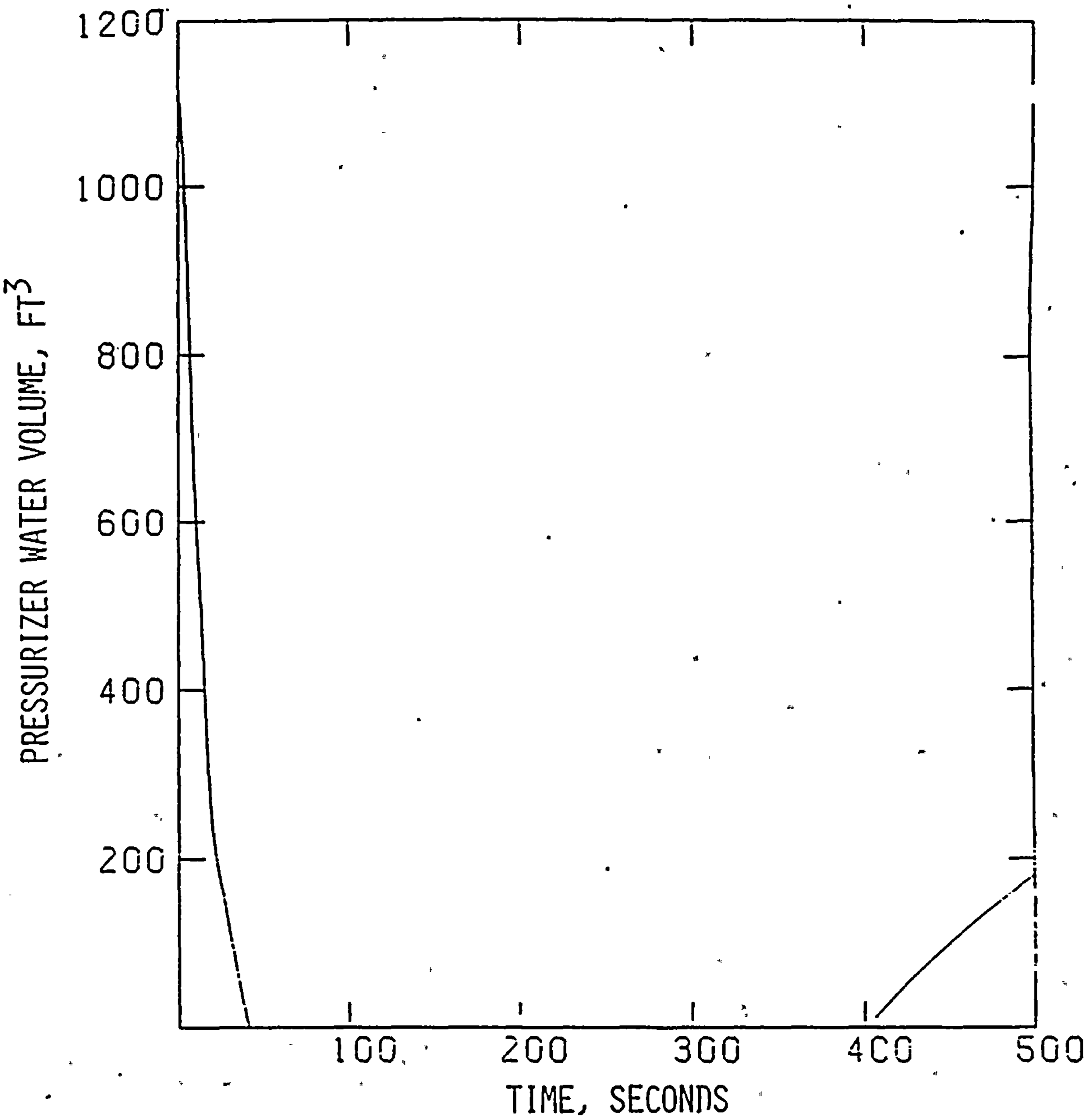
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTOR COOLANT TEMPERATURES VS TIME
Figure 15.1.4-6



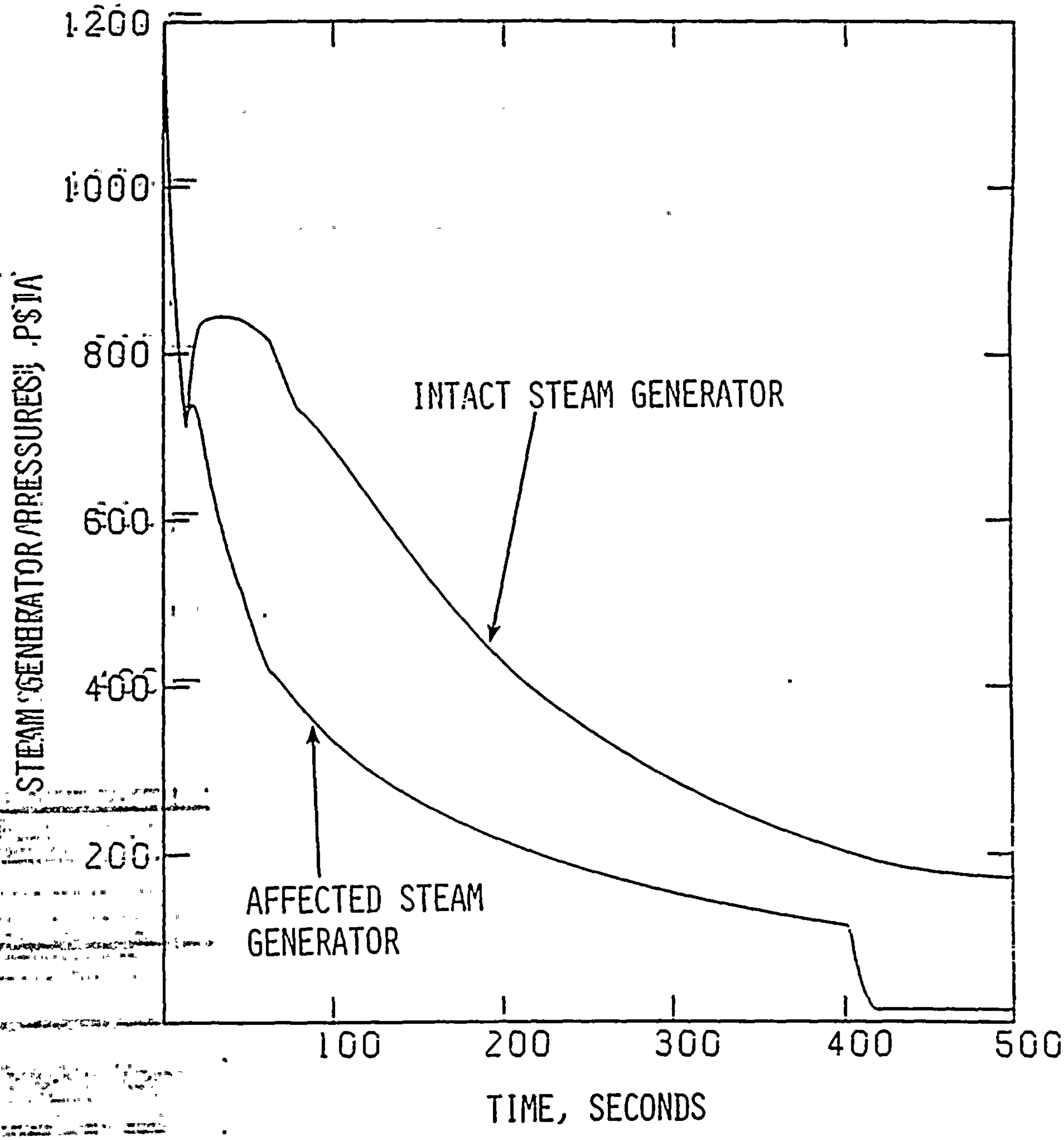
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTIVITY CHANGES VS TIME
Figure 15.1.4-7



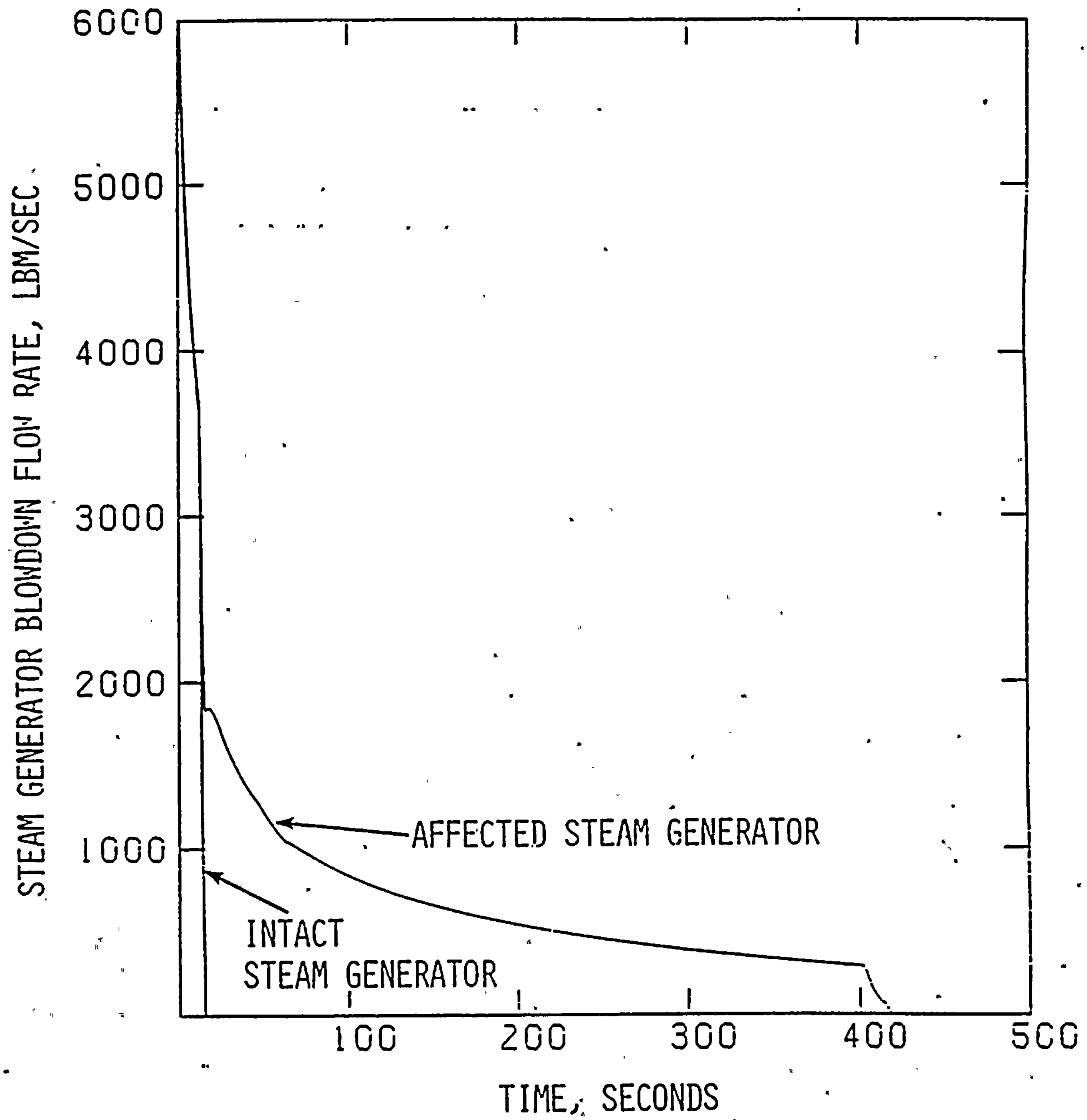
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
PRESSURIZER WATER VOLUME VS TIME
Figure 15.1.4-8



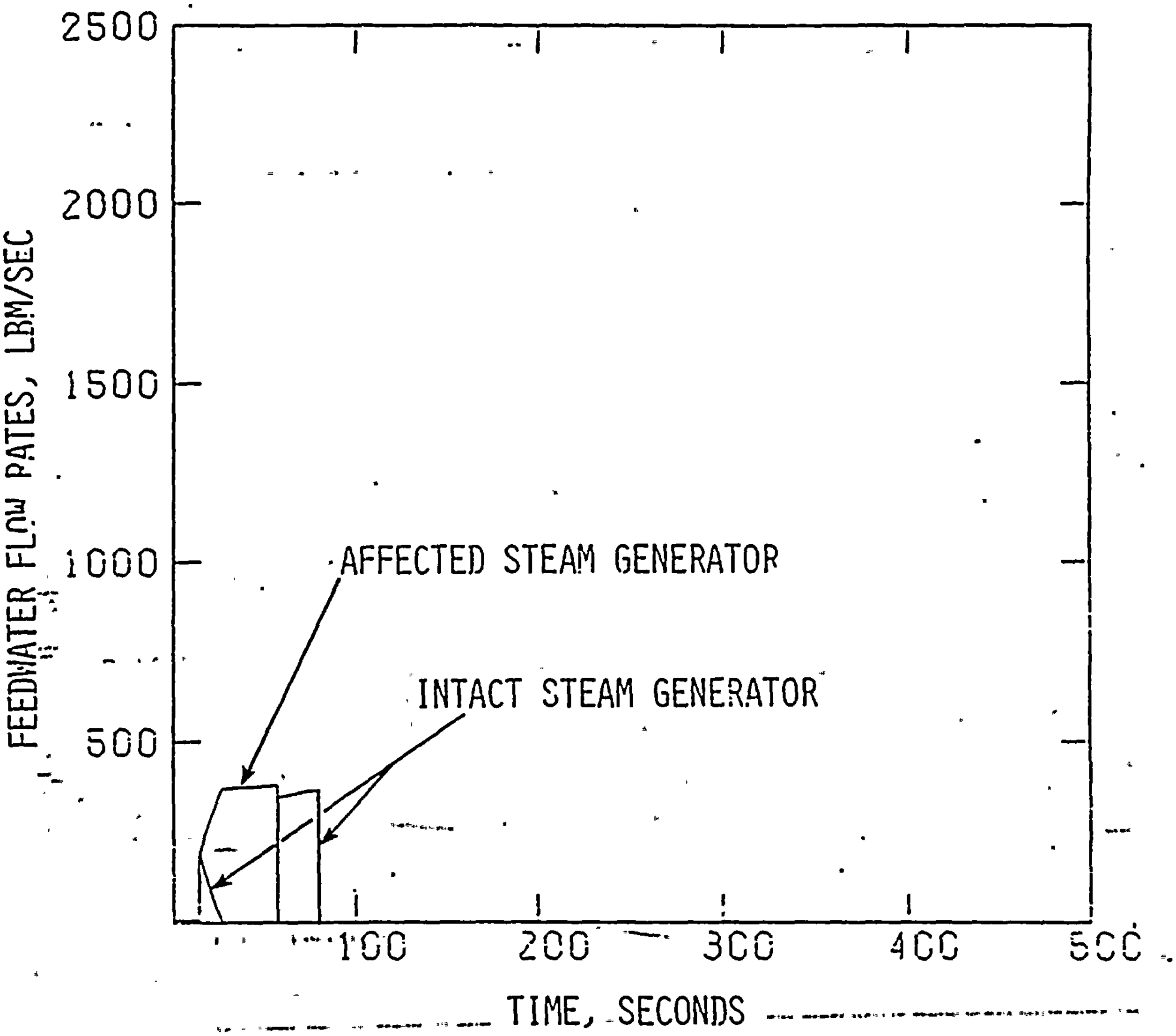
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
STEAM GENERATOR PRESSURES VS TIME
Figure 15.1.4-9



Palo Verde Nuclear Generating Station
FSAR

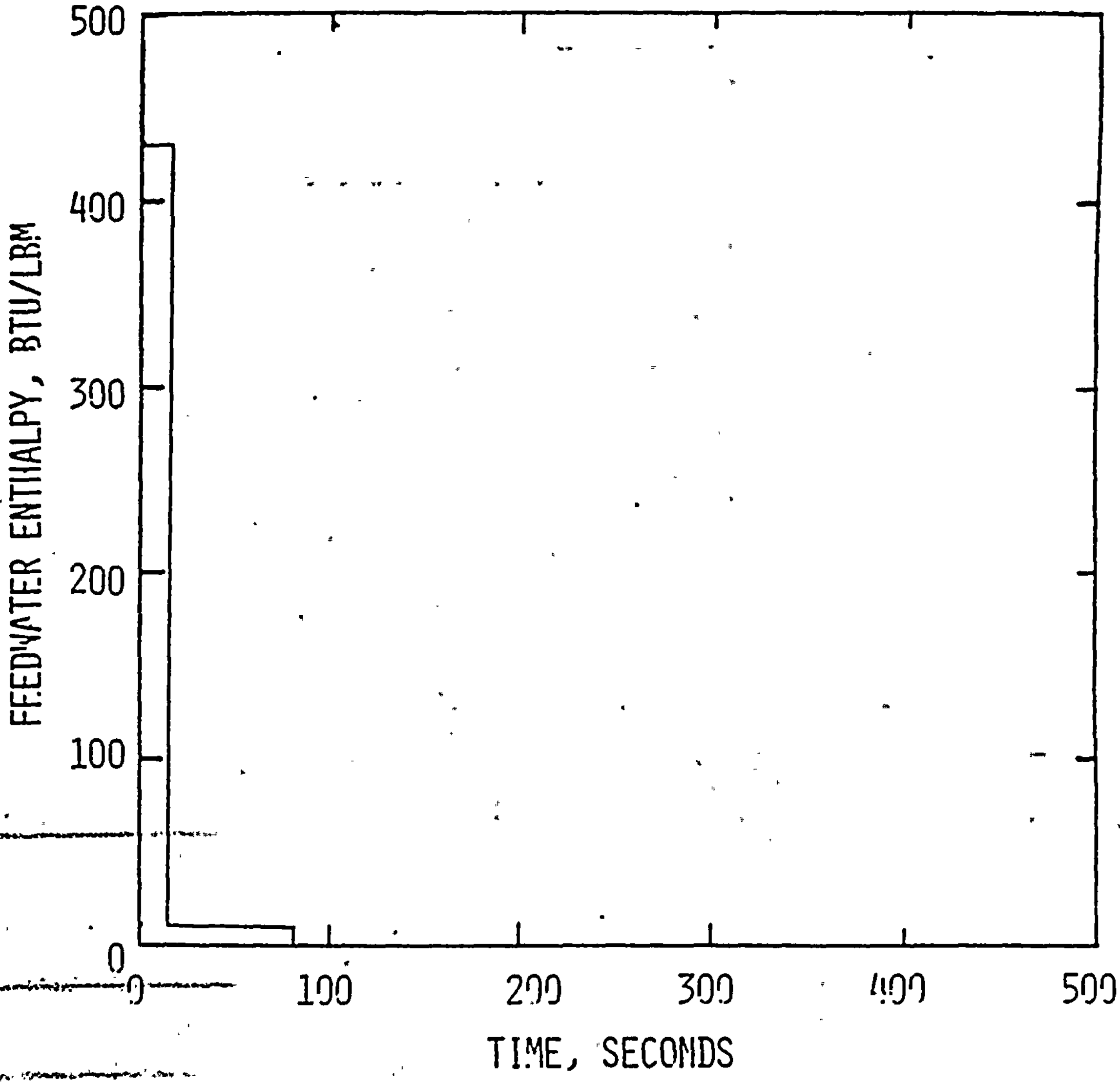
ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
STEAM GENERATOR BLOWDOWN RATES VS TIME
Figure 15.1.4-10



Palo Verde Nuclear Generating Station
FSAR

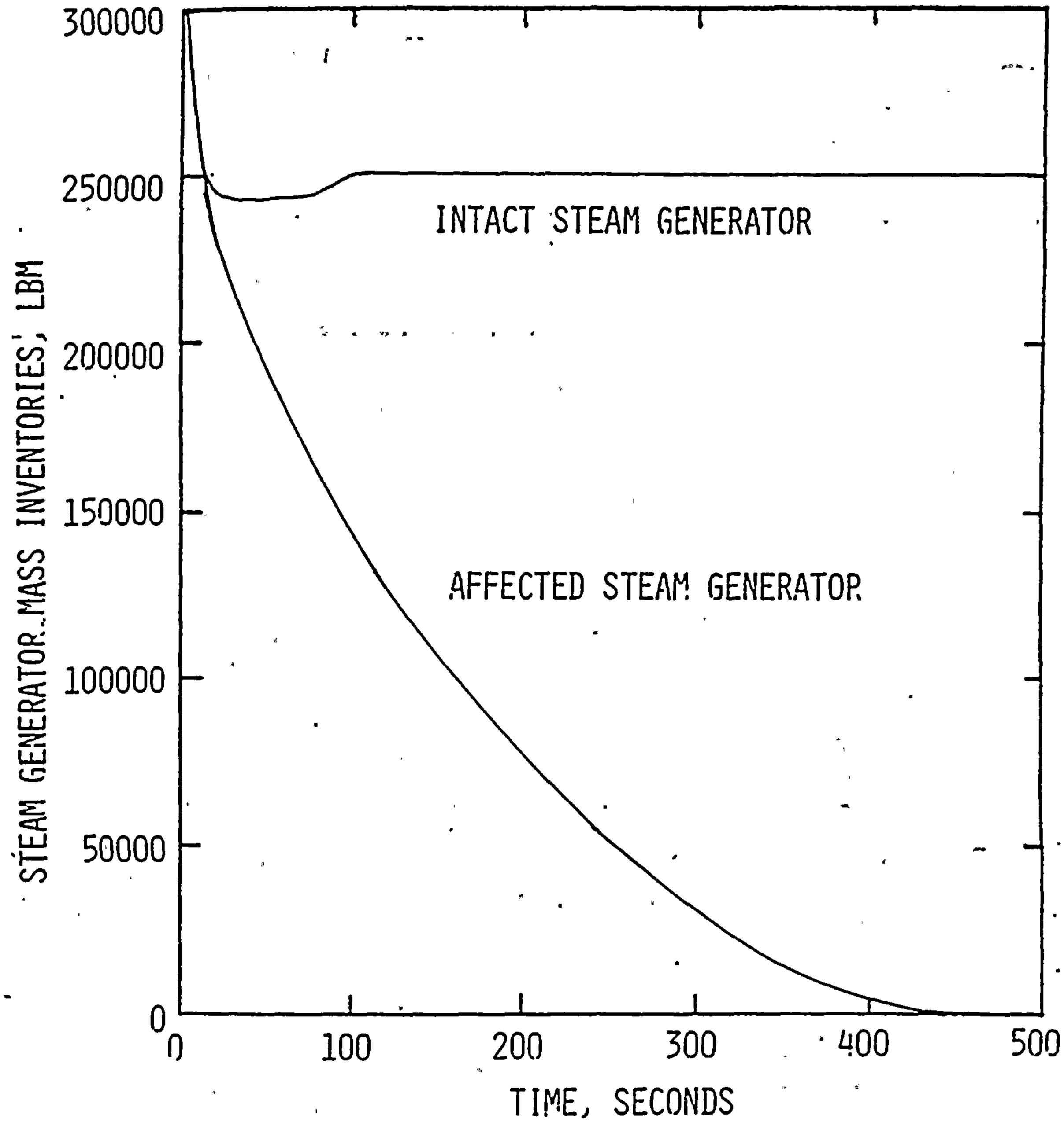
ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE FEEDWATER FLOW RATES VS TIME

Figure 15.1.4-11



Palo Verde Nuclear Generating Station
FSAR

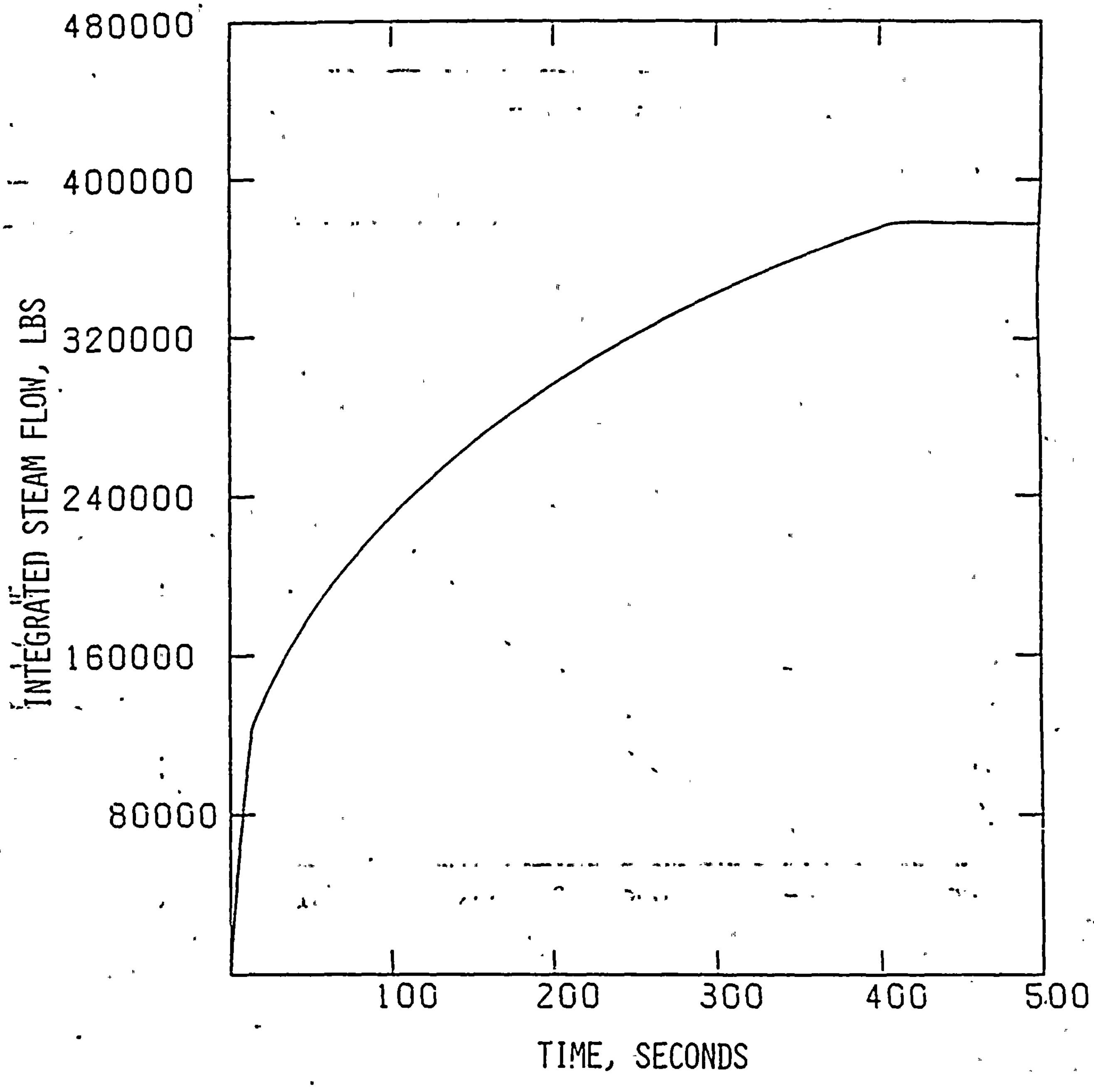
ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
FEEDWATER ENTHALPY VS TIME
Figure 15.1.4-12



Palo Verde Nuclear Generating Station
FSAR

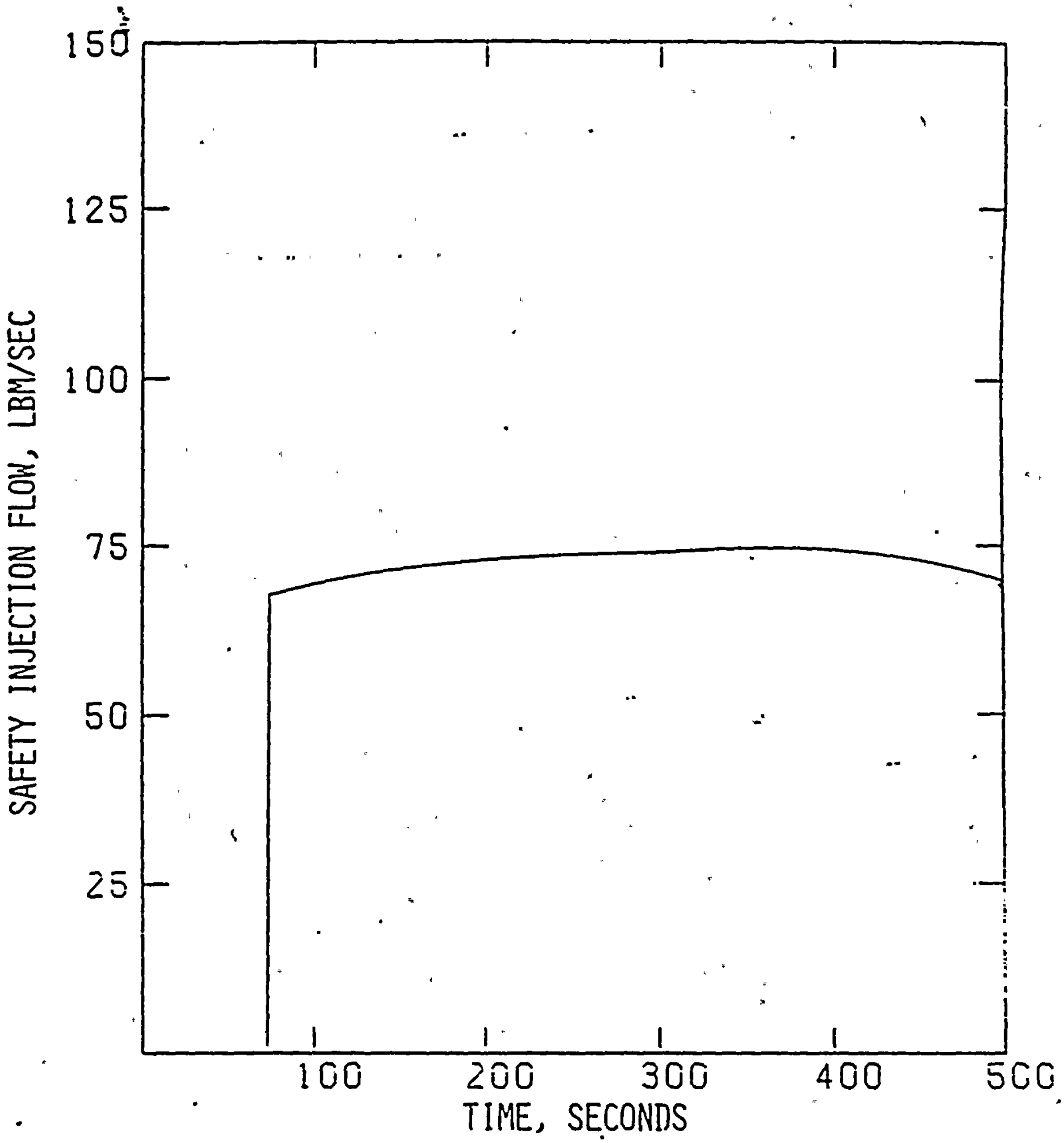
ZERO POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
STEAM GENERATOR MASS INVENTORIES VS TIME

Figure 15.1.4-13



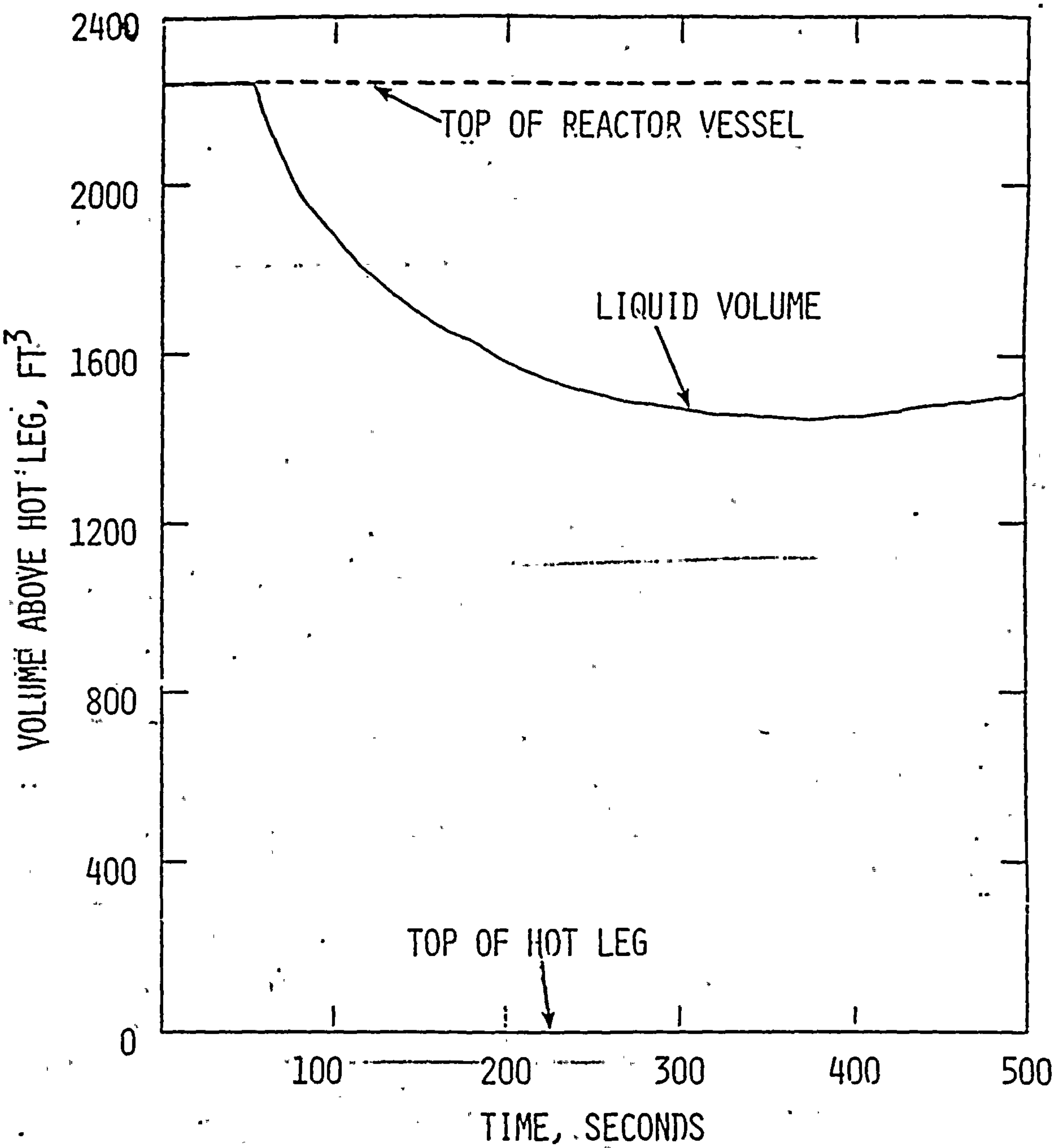
Palo Verde Nuclear Generating Station
FSAR

ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
INTEGRATED STEAM RELEASE VS TIME
Figure 15.1.4-14



Palo Verde Nuclear Generating Station
FSAR

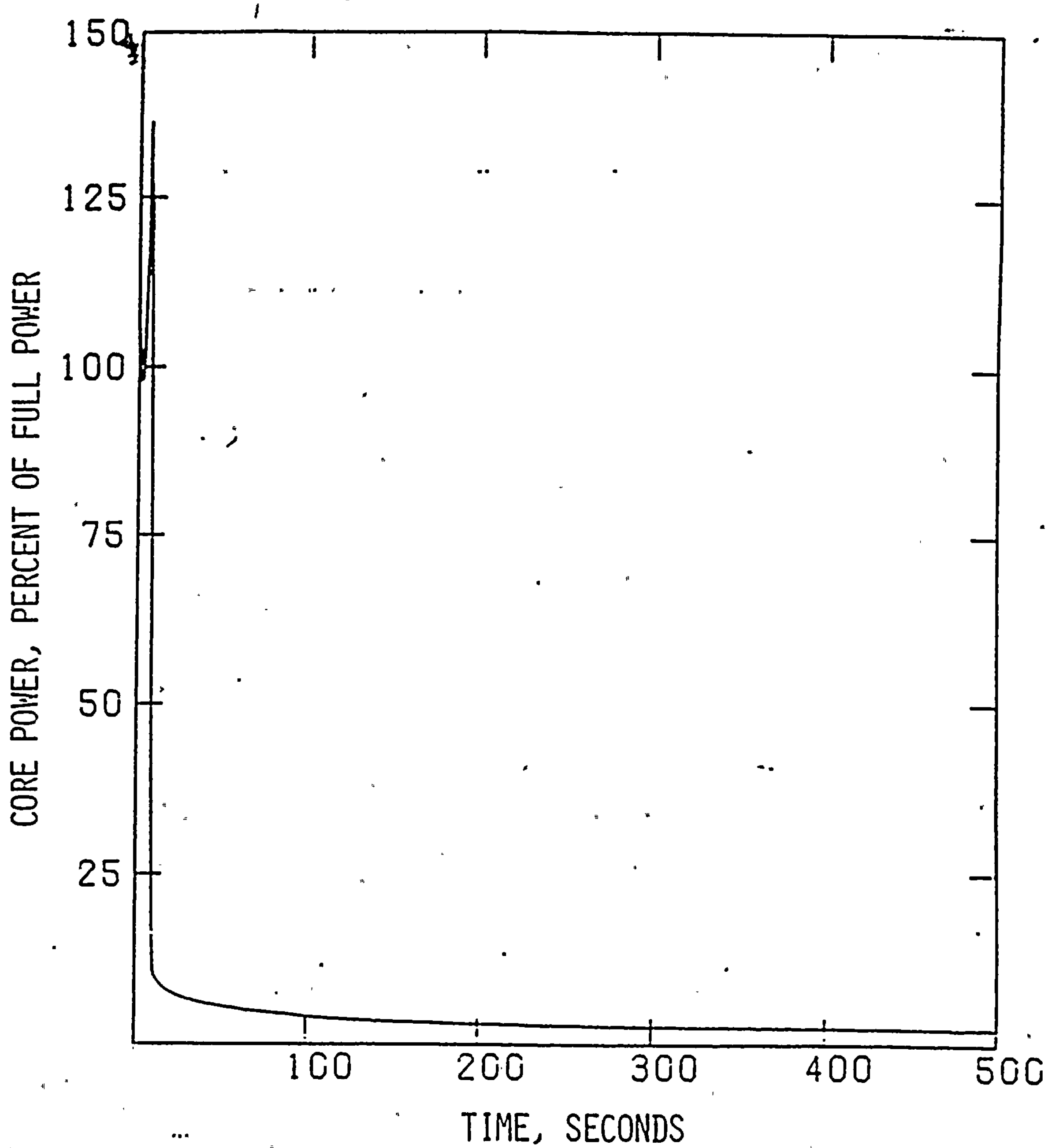
ZERO POWER LARGE STEAM LINE BREAK WITH
OFFSITE POWER AVAILABLE
SAFETY INJECTION FLOW VS TIME
Figure 15.1.4-15



Palo Verde Nuclear Generating Station
FSAR

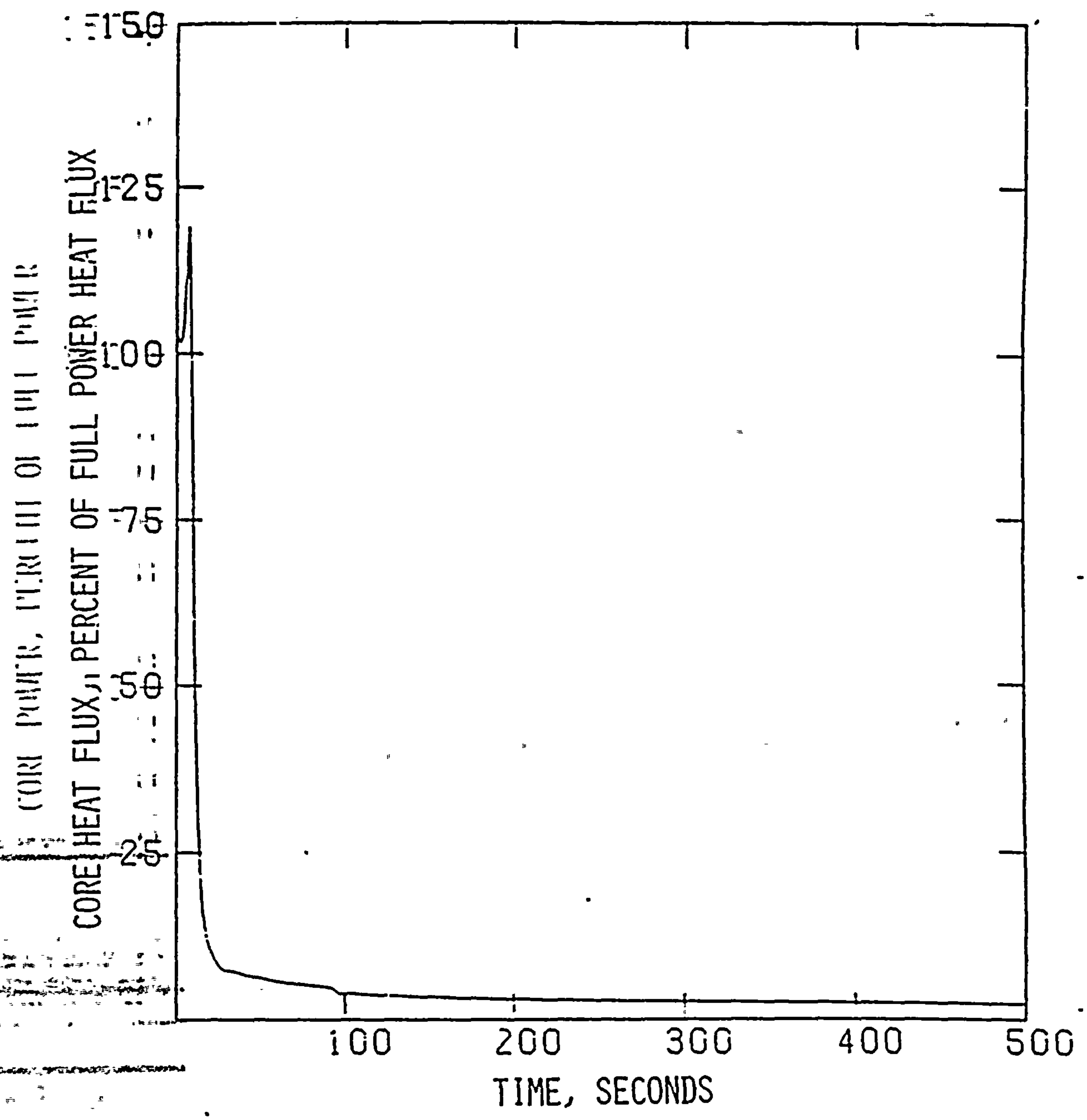
ZERO POWER LARGE STEAM LINE BREAK WITH OFFSITE POWER AVAILABLE
REACTOR VESSEL LIQUID VOLUME VS TIME

Figure 15.1.4-16



Palo Verde Nuclear Generating Station
FSAR

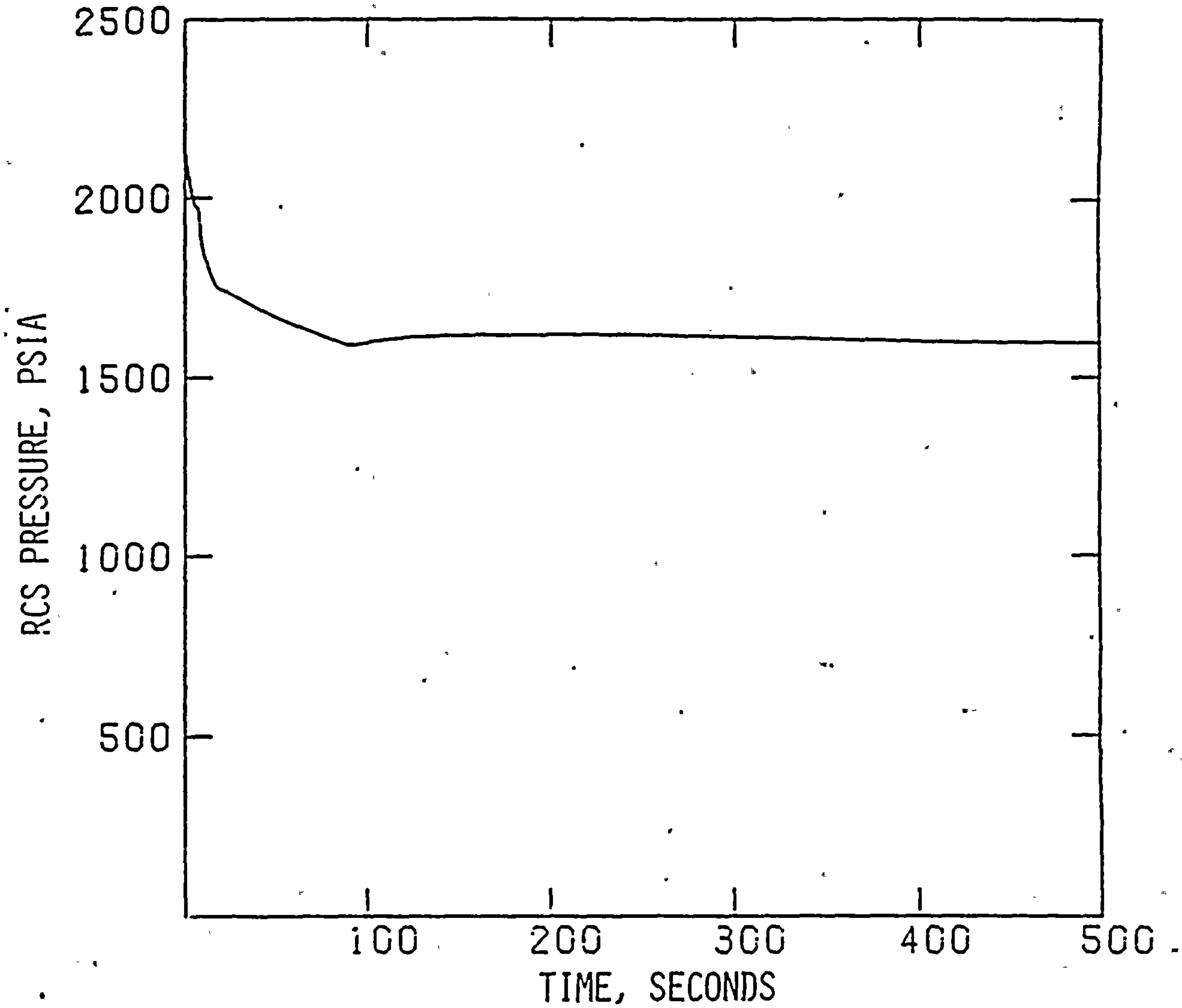
FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
CORE POWER VS TIME
Figure 15.1.5-1



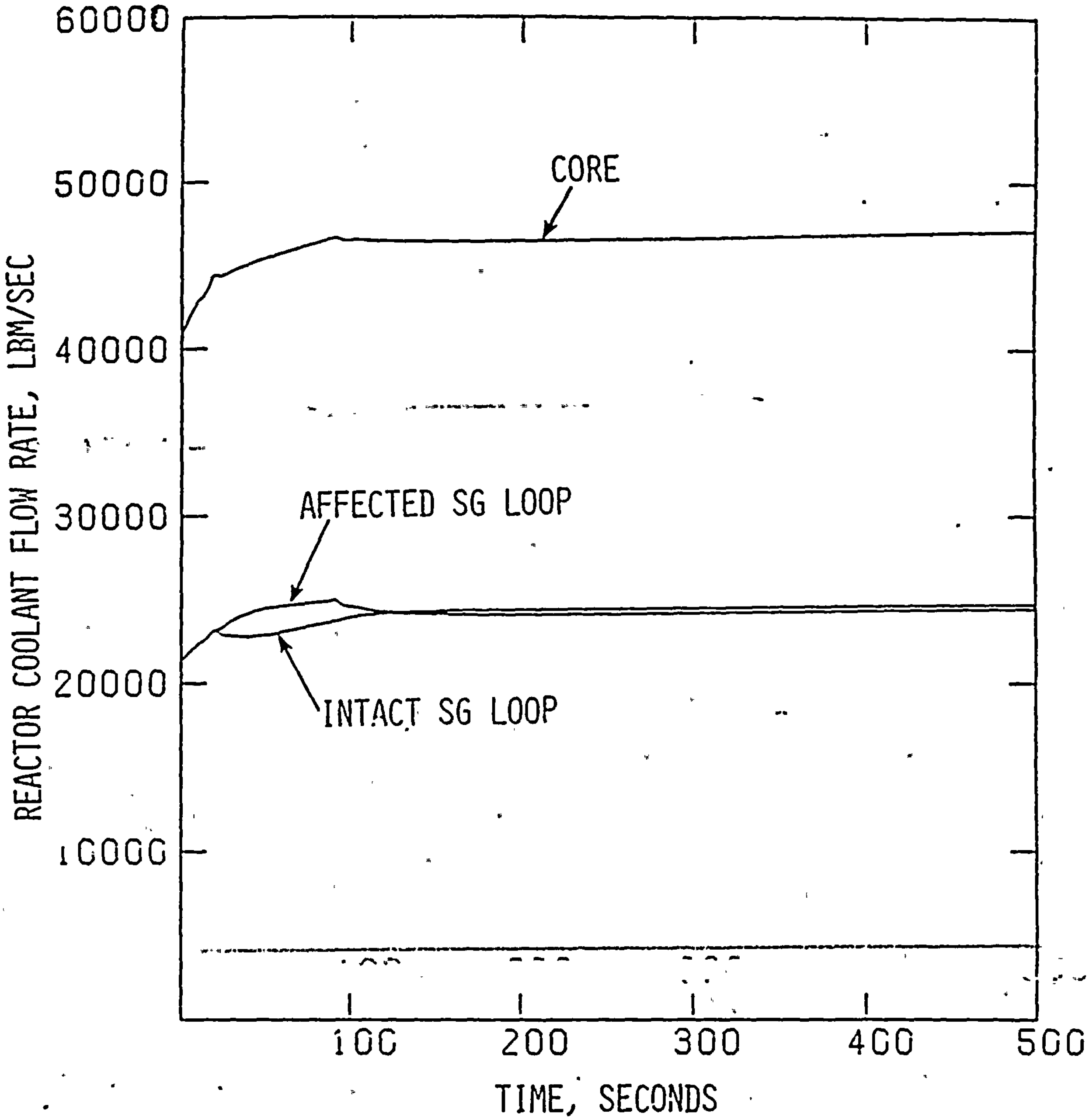
Palo Verde Nuclear Generating Station
FSAR

FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
CORE HEAT FLUX VS TIME

Figure 15.1.5-2



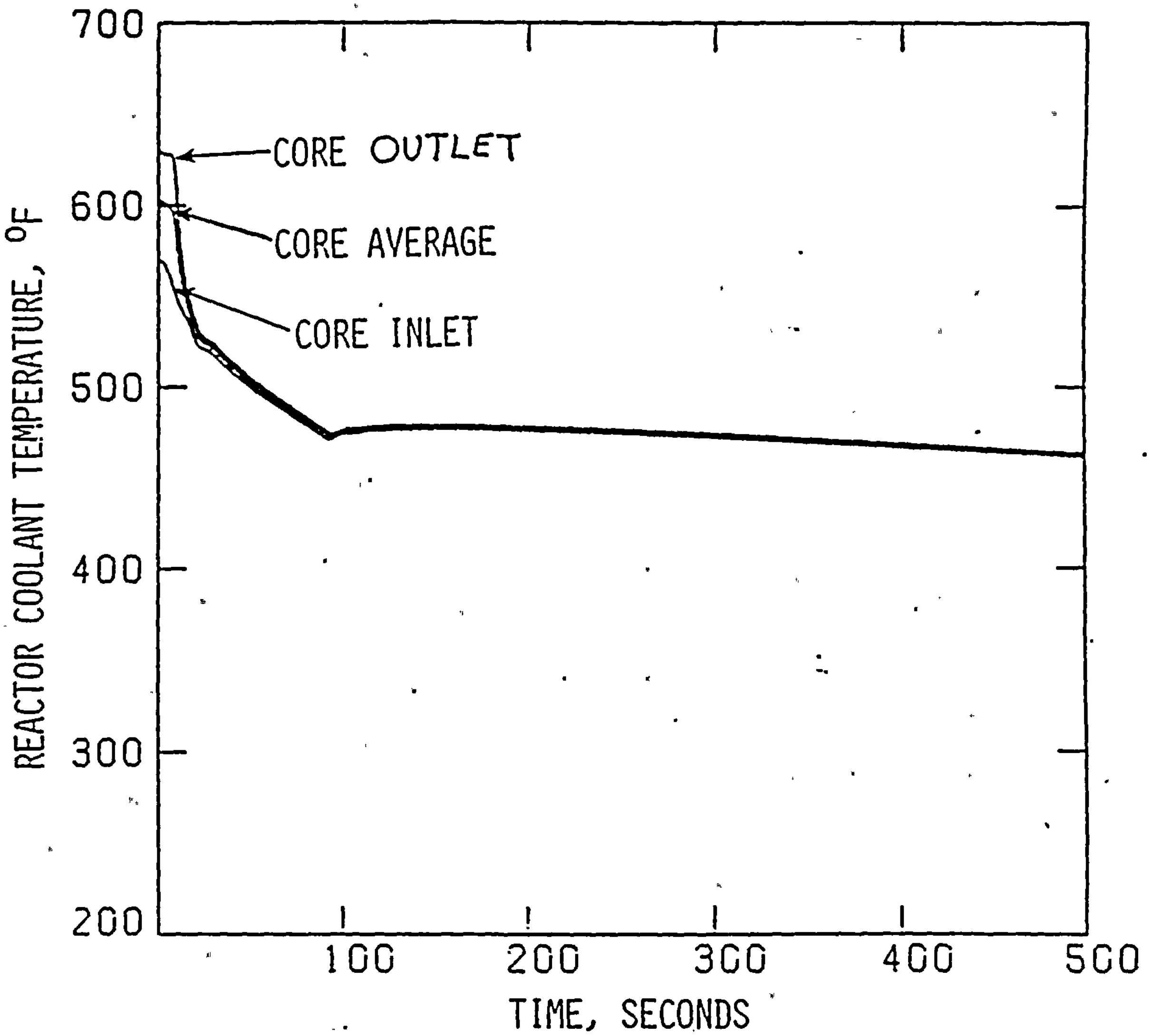
FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
RCS PRESSURE VS TIME
Figure 15.1.5-3



Palo Verde Nuclear Generating Station
FSAR

FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
REACTOR COOLANT FLOW RATE VS TIME

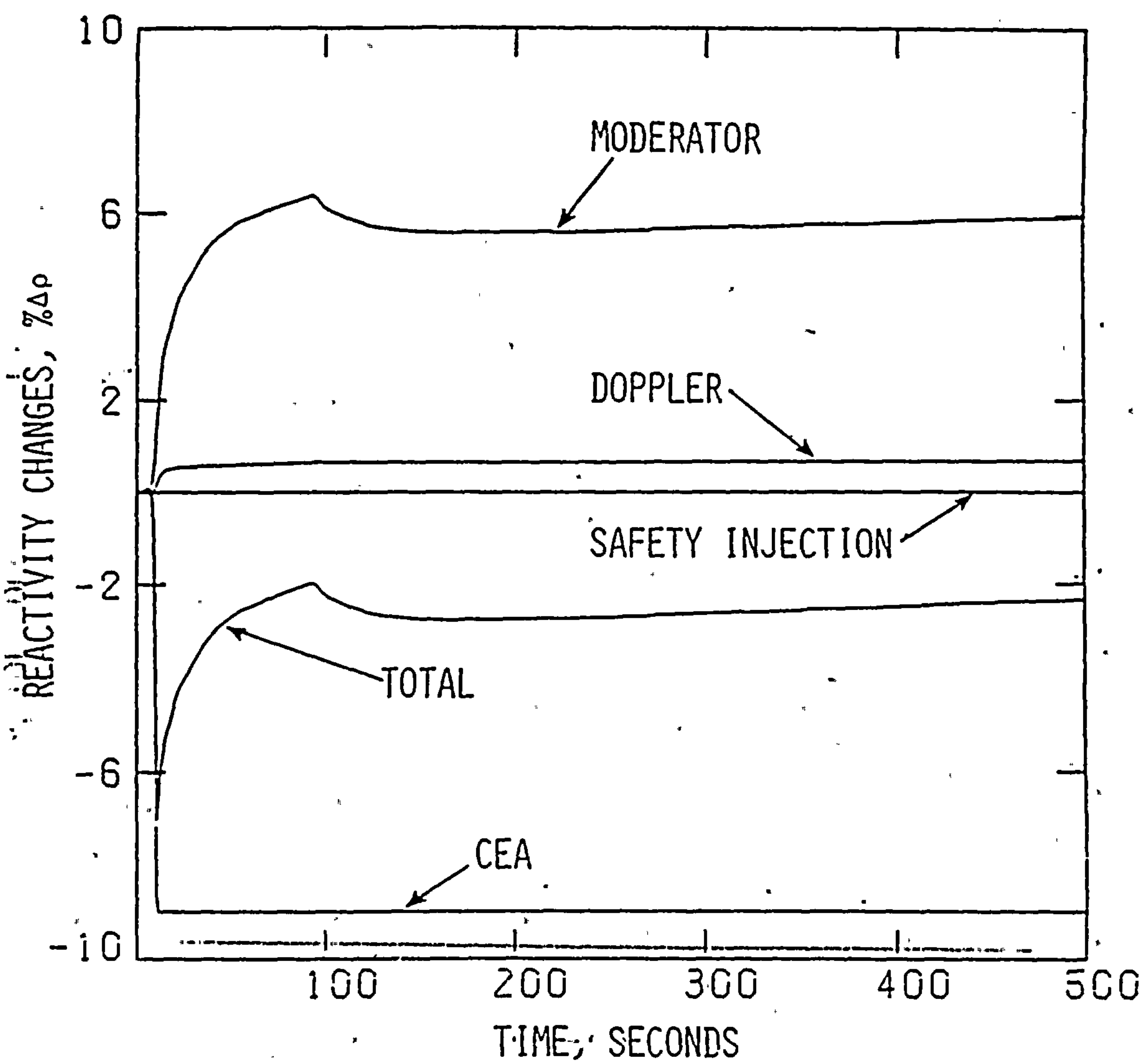
Figure 15.1.5-4



Palo Verde Nuclear Generating Station
FSAR

FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
REACTOR COOLANT TEMPERATURES VS TIME

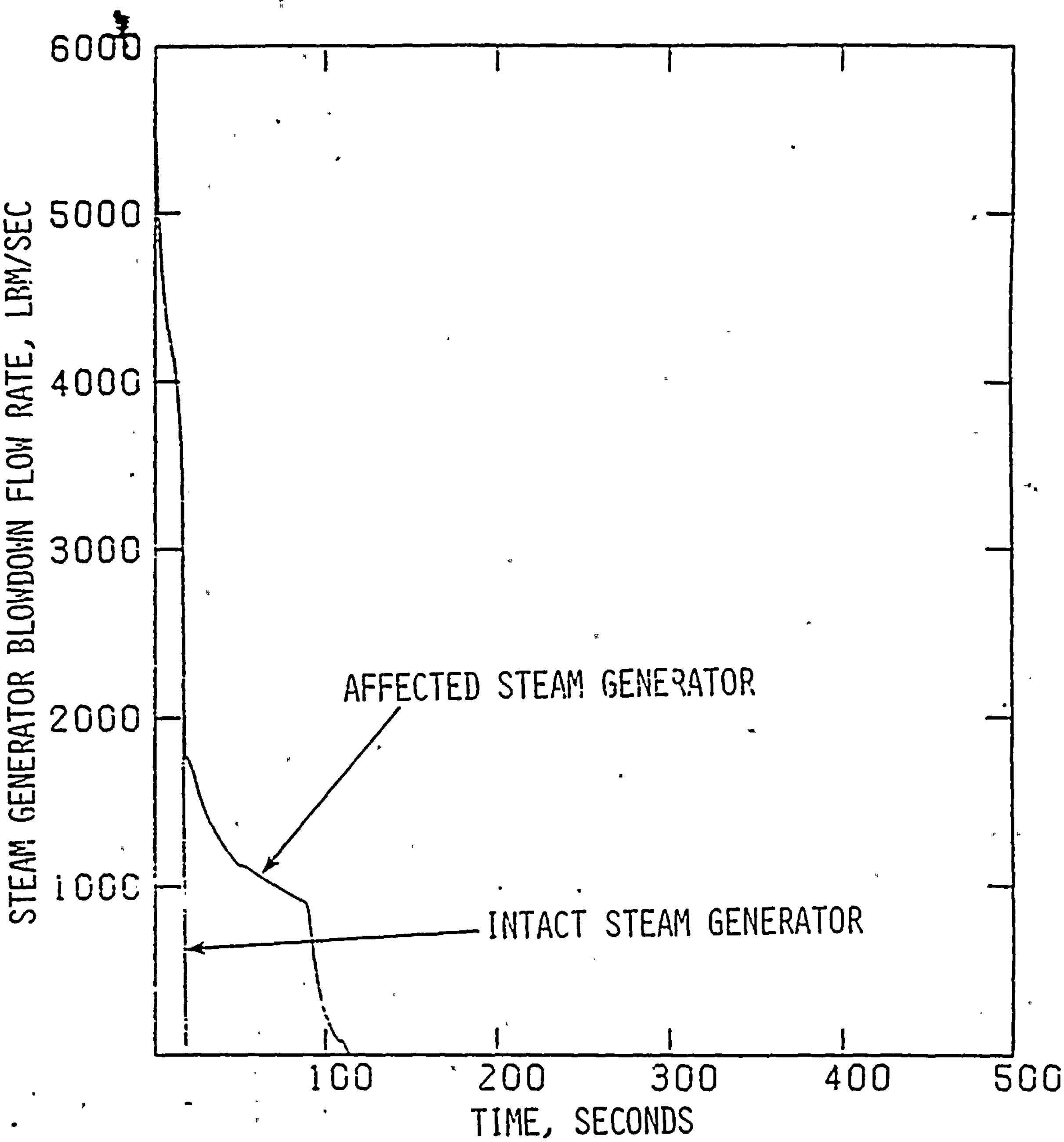
Figure 15.1.5-5



Palo Verde Nuclear Generating Station
FSAR

FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
REACTIVITY CHANGES VS TIME

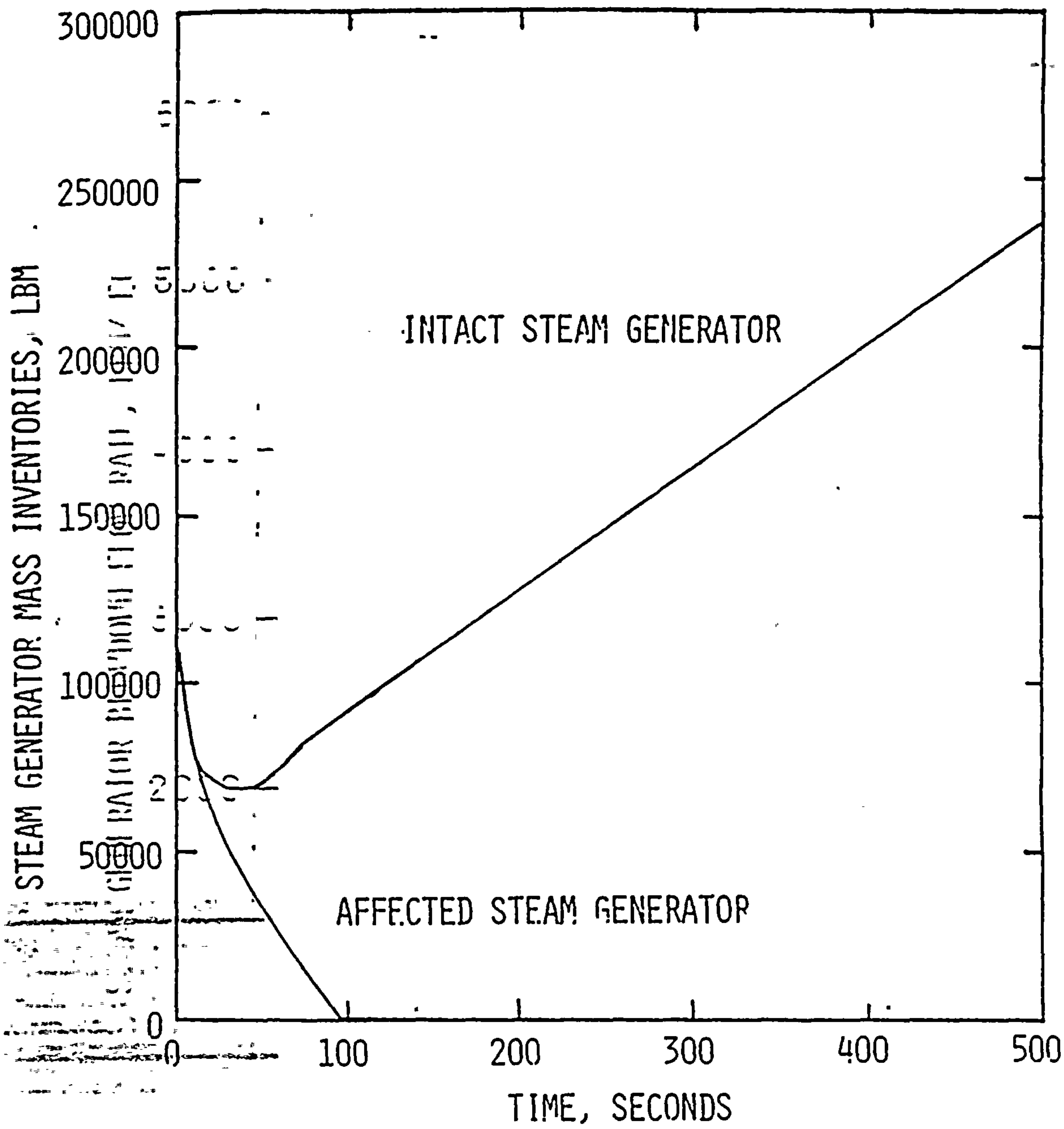
Figure 15.1.5-6



Palo Verde Nuclear Generating Station
FSAR

FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
STEAM GENERATOR BLOWDOWN RATES VS TIME

Figure 15.1.5-7



Palo Verde Nuclear Generating Station
FSAR

FULL POWER STEAM LINE BREAK WITH
AC POWER AVAILABLE
STEAM GENERATOR MASS INVENTORIES VS TIME
Figure 15.1.5-8

APPENDIX 15C

CONTENTS

APPENDIX 15C

ANALYSIS METHODS FOR LARGE STEAM LINE BREAKS

15C.1 INTRODUCTION

Refer to CESSAR Appendix 15C Section 15C.1.

15C.2 MATHEMATICAL MODELS

15C.2.1 PRIMARY AND SECONDARY SYSTEM THERMAL-HYDRAULIC MODEL

Refer to CESSAR Appendix 15C Section 15C.2.1 except that in addition to the five major model changes relative to versions of CESEC used for earlier FSAR analyses, the following two model changes are also applicable:

(6) an explicit model for the Palo Verde auxiliary feedwater system, and

(7) a boron injection and mixing model.

An explicit hydraulic model for the Palo Verde auxiliary feedwater system has been developed to account for variations in flow with steam generator pressure and pump speed. All major pumps, valves, and recirculation lines are explicitly modeled; elevation and frictional line loss effects are included. In addition, the CESEC auxiliary feedwater control model has been modified to account for the possibility of spurious, coincident actuation of flow to both steam generators. The impact of these changes (in conjunction with corresponding analysis assumptions presented in section 15C.3.3.3) is an auxiliary feedwater system model which accurately reflects the Palo Verde design while at the same time is conservative for steam line break assessments when compared to the assumed System 80 auxiliary feedwater system response.

The CESEC boron injection model (accounting for boron transport between the refueling water storage tank and the RCS cold legs) has been upgraded to account for two effects. First, the safety injection lines between the high and low pressure safety injection pumps and the cold legs have been nodalized to account for the possibility of a non-uniform initial boron concentration distribution in this region. The second change affects the mixed boron concentration in that portion of the RCS between the boron injection nozzle and the core following introduction of safety injection boron. Newly injected boron is now assumed to mix with only the downstream portion of the thermal hydraulic node into which it is injected. The fraction of the cold leg node which contributes to the resultant mixed boron concentration is conservatively calculated to be 0.28.

In addition, the mathematical model which deals with voiding in the RVUH region has been modified to provide a more realistic simulation of void behavior during times when the RCS pressure is increasing. The previous model provided an accurate simulation of void behavior during periods of decreasing RCS pressure. But, in the absence of a non-equilibrium model for the RVUH, the voids in the RVUH region collapse relatively rapidly when the RCS pressure begins to increase. More realistically, the voids would be expected to be compressed by the increasing pressure and dissipate slowly. The more rapid collapse of the voids results in a lower RCS pressure than would be expected from a non-equilibrium modeling of the RVUH. The modification that has been made to the mathematical model rectifies the problem by shifting the RVUH steam void volume to the pressurizer region when the RVUH voids begin to condense. The pressurizer volume is correspondingly increased. This modification results in a relatively higher RCS pressure, resulting in lower safety injection and boron flow into the RCS. The lower boron reactivity insertion is conservative with respect to return to power and minimum DNBR calculation.

15C.2.2 NUCLEAR MODEL

Refer to CESSAR Appendix 15C Section 15C.2.2.

15C.2.3 DNBR EVALUATION METHODOLOGY

Refer to CESSAR Appendix 15C Section 15C.2.3 except that for the return to power DNBR calculations a conservative three dimensional peaking factor of 100 is used to bound all possible power distributions.

15C.3 INPUT PARAMETERS AND INITIAL CONDITI[illegible]

15C.3.1 GENERAL

Refer to CESSAR Appendix 15C Section 15C.3.1 except that input parameters and initial conditions which maximize the potential for post-trip degradation in fuel performance are discussed in section 15C.3.3.

15C.3.2 PARAMETERS AND CONDITIONS FOR MAXIMIZING PRE-TRIP DEGRADATION IN FUEL PERFORMANCE

Refer to CESSAR Appendix 15C Section 15C.3.2.

15C.3.3 PARAMETERS AND CONDITION FOR MAXIMIZING POST-TRIP DEGRADATION IN FUEL PERFORMANCE

15C.3.3.1 Background

Refer to CESSAR Appendix 15C Section 15C.3.3.1.

15C.3.3.2 Plant Initial Conditions

Refer to CESSAR Appendix 15C Section 15C.3.3.2 except that for PVNGS the most adverse initial plant state for Return-to-Power (R-t-P) has been found to be the maximum core power, most positive ASI, minimum core flowrate, maximum pressurizer water level, maximum core inlet coolant temperature, maximum reactor coolant system pressure, and maximum water level in both steam generators. The departure from the System 80 asymmetric initial water

level conditions more accurately reflects actual, automatically controlled steady state steam generator water levels. As is discussed in section 15C.3.3.3, auxiliary feedwater actuation is conservatively assumed to occur despite steam generator water levels which very likely remain above the actuation setpoint.

15C.3.3.3 Analysis Assumptions

Refer to CESSAR Appendix 15C Section 15C.3.3.3 except that:

- CESSAR Table 15C-1 is not applicable for PVNGS.

- If actuated during a steam line break, auxiliary feedwater will contribute to the cooldown of the RCS and may result in higher R-t-P through the negative moderator reactivity term. Protection against such an occurrence is provided by a high steam generator differential pressure interlock which isolates auxiliary feedwater to a generator identified as ruptured. For the Palo Verde steam line break analysis, however, auxiliary feedwater is assumed to be actuated simultaneously to both steam generators before the lockout condition is reached, due to the turbulent behavior of the steam generator water levels during blowdown. Actuation is conservatively assumed to occur at the time of main steam isolation valve closure following the main steam isolation signal, at which time water level drops abruptly due to a reduction in steaming rate.

INSERT Z

- ~~The safety injection line sweep out volume, summed over all four injection points, is assumed to be 148 ft³ to more accurately reflect the most adverse condition anticipated for Unit 1 Cycle 1. This more limiting assumption reduces the impact of both the modified cold leg boron mixing model discussed in section 15C.2.1 and the negative safety injection reactivity itself.~~

- End of ~~first cycle~~ equilibrium (SCR) burnup core conditions to yield the most negative moderator coefficient were used in the PVNGS SLB analysis.

15C.3.3.4 Single Failures

Refer to CESSAR Appendix 15C Section 15C.3.3.4 except that:

- CESSAR Table 15C-2 is not applicable for PVNGS.

- The differences between the Palo Verde specific design and the generic System 80 design have a potential impact on the most adverse single failure. ~~First, the volume of water to be swept out of the safety injection line before boron reaches the cold leg is 148 ft³ for Palo Verde (section 15.3.3.3) and 120 ft³ for System 80. Thus, the relative contribution of safety injection boron with both operating is smaller for Palo Verde, as is the detrimental effect of failing one pump. Second,~~ The total steam flow path downstream of the main steam

INSERT "Z"

The volume of water that is swept out of the safety injection lines before boron reaches the RCS is a maximum of 60.6 cubic feet summed over all four cold leg safety injection lines, to more accurately reflect the PVNGS configuration.

isolation valves which is not automatically isolated is: 0.034 ft^2 for Palo Verde versus 0.256 ft^2 for System 80. Therefore, the impact of a single main steam isolation valve failure or the RCS cooldown is less adverse for Palo Verde than for System 80. An assessment of these two single failures reveals that the failure of a high pressure safety injection pump produces the most adverse transient results.

15C.3.4 PARAMETERS AND CONDITIONS FOR MAXIMIZING SECONDARY SYSTEM CONTRIBUTION TO RADIOLOGICAL RELEASES

Refer to CESSAR Section 15C.3.4.

15C.4 REFERENCES

Refer to CESSAR Appendix 15C for references.