NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO: 3653

FILE:

FROM:	DATE OF DOC	DATE REC'D	LTR	TWX	RPT	OTHER
Florida Power & Light Co. Miami, Fla Robert E. Uhrig	3-31-75	4-4-75	xxx			

TO:	ORIG	CC	OTHER	
Mr. Benard C. Rusche	3-signed			SENT AEC PDR xxx SENT LOCAL PDR xxx

CLASS	UNCLASS	PROP INFO	INPUT	NO CYS REC'D	DOCKET NO:
	xxxxx			3-cys	50-335

DESCRIPTION:

Ltr ref WASH-1270 advising that CENPD-158 is the applicable analysis to St. Lucie Unit 1 . . . trans the following:.....

ENCLOSURES:

Conclusions drawn from CENPD-158/149 and WASH-1270 as relating to St. Lucie Unit 1
......

PLANT NAME: St, Lucie #1

4-7-75 JGB

FOR ACTION/INFORMATION

BUTLER (L) W/ Copies	SCHWENCER (L) W/ Copies	ZIEMANN (L) W/ Copies	REGAN (E) W/ Copies
CLARK (L) W/ Copies	STOLZ (L) W/ Copies	DICKER (E) W/ Copies	LEAR (L) W/ Copies
PARR (L) W/ 4 Copies	VASSALLO (L) W/ Copies	KNIGHTON (E) W/ Copies	SPELS W/ Copies
KNIEL (L) W/ Copies	PURPLE (L) W/ Copies	YOUNGBLOOD (E) W/ Copies	W/ Copies

INTERNAL DISTRIBUTION

REG FILE
~~NRC PDR~~
OGC, ROOM P-506A
GOSSICK/STAFF
CASE
GIAMBUSSO
BOYD
MOORE (L)
DEYOUNG (L)
SKOVHOLT (L)
GOLLER (L) (Ltr)
P. COLLINS
DENISE
REG OPR
FILE & REGION (2)
T.R. WILSON
STEELE

TECH REVIEW
SCHROEDER
MACCARY
KNIGHT
PAWLICKI
SHAO
STELLO
HOUSTON
NOVAK
ROSS
IPPOLITO
TEDESCO
LONG
LAINAS
BENAROYA
VOLLMER
Thadani

DENTON
GRIMES
GAMMILL
KASTNER
BALLARD
SPANGLER

ENVIRO
MULLER
DICKER
KNIGHTON
YOUNGBLOOD
REGAN
PROJECT LDR

HARLESS

LIC ASST
R. DIGGS (L)
H. GEARIN (L)
E. GOULBOURNE (L)
P. KREUTZER (E)
J. LEE (L)
M. MAIGRET (L)
S. REED (E)
M. SERVICE (L)
S. SHEPPARD (L)
M. SLATER (E)
H. SMITH (L)
S. TEETS (L)
G. WILLIAMS (E)
V. WILSON (L)
R. INGRAM (L)

A/T IND.
BRAITMAN
SALTZMAN
MELTZ

PLANS
MCDONALD
CHAPMAN
DUBE (Ltr)
E. COUPE
PETERSON
HARTFIELD (2)
KLECKER
EISENHUT
WIGGINTON
VARGA

EXTERNAL DISTRIBUTION

1 – LOCAL PDR Ft. Pierce, Fla
1 – TIC (ABERNATHY)
1 – NSIC (BUCHANAN)
1 – ASLB
1 – Newton Anderson
14 – ACRS ~~HOLDING~~/SENT
to Lic Asst.

(1)(2)(10) – NATIONAL LABS
1 – W. PENNINGTON, Rm E-201 GT
1 – CONSULTANTS
NEWMARK/BLUME/AGBABIAN

(2)
1 – PDR-SAN/LA/NY
1 – BROOKHAVEN NAT LAB
1 – G. ULRIKSON, ORNL
1 – AGMED (RUTH GUSSMAN) Rm B-127 GT
1 – J. D. RUNKLES, Rm E-201 GT

P. O. BOX 3100, MIAMI, FLORIDA 33101
FLORIDA POWER & LIGHT COMPANY

March 31, 1975
L-75-161

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Rusche:

Re: St. Lucie Unit 1
Docket No. 50-335
ATWT Analysis

The AEC technical report WASH-1270 requests that all plants falling under Section IB, Category 2, of the Regulatory Staff ATWT position submit analyses of ATWT events and any modifications that are necessary to mitigate the consequences of ATWT events. Florida Power & Light Company, in compliance with this request, wishes to advise you that we are referencing the Combustion Engineering topical report, CENPD-158, "Anticipated Transients Without Reactor Trip", December, 1974, as the analysis applicable to St. Lucie Unit 1.

The analyses and parameters listed in CENPD-158 are applicable to St. Lucie 1 with the following qualifications:

1. The auxiliary feedwater system for St. Lucie 1 is actuated by operator action in response to an audible alarm rather than automatically, as assumed in CENPD-158. Therefore, a delay time of 600 seconds for manual actuation of the auxiliary feedwater should be considered for those transients where the main feedwater shuts off, compared to the 30 seconds delay time assumed in CENPD-158.

2. The total pressurizer relieving area for St. Lucie 1 is 0.061 ft^2 compared to the 0.054 ft^2 assumed in CENPD-158.

3. A value for the most positive moderator temperature coefficient of -0.2×10^{-4} $\Delta\rho$/F at BOL is more applicable to St. Lucie 1 than the value of -0.6×10^{-4} $\Delta\rho$/F at BOL assumed in CENPD-158 for the same operating conditions.

3653

4. Minor differences in reactor inlet temperature and steam generator pressure are noted for St. Lucie 1.

Table 1 compares the initial conditions used in CENPD-158 with the applicable values for St. Lucie 1.

In CENPD-158, ten separate transients were analyzed. Section 1.2 of CENPD-158 lists the generalized criteria given in Appendix A of WASH-1270 for assessing the results of ATWT events. The results reported in Table 3.1 of CENPD-158 show that only during three of the transients, Loss of External Load, Loss of Normal Feedwater and Loss of Normal On-Site and Off-Site Electrical Power (Station Blackout) are the limits of the criteria approached. CENPD-158 also shows that the only limit of concern is that of the Reactor Coolant System (RCS) pressure.

Of the three limiting transients, the Loss of Normal On-Site and Off-Site Electrical Power can be eliminated because it has the lowest primary system pressure (CENPD-158) of the three transients and is considered to be of a lower probability than 10^{-7} per year. The reason is that for this transient, there are two types of common mode failure that have been discussed by the Regulatory Staff. The first is a common mode failure in the Reactor Protective System (RPS); however, whatever this failure may be, power is always lost to the control element drive mechanisms (CEDM's) and the control element assemblies (CEA's) will fall by gravity into the core. The results of this transient are shown in the SAR (i.e., pressure less than 2500 psia and a minimum DNBR greater than 1.3). The second common mode failure suggested by the Staff is that in which the CEDM's suffer a mechanical common mode failure and no CEA's fall into the core when power is removed from the holding coils. This type of mechanical common mode failure has a very low probability (because of the magnetic jack design and the option of frequent exercising of rods during plant operation) and, therefore, could meet the unreliability goal set by the Staff in WASH-1270.

As for a complete Loss of Feedwater, the basic design of the Condensate and Feedwater System (CFWS) does not make such a transient an anticipated event. The CFWS is comprised of two interconnected trains, each consisting of condensate pumps, heater drain pumps, main feedwater pumps, and feedwater heaters. With the normal anticipated single failure of any of these active components, the system still supplies 50% of the design feedwater flow. Therefore, only a partial Loss of Feedwater is an anticipated transient.

A partial Loss of Feedwater results in a peak reactor coolant system pressure of less than 2750 psia for St. Lucie 1 (the difference in peak system pressure between manual and automatic auxiliary feedwater initiation is small for this transient).

Thus, a Loss of External Load is the most limiting ATWT event for the St. Lucie plant. An analysis of this transient utilizing St. Lucie 1 parameters has been performed by the reactor vendor (Combustion Engineering) and is presented in the Appendix. The analysis shows that during this transient the reactor coolant system pressure peaks at 3393 psia.

According to WASH-1270, the maximum primary stress anywhere in the reactor coolant system boundary has to be less than that of the emergency conditions as defined in the ASME Nuclear Power Plant Components Code, Section III. The investigation of the primary system pressure limit for St. Lucie Unit 1 based on these criteria has not been completed by the reactor vendor. While there are scores of valves at the primary system pressure boundary whose pressure limit has yet to be determined, pressure limits for the major primary system components such as the reactor coolant pumps, piping and pressurizer are above 3900 psia based on "as built" dimensions of the components and on stress intensity limits derived from QC records of mill test data. It is expected that the pressure limit evaluation will show that peak primary coolant pressure is below the emergency limit of the primary system and that therefore, no design modifications are required. If further investigations show that primary pressures following an ATWT are higher than acceptable limits they could be reduced by the installation of additional pressurizer safety valves to increase pressurizer relieving area. The effects of this modification on the consequences of the most severe ATWT have been addressed in the Loss of Load analysis for St. Lucie Unit 1 presented in the Appendix.

In addition to the analyses to demonstrate the consequences of ATWT, WASH-1270 requests that a review of the reactor shutdown system be performed to assess the system susceptibility to common mode failure. These analyses have been performed by Combustion Engineering and are documented in CENPD-149, "Review of Reactor Shutdown System (RPS Design) for Common Mode Failure Susceptibility", November, 1974.

Based on the criteria for analyzing ATWT events as specified in WASH-1270, the analyses in CENPD-158 and CENPD-149, and the analysis included in the Appendix, the following conclusions are drawn for St. Lucie Unit 1:

1. the calculated radiological consequences are within the guidelines set forth in 10 CFR 100;

2. the reactor fuel rods are able to withstand the internal and external transient pressure as to maintain a long-term coolable geometry and there will be no significant fuel melting during these transients,

3. the calculated maximum containment pressure does not exceed the design pressure of the containment structure,

4. the fuel cladding does not experience DNB nor,

5. does it appear that the calculated reactor coolant system transient pressure exceeds the emergency stress limits of the primary system.

Yours very truly,

Robert E. Uhrig
Vice President

REU:nch
Attachment

cc: Mr. Jack R. Newman

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

TABLE 1

Parameter	CENPD-158	St. Lucie Unit 1
Moderator Temperature Coefficient	BOC-0.6x$10^{-4}\Delta\rho$/F	BOC-0.2x$10^{-4}\Delta\rho$/F
Doppler Coefficient	Figure 1.5-1 CENPD-158	Figure 4.3-21A St. Lucie 1 FSAR
Axial Power Distribution	Ip* = -.08 Peak = 1.40	Ip* = -.08 Peak = 1.40
Radial Peaking Factor	1.45	1.45
Core Power	2560 MW_t	2560 MW_t
Inlet Temperature	546 F	542 F
Reactor Vessel Flow	381,100 gpm	381,100 gpm
Decay Heat Function	ANS Decay Heat Standard	ANS Decay Heat Standard
Pressurizer Water Volume	769 ft^3 (total pressurizer volume = 1500 ft^3)	769 ft^3 (total pressurizer volume = 1500 ft^3)
Pressurizer Pressure	2250 psia	2250 psia
Pressurizing Relieving Area	0.054 ft^2	0.061 ft^2
Steam Generator Pressure	850 psia	815 psia
Steam Generator Mass	137,800 lbs	137,800 lbs
Feedwater Enthalpy	415 Btu/lb	415 Btu/lb
Auxiliary Feedwater Initiation	30 sec.	600 sec.
Engineering Factor on Enthalpy rise	1.0	1.0
Azimuthal Tilt Factor	1.0	1.0
Pitch and Bowing Factor	1.0	1.0
Inlet Plenum Flow Mal-distribution Factor	1.0	1.0

* Peripheral Axial Shape Index (Ip) $= \dfrac{\text{Power in lower core half - Power in upper core half}}{\text{Total Core Power}}$

all measured at the core periphery.

APPENDIX

Loss of Load and/or Turbine Trip
ATWT Analysis for St. Lucie Unit 1

LOSS OF EXTERNAL LOAD AND/OR TURBINE TRIP

1. Introduction

1.1 Identification of Causes

Loss of External Load and/or Turbine Trip results in a reduction of steam flow from the steam generators to the turbine generator. Cessation of steam flow to the turbine generator occurs because of closure of the turbine stop valves or turbine control valves. The cause of loss of load may be the result of abnormal events in the electrical distribution network or turbine trip. In either of these situations, offsite power is available to provide ac power to the auxiliaries.

A complete loss of load results in a partial loss of secondary heat removal capability causing a rise in the nuclear steam supply system (NSSS) pressure and temperatures. The heat capacity of the water in the reactor coolant system and in the steam generators, the steam and water discharge capacity of the pressurizer relief and safety valves and steam generator safety valves, and the action of the feedwater system all provide a heat removal capacity sufficient to insure acceptable consequences.

1.2 Loss of Load Protection

The Reactor Protective System (RPS) would rapidly terminate this incident by any one of the redundant and diverse trip signals given in Table 1. In addition to the above, pretrip alarms associated with each automatic trip, as well as the high pressurizer water level alarm, would provide audible and visual indications to the operator during the course of the incident.

A common mode failure that would prevent the CEA's from inserting during this transient is not considered credible. Nevertheless, this incident was analyzed assuming that no CEA's will be inserted into the core upon any trip signal.

2. Analysis of Effects and Consequences

2.1 Method of Analysis

The analysis of a complete loss of load incident was performed with the CESEC computer program, described in Section 1.3.1 of CENPD-158.

The parameters used in the analysis are listed in Table 2.

Parametric analyses were conducted to determine the sensitivity of peak transient pressure to pressurizer relieving area (0.061 to 0.100 ft^2).

2.2 Initial Conditions and Assumptions

The analyses are based on NSSS parameters for St. Lucie Unit 1. Unless indicated otherwise, it is assumed that:

1. Emergency feedwater is actuated 600 seconds after the turbine trip by manual initiation of the operator.

2. Power operated relief valves function on demand.

3. A conservative value of 10 minutes delay, following initiation of the transients, is assumed prior to manual actuation of the Safety Injection Actuation Signal (SIAS). Actually, an automatic SIAS will be initiated when the containment pressure rises above 5 psig or when the RCS pressure drops below 1600 psia.

4. When a turbine trip occurs the normal feedwater flow ramps down from full flow to 5% of full flow within 20 seconds.

Guidance in selecting initial conditions was taken from Reference 2 of CENPD-158. The parameters addressed in this section are taken from the list presented in the above mentioned document and the justification for the value of each parameter is based on the method of justification of conditions presented in Section 2.2.2 of that document. Section 2.2.2 of Reference 2 of CENPD-158 presents four different possible methods of justification for the selection of initial conditions. These methods are:

"a) Selection of a conservative value defined by either the design basis FSAR analysis, or the Technical Specification limit.

b) For variables regulated by either automatic control systems or manually under administrative control, selection of the design

operational value including allowance for control band, but excluding any allowance for measurement uncertainty.

c) Selection of either the measured or design value excluding any allowance for design margin or measurement uncertainty.

d) Selection of a calculated value which is not expected to be exceeded (i.e., more adverse) during the preponderance (at least 95%) of plant lifetime. Justifications of this probability argument need not consider allowance for calculational uncertainty or for random statistical fluctuations."

Each of these four methods has been employed in this study because some parameters are design parameters; whereas other parameters are calculated variables and still other parameters are under administrative or automatic control. In each case the method of justification is based on the most appropriate method for that particular parameter. Table 2 presents the values, bases, and justifications for the initial conditions used in this report supplementing CENPD-158. The beginning of cycle conditions were used and the transient was initiated from full power; four-pump operation.

2.3 Results

The RPS and NSSS responses are delineated in the sequences of events listed in Table 3. The core power, core average heat flux, reactor coolant system pressure, reactor coolant system temperatures, steam generator pressures, and reactivities, all as a function of time are presented in Figures 1 to 6.

The Loss of Load incident is initiated from full power BOL conditions by a turbine trip caused by loss of condenser vacuum and, therefore, the dump and bypass to condenser is not available. In addition to the general assumptions discussed in Table 2 of this report, credit is taken for the SIAS initiated by the operator at 600 seconds. The feedwater system is assumed to be in the automatic mode.

On turbine trip, it is assumed that the turbine stop and control valves close instantaneously stopping all steam flow to the turbine. This would result in initiation of both a loss of load and low steam generator water level reactor trip signal. The feedwater controller ramps back main feedwater to 5% of nominal by 20 seconds. The temperature and pressure of both the reactor coolant system and steam generators begin to increase immediately. By 6.6 seconds, the steam generator pressure reaches 1000 psia and the steam generator safety valves begin to open. The steam flow rate rapidly exceeds the feedwater flow rate and the steam generator secondary inventory begins to decrease. By

8.3 seconds, the RCS pressure reaches 2400 psia and the pressurizer relief valves open limiting the RCS pressure increase. At this point, a High Primary Pressure Trip would have occurred. At 47 seconds, the RCS pressure drops below 2400 psia due to the action of the primary relief valves, secondary safety valves, letdown flow, and the decrease in core power level. During this time the reactor power has decreased to approximately 88% of full power due to the negative moderator temperature coefficient. Steam generator pressure reaches a maximum value of 1068 psia at 63 seconds and remains at this pressure until 89 seconds when the decreasing steam generator liquid inventories begin to result in a reduction in the primary-to-secondary heat transfer area. As the primary-to-secondary heat transfer is reduced, the steam generator pressure and steam flow rate decrease. The resulting primary-to-secondary power mismatch causes an increase in RCS pressure such that by 97 seconds the pressurizer pressure again reaches 2400 psia and the pressurizer relief valves open. By 105 seconds, the pressurizer pressure reaches 2500 psia and the pressurizer safety valves open. At 111.5 seconds, the pressurizer fills, the pressurizer relief and safety valves begin relieving water and the RCS pressure begins to increase rapidly. The reactor power has decreased to 75% at this time.

The reactor coolant system reaches a maximum pressure of 3393 psia by 163.6 seconds, while the reactor power has decreased to 35% of nominal due to the effect of decreasing moderator density on reactivity feedback.

The maximum reactor coolant system temperature of 697°F is reached at about 250 seconds, at which time the core power has decreased to 6%. By 251.3 seconds, the RCS pressure has decayed to the point where the main coolant pumps trip due to cavitation. The resulting flow reduction results in the low coolant flow reactor trip setpoint being reached at 252.7 seconds. At this time, the primary-to-secondary heat transfer is insufficient to completely boil the reduced flow (5%) of normal feedwater and the steam generator liquid inventories begin to increase. As the pressure decay reaches the saturation temperature at 260 seconds, steam forms in the reactor vessel head, the hottest region in the reactor coolant system. After 282 seconds, the pressurizer empties while core power has decreased to 4%.

The operator manually initiates an SIAS and auxiliary feedwater at 600.0 seconds and the boron reaching the core at 745 seconds insures a shutdown condition and a controlled cooldown can be initiated.

Figure 7 demonstrates the sensitivity of the maximum RCS pressure to changes in the pressurizer relieving area.

3. Conclusions

The results of the Loss of Load incident without reactor trip show that the NSSS will experience a pressure in excess of the hydrostatic test pressure (3125 psia). The peak RCS pressure for manual initiation of auxiliary feedwater is 3393 psia.

The minimum DNBR in the hot channel throughout the transient is greater than 3.5, thus precluding any fuel damage.

Boric acid injection from the SIAS returns the reactor to a stable shutdown condition.

The results of the parametric analyses show that the RCS pressure, the most limiting condition for this transient, is significantly affected by the total pressurizer relieving area. The addition of a fourth safety valve (area = 0.013 ft^2) reduces the peak RCS pressure to 3040 psia.

TABLE 1

Reactor Protective System Trip Signals Applicable
To the Loss of External Load Incident

High Pressurizer Pressure Trip

Low Reactor Coolant Flow Trip

Low Steam Generator Water Level Trip

Loss of Load Reactor Trip

High Containment Pressure Trip

TABLE 2

Initial Condition for St. Lucie 1 ATWS Analyses

Parameter	Value	Basis	ANS-N661 Classification
Moderator Temperature	BOC=$0.2 \times 10^{-4} \Delta\rho/°F$	Least negative calculated value for St. Lucie 1 (without uncertainties)	d
Doppler Coefficient	Figure 4.3-21A St. Lucie 1 FSAR	Calculated value for beginning of cycle (without uncertainties)	d
Core Power	2560 Mwt	St. Lucie 1 design power (without uncertainties)	c
Inlet Temperature	542°F	St. Lucie 1 design power (without uncertainties)	c
Reactor Vessel Flow	381,100 gpm	St. Lucie 1 minimum calculated value (without uncertainties)	d
Core Mass Flow Rate	137.6×10^6 lbm/hr	St. Lucie 1 minimum calculated value (without uncertainties)	d
Decay Heat Function	ANS Decay Heat Standard	Reference 15 of CENPD-158	d
Pressurizer Water Volume	769 $ft.^3$ (total pressure volume = 1500 $ft.^3$	St. Lucie 1 design pressurizer water level program (without uncertainties)	c
Pressurizer Pressure	2250 psia	St. Lucie 1 design pressurizer pressure (without uncertainties)	c
Steam Generator Pressure	815 psia	St. Lucie 1 design steam generator pressure (without uncertainties)	c
Steam Generator Mass	137,800 lbs.	St. Lucie 1 design steam generator mass (without uncertainties)	c

TABLE 2 (Cont'd)

Parameter	Value	Basis	ANS-N661 Classification
Feedwater Enthalpy	415 Btu/lb	St. Lucie 1 design feedwater enthalpy (without uncertainties)	c
Pressurizer Relieving Area	.061 ft.^{2}	2 relief valves at .011 ft.^{2} each plus 3 safety valves at .013 ft.^{2} each	d
Axial Power Distribution	Ip* = -.08 peak = 1.40	Most adverse axial power distribution within control band (without uncertainties)	b
Radial Peaking Factor	1.45	Maximum calculated value for full power conditions (without uncertainties)	d
Engineering Factor on Enthalpy rise	1.0	Value consistent with nominal conditions	d
Azimuthal Tilt Factor	1.0	Value consistent with nominal conditions	d
Pitch and Bowing Factor	1.0	Value consistent with nominal conditions	d
Inlet Plenum Flow Maldistribution Factor	1.0	Value consistent with nominal conditions	d

*Peripheral Axial Shape Index (Ip) = $\frac{\text{Power in lower core half-Power in upper half}}{\text{Total core half}}$

all measured at the core pheriphery

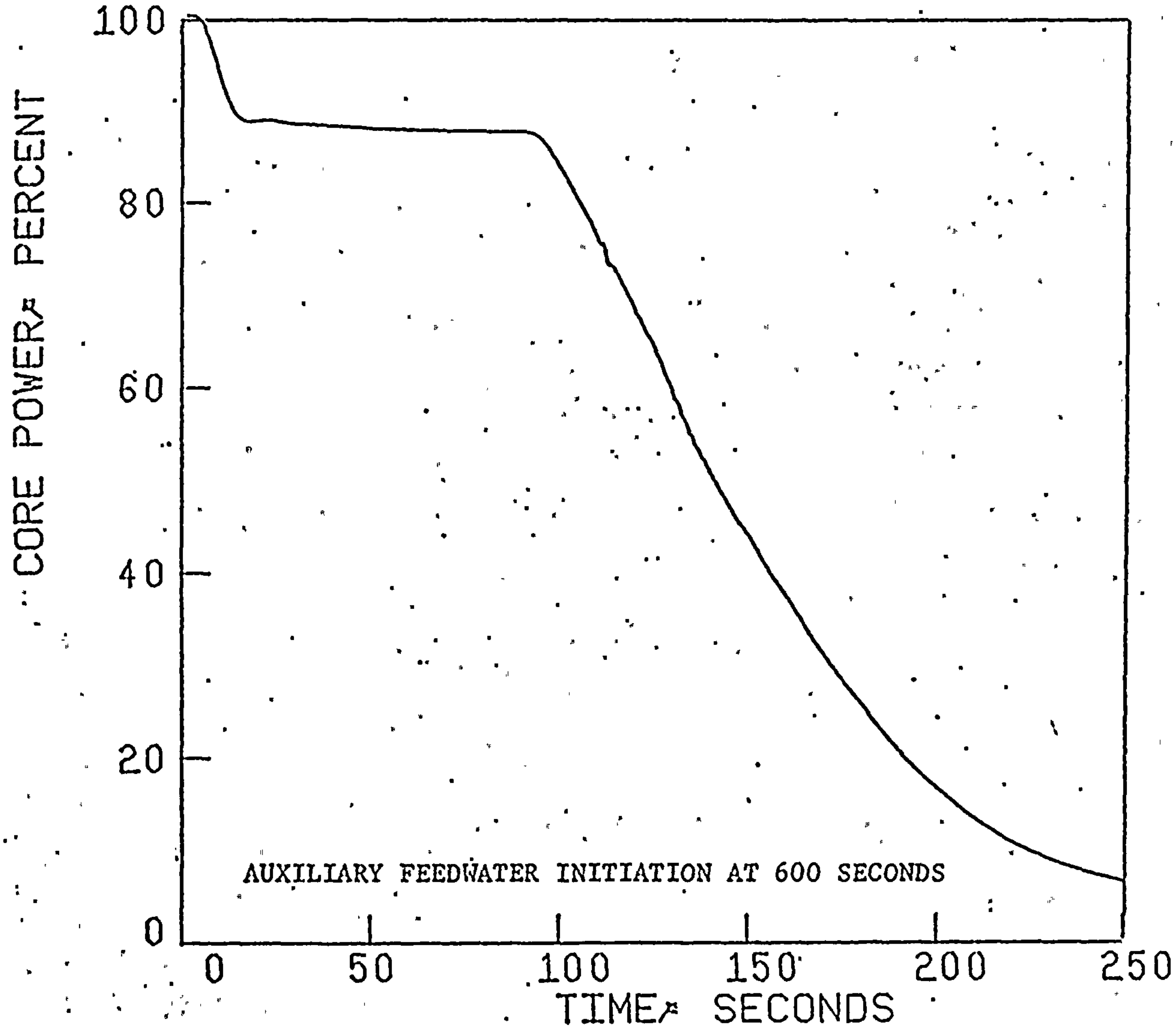
TABLE 3

Sequence of Events for the Loss of External Load
St. Lucie Unit 1 - Manual Auxiliary Feedwater Initiation

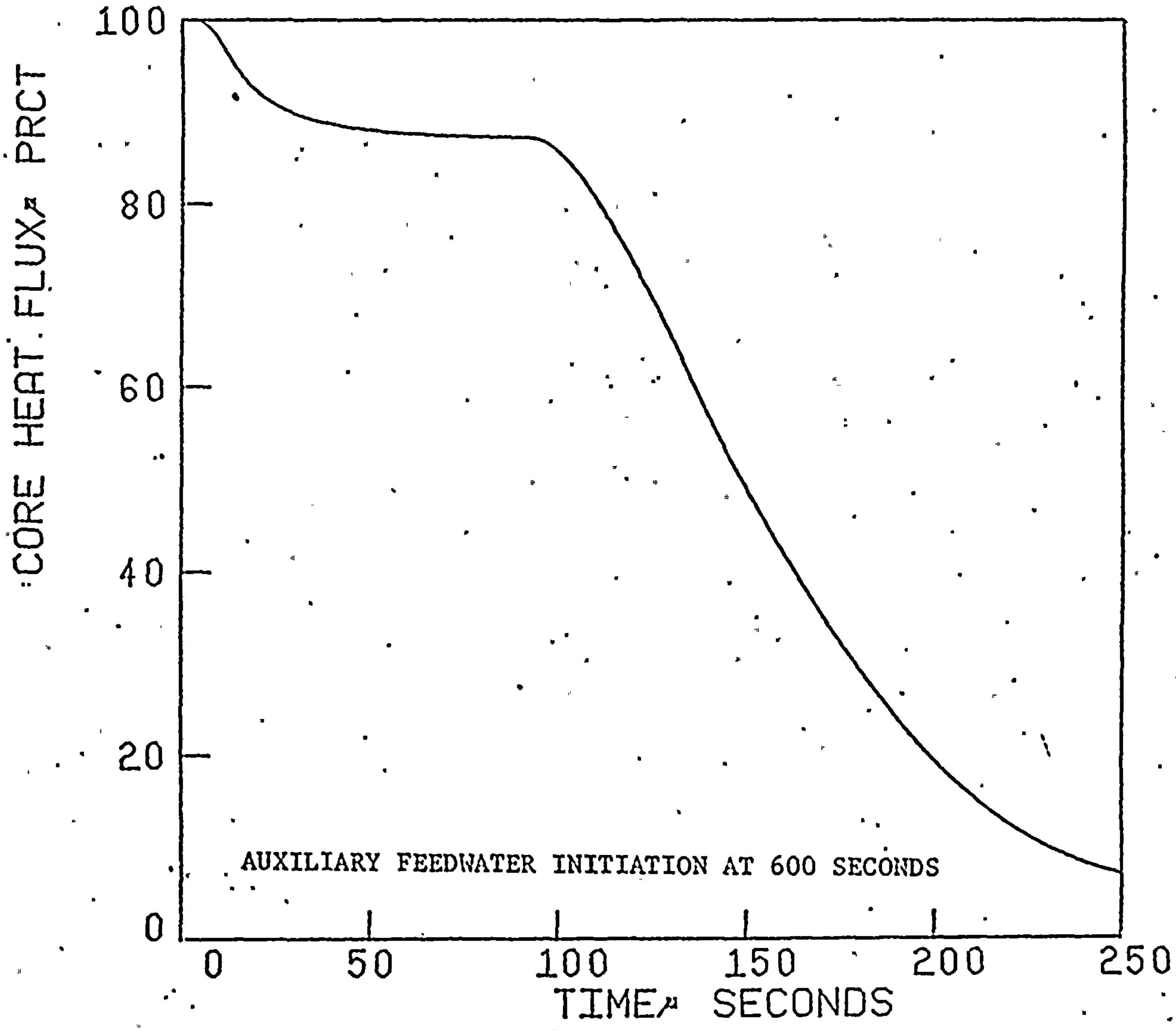
Time	Event	Setpoint or Value
0.0	Turbine stop valves close instantaneously	---
0.3	Low steam generator level alarm setpoint reached	2.77 ft. below programmed level
1.6	Low steam generator level trip setpoint reached	4.27 ft. below programmed level
1.8	High coolant average temperature alarm setpoint reached	567°F
6.6	High pressurizer pressure alarm setpoint reached	2350 psia
6.6	Steam generator safety valves begin to open	1000 psia
8.3	Pressurizer relief valves begin to open	2400 psia
8.3	High pressurizer pressure trip setpoint reached	2400 psia
11.9	High pressurizer level alarm setpoint reached	2.0 ft. above programmed level
46.5	Pressurizer relief valves close	---
89.0	Primary-to-Secondary total heat transfer coefficient (UA) begins to decrease rapidly	---
95.1	High pressurizer pressure alarm setpoint reached	2350 psia
97.3	Pressurizer relief valves begin to open	2400 psia
97.3	High pressurizer pressure trip setpoint reached	2400 psia
104.9	Pressurizer safety valves begin to open	2500 psia
111.5	Pressurizer fills	---
163.6	Maximum RCS pressure	3393 psia
251.3	Reactor coolant pumps trip on cavitation	---
252.7	Low RCS flow trip setpoint reached	95% of full power flow
253.0	Steam generators begin to regain liquid inventory	---

TABLE 3 (Cont'd)

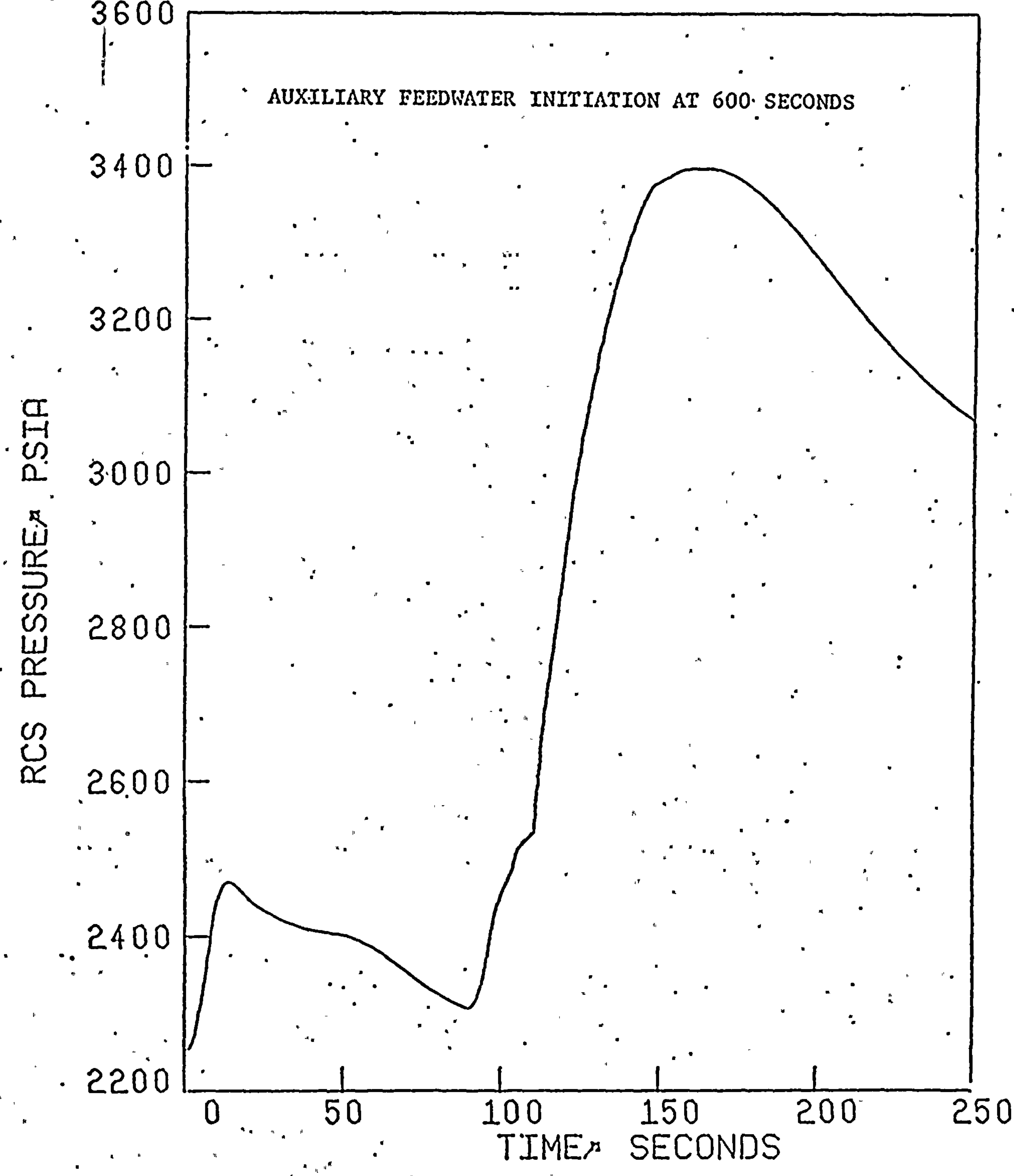
Time	Event	Setpoint or Value
260.3	Steam bubble begins to form in pressure vessel head	---
281.9	Pressurizer empties	---
600.0	Operator initiates auxiliary feedwater	5% of full power feedwater flow
600.0	Operator initiates SIAS	---
745.0	Boron reaches core	---



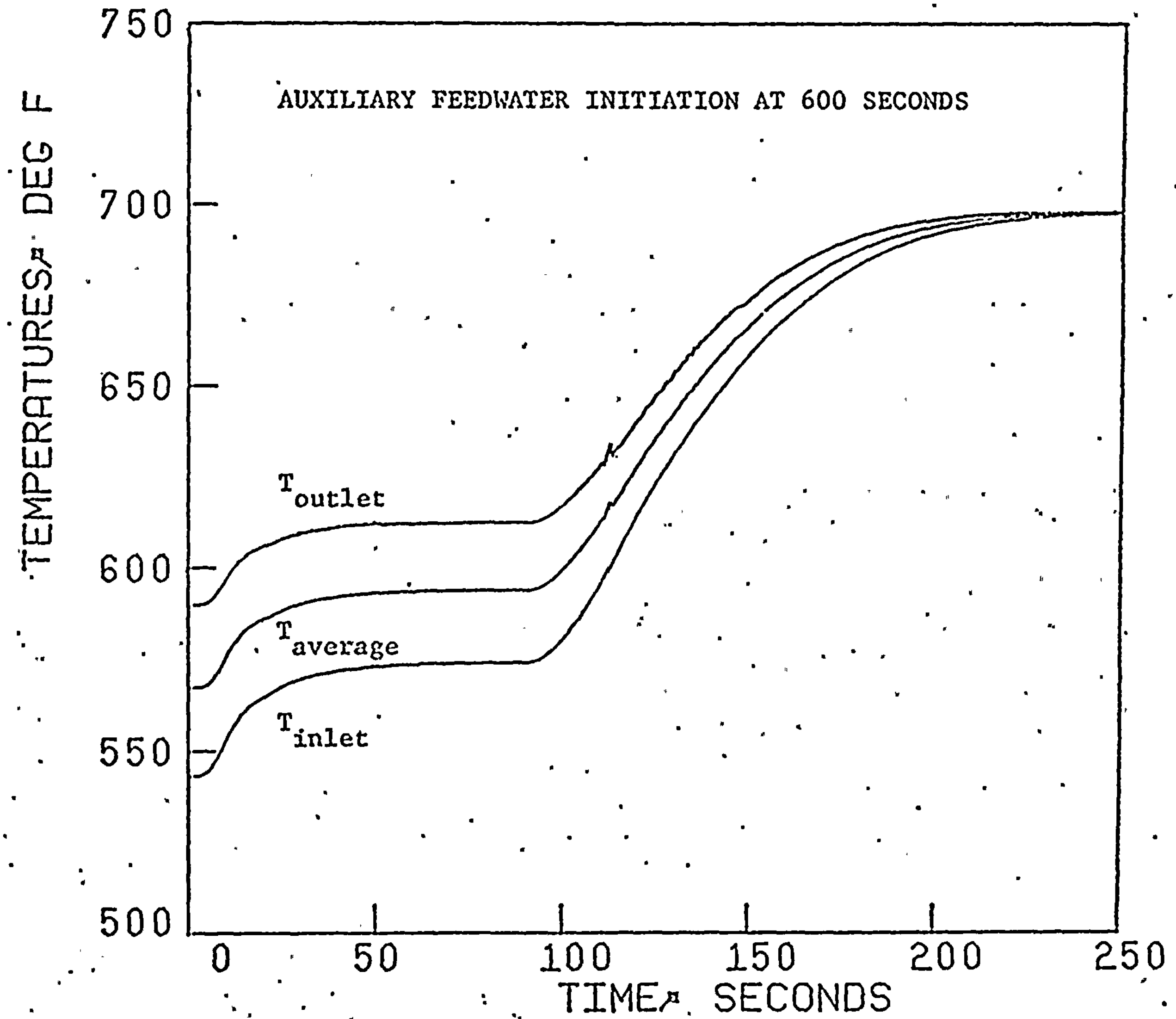
ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM CORE POWER vs TIME	FIGURE 1F



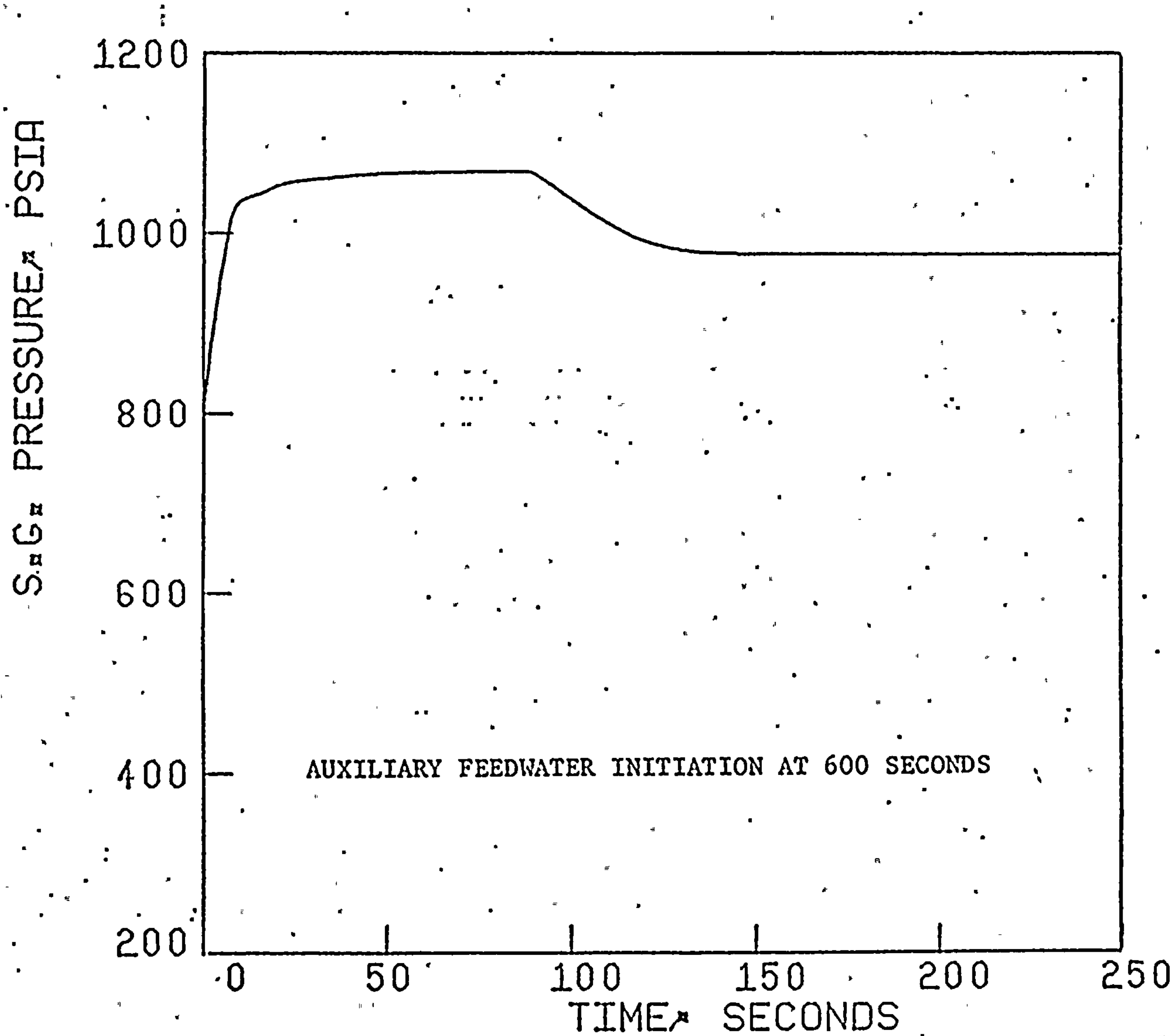
ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM CORE HEAT FLUX vs TIME	FIGURE 2



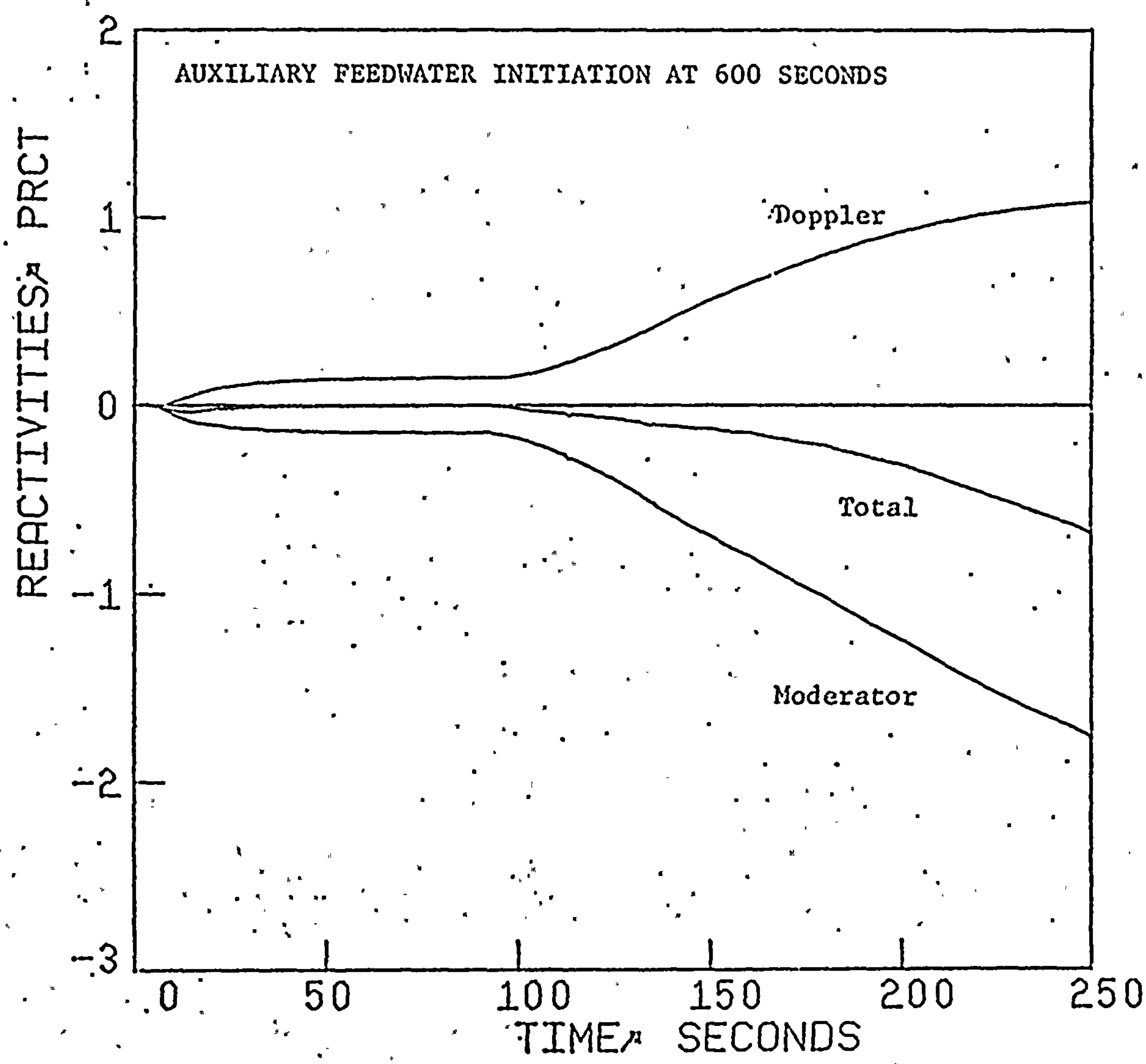
ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM REACTOR COOLANT SYSTEM PRESSURE vs TIME	FIGURE 3



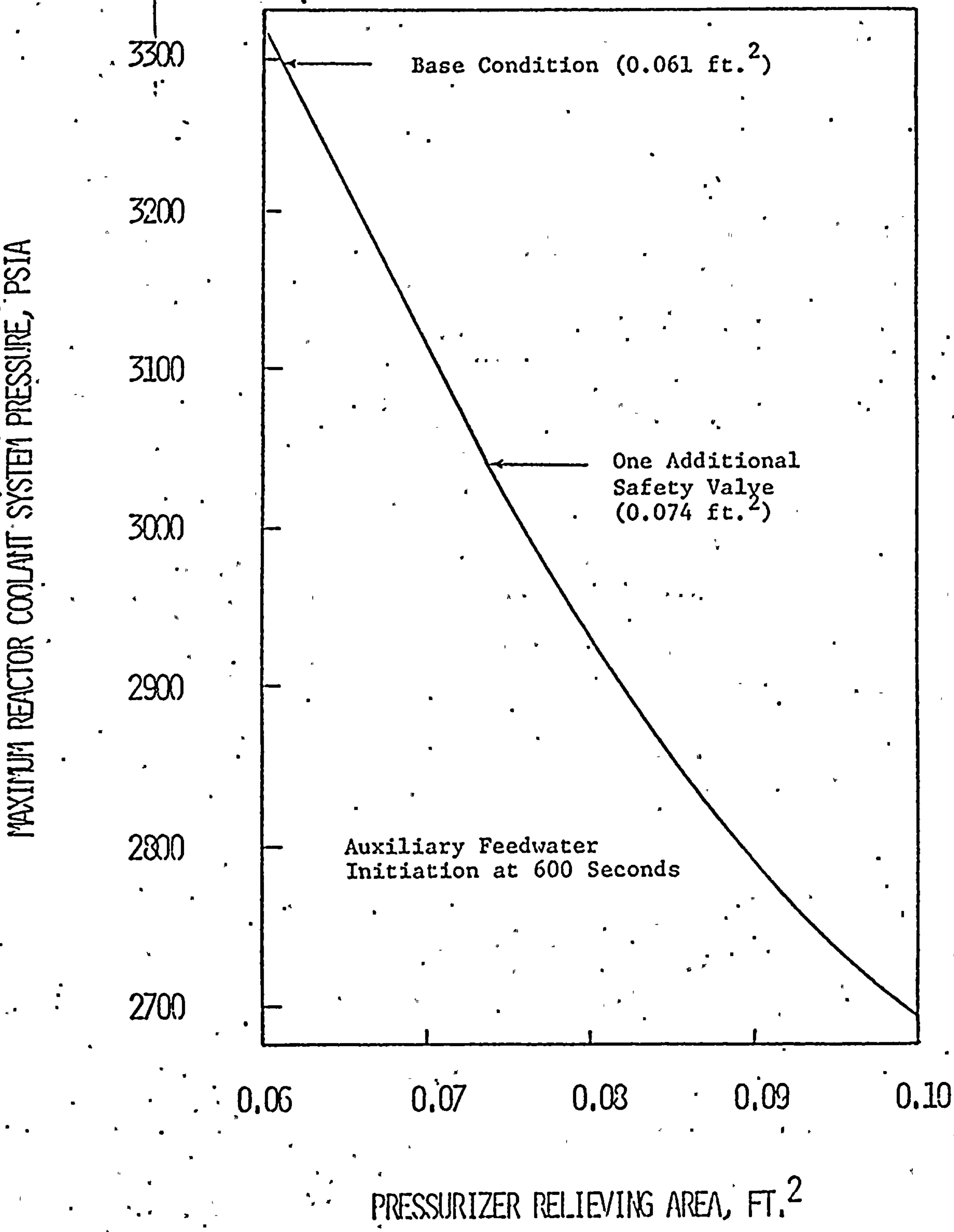
ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM REACTOR COOLANT TEMPERATURE vs TIME	FIGURE 4



ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM STEAM GENERATOR PRESSURE vs TIME	FIGURE 5



ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM REACTIVITY vs TIME	FIGURE 6



ST. LUCIE 1 ATWS REPORT	LOSS OF LOAD WITH FAILURE TO SCRAM MAXIMUM REACTOR COOLANT SYSTEM PRESSURE VS PRESSURIZER RELIEVING AREA	FIGURE 7