Proposed

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than $+0.5 \times 10^{-4}$ delta k/k/°F at ≤ 70% RATED THERMAL POWER,

b. Less positive than ~~0.0~~ +0.3 $\times 10^{-4}$ delta k/k/°F at > 70% RATED THERMAL POWER, and

c. Less negative than -2.7×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exceptions 3.10.2 and 3.10.5.

NO SIGNIFICANT HAZARDS CONSIDERATION

NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed change to the St. Lucie Unit 2 Technical Specifications on Moderator Temperature Coefficient (MTC) provides more operating flexibility and removes restrictive operational requirements above 70% power. The desired change is to permit an MTC of $+0.3 \times 10^{-4} \Delta\rho/^{o}F$ above 70% power.

In accordance with the provisions of 10CFR50.92, the proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change to the MTC limit is an input parameter in various transient and accident analyses. Allowing the operating MTC to be more positive does not influence whether or not the transient is more or less likely to occur. As described in the attached Safety Evaluation, safety analyses have been performed to demonstrate that any transients or accidents whose results would be affected by a more positive MTC limit do not have consequences that are significantly worse than previously evaluated. In addition, the revised analyses, incorporating the proposed change in MTC limit continue to demonstrate that all appropriate analyses criteria reported in the Reload Analysis Report (Reference 1), are met. In particular, the change does not increase previously calculated site boundary doses and the upset pressure limit for peak RCS pressure is not exceeded. Therefore, the proposed change in the Technical Specification MTC limit does not involve any increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change in the positive Technical Specification MTC limit does not constitute any change in the procedures for plant operation or hardware nor does it require any change in the accident analysis methodology discussed in the Reload Analysis Report (Reference 1). Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change in the positive MTC Technical Specification limit, does not significantly change the degree of protection for the design basis events as was discussed in response to Question 1 above because detailed calculations showed that incorporation of the more positive MTC limit, yield results which are within the existing acceptance criteria. Therefore, the operation of the facility in accordance with the proposed amendment involves no significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered least likely to involve significant hazards considerations. Example (vi) from the Federal Register relates to "a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

The proposed change in the Technical Specification MTC limit is similar to Example (vi) in that the requirements are relaxed; however, the results of the changes are clearly within the acceptance criteria with respect to the SRP requirements.

From the considerations detailed above, it is concluded that the proposed change to the MTC Technical Specification 3.1.1.4 does not present a significant hazard as discussed in 10CFR50.92.

Based on the attached Safety Evaluation, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Environment Statement (Reference 2).

REFERENCES

1. Letter, J. W. Williams, Jr. to D. G. Eisenhut, Docket No. 50-389, "St. Lucie Unit 2 Proposed License Amendment Cycle 2 Reload," L-84-148, June 4, 1984.

2. Final Environmental Statement Related to the Operation of the St. Lucie Plant Unit No. 2, NUREG-0842, April 1982.

SAFETY EVALUATION

Safety Evaluation for a Proposed Change to the St. Lucie Unit 2 Technical Specifications on Moderator Temperature Coefficient

1.0 Introduction

As a result of the high RCS dissolved boron concentrations (Cycle 3, 1188 ppm) required to accommodate 18-month cycles and 2700 MWt operation, Beginning-of-Cycle (BOC) Moderator Temperature Coefficients (MTCs) at St. Lucie Unit 2 have become more positive, at times approaching the Technical Specification Limits on MTC. This poses an operational constraint during power ascensions, particularly at the 70% power plateau. At this power level, the current Technical Specification limit on MTC changes abruptly from $\leq +0.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ (at and below 70% power) to $+0.0 \times 10^{-4}$ $\Delta\rho/^{o}F$ (above 70% power). To satisfy the more restrictive limit above 70% power it has been necessary to hold at 70% power while xenon builds up. The buildup of xenon adds the negative reactivity necessary to reduce the critical boron concentration. The reduction in dissolved boron concentration, in turn, reduces the MTC below the Technical Specification limit. To avoid such time consuming and costly delays in power ascension, FP&L proposes to revise the current Technical Specification limit to provide more operating flexibility. The desired change is to permit an MTC of $+0.3 \times 10^{-4}$ $\Delta\rho/^{o}F$ above 70% power.

2.0 Safety Evaluation

This safety evaluation has been prepared to support the proposed Technical Specification 3.1.1.4 change described above. The safety evaluation performed addresses the impact of a $+0.3 \times 10^{-4}$ $\Delta\rho/^{o}F$ MTC above 70% reactor power on the entire spectrum of anticipated operational occurrences and postulated accidents. These analyses can be grouped in the following categories:

1. Increase in heat removal by the secondary system.

2. Decrease in heat removal by the secondary system.

3. Decrease in reactor coolant flow rate.

4. Reactivity and power distribution anomalies.

5. Decrease in reactor coolant system inventory.

6. Loss of coolant events.

An increase in the positive Moderator Temperature Coefficient (MTC) limit has an impact primarily on the heatup transients (Categories 2, 3, 4) and the peak reactor coolant system pressure resulting from such events. The more positive MTC limit may also affect the reactivity insertion as a function of moderator density input to the LOCA evaluations (category 6 above).

2.1 Decrease in Heat Removal by the Secondary System

The transients considered under Category No. 2, "Decrease in Heat Removal by the Secondary System" are presented below.

2.1.1 Loss of External Load

As stated in Reference 1, the core and system performance following a loss of External Load would be no more adverse than those following a Loss of Condenser Vacuum (LOCV), which is discussed below.

2.1.2 Turbine Trip

As stated in Reference 1, this transient is bounded by the LOCV analysis discussed below.

2.1.3 Total Loss of Forced Reactor Coolant Flow

This event was analyzed assuming an MTC of $+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore, the conclusions of the Reference analysis remain valid.

2.1.4 Loss of Normal Feedwater

As stated in Reference 1, this event is bounded by the LOCV analysis discussed below.

2.1.5 Loss of Condenser Vacuum

This event was analyzed for Cycle 2 assuming an MTC of 0.0×10^{-4} $\Delta\rho/^{o}F$. To support a $+.3 \times 10^{-4}$ $\Delta\rho/^{o}F$ MTC limit at > 70% reactor full power, a reanalysis was performed. The results of this reanalysis demonstrate that the Loss of Condenser Vacuum event with the new MTC limit (and 1500 plugged tubes per steam generator) will not result in peak RCS pressure or main steam pressure in excess of their respective upset pressure limits. A summary of the results and comparison of key input parameters with Cycle 2 values are presented in Attachment 1. Attachment 2 provides a thorough discussion of the analysis, the assumptions and results, following the Reference Analysis format.

2.1.6 Feedwater Line Break

The Feedwater Line Break event was analyzed for Cycle 2 assuming an MTC of 0.0×10^{-4} $\Delta\rho/^{o}F$. To

support the new MTC limit, a reanalysis was performed. From this analysis, it is concluded that the Feedwater Line Break Event with the new MTC limit (and 1500 plugged tubes per steam generator) will not lead to a DNBR that is less than the design limit of 1.28 during the transient, the radiological consequences for this event remain a small fraction of 10CFR100 guidelines, and the RCS peak pressure does not exceed the upset pressure limit of 2750 psia. A summary of the results and comparison of key input parameters with the Cycle 2 (Reference analysis) values are presented in Attachment 1. Attachment 2 provides a detailed discussion of the analysis, the assumptions and results following the Reference Analysis format.

2.1.7 Loss of Offsite Power to the Station Auxiliaries (LOAC)

This event was analyzed assuming an MTC of $+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore, the conclusions of the analysis remain valid.

2.2 Decrease in Reactor Coolant Flow Rate

The transients considered under Category 3 above, "Decrease in Reactor Coolant Flow Rate" are discussed below.

2.2.1 Partial Loss of Forced Reactor Coolant Flow

As stated in Reference 1, this event is bounded by the Total Loss of Forced Reactor Coolant Flow analysis discussed below.

2.2.2 Total Loss of Forced Reactor Coolant Flow

This event was analyzed assuming an MTC of $+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore the conclusions of the Reference Analysis remain valid.

2.2.3 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

This event was analyzed assuming an MTC of $+.4 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore the conclusions of the Reference Analysis remain valid.

2.3 Reactivity and Power Distribution Anomalies

The transients considered under Category 4, "Reactivity and Power Distribution Anomalies" are discussed below.

2.3.1 Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Condition

This event was analyzed assuming an MTC of $+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore, the conclusions of the Reference analysis remain valid.

2.3.2 Uncontrolled Control Element Assembly Withdrawal at Power (CEAW)

The CEAW event was analyzed assuming an MTC of $+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore, the conclusions of the Reference Analysis remain valid.

2.3.3 Boron Dilution

The conclusions of Reference 1 remain valid.

2.3.4 CEA Ejection

This event was analyzed assuming an MTC of $+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ in the Reference 1 analysis, therefore, the conclusions of the Reference analysis remain valid.

2.4 LOCA - Category 6

In addition to the non-LOCA transient evaluations, it was necessary to address the impact of the proposed changes in the positive MTC limit on the LOCA analysis results. An evaluation of this sort was necessary because the $+0.3 \times 10^{-4}$ $\Delta\rho/^{o}F$ MTC limit is more positive than the $+0.2 \times 10^{-4}$ $\Delta\rho/^{o}F$ MTC assumed in deriving the reactivity insertion ($\Delta\rho$) versus moderator density curve used in the Cycle 2 LOCA evaluations. The impact of the more positive MTC limit was addressed by re-calculating a $\Delta\rho$ versus moderator density curve for Cycle 3. To do so, several 2-D ROCS calculations were run, based on a $+0.3 \times 10^{-4}$ $\Delta\rho/^{o}F$ MTC, for the range of moderator densities considered in the full power LOCA analysis. These 2-D ROCS calculations produced conservative estimates of the reactivity insertion ($\Delta\rho$) as a function of moderator density because the ROCS model used does not have the

capability of representing the changes in moderator density separately from changes in moderator temperature. That is, the interpolation on both moderator density and temperature are based on a single parameter: moderator density. Thus, the reactivity insertion versus moderator density calculated by this ROCS model includes the additional reactivity insertion due to the change in moderator temperature. Nevertheless, these calculations showed that the reactivity insertion (Δp) versus moderator density curve previously used in the Cycle 2 LOCA analysis conservatively bounds the one calculated for a $+0.3 \times 10^{-4}$ $\Delta p/^{o}F$ MTC limit. Consequently, the LOCA results (both Cycle 2 and Cycle 3) are not affected by increasing the MTC limit from +0.2 to $+0.3 \times 10^{-4}$ $\Delta p/^{o}F$.

REFERENCES

1. Letter, J. W. Williams, Jr. to D. G. Eisenhut, Docket No. 50-389, "St. Lucie Unit 2 Proposed License Amendment Cycle 2 Reload", L-84-148, June 4, 1984.

2. Letter, E. L. Trapp (CE) to J. L. Perryman (FPL), "Transmittal of Analytical Results Supporting Increased Positive Technical Specification MTC Limits for St. Lucie 2", F2-CE-R-037, October 11, 1985.

ATTACHMENT 1

I. FEEDWATER LINE BREAK EVENT

The Feedwater Line Break (FLB) event was analyzed for St. Lucie Unit II Cycle 2 using an MTC of 0.0×10^{-4} $\Delta\rho/^{o}F$ and an assumption that 200 tubes were plugged in each steam generator. To support the new MTC limit and the new steam generator plugged tube limit, several computer runs were made at full power with an MTC of $+0.3 \times 10^{-4}$ $\Delta\rho/^{o}F$ and 1500 plugged tubes per steam generator to determine the limiting break size which yields the maximum peak RCS pressure. The analysis results were acceptable; that is, the maximum peak RCS pressure for the limiting feedline break size did not exceed the RCS pessure upset limit of 2750 psia. A comparison of the more important inputs and final results for this event against those reported in the original Cycle 2 license submittal is provided below.

Analysis Input	Units	New Value	Old Cycle 2 Value	Comment
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774	2774	----
Initial Core Coolant Inlet Temperature	^{o}F	552	552	maximum T_{inlet} used to maximize the potential for filling the pressurizer
Initial RCS Vessel Flow Rate	gmp	363,000	363,000	minimum guaranteed flow assumed available after tube plugging
Initial Reactor Coolant System Pressure	psia	2180	2180	----
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/^{o}F$	0.3	0.0	----
Doppler Coefficient Multiplier	----	.85	.85	----
CEA Worth at Trip	$\%\Delta\rho$	-7.0	-7.0	----
Number of Plugged Tubes	#/S.G.	1500	200	new value has a 200 plugged tube asymmetry limit
Results				
Limiting Break Size	ft^2	.375	.300	higher MTC shifts limiting break size due to earlier HPPT
Initial S.G. Pressure	psia	893	809	815 psia design value
Peak RCS Pressure	psia	2725*	2690	<2750 psia reported

*Includes extra 10 psia for conservatism.

Analysis Input	Units	New Value	Old Cycle 2 Value	Comment
Minimum DNBR	----	>1.42	>1.31	>1.28 reported
Thyroid Dose	rem (EAB,LPZ)	2.05,.90	2.05,.90	The method used to calculate dose rates is independent of the number of plugged S.G. tubes
Whole Body Dose	rem (EAB,LPZ)	.0014,.0055	.0014,.0055	

II. LOSS OF CONDENSER VACUUM EVENT

The Loss of Condenser Vacuum (LOCV) event was analyzed for St. Lucie Unit II Cycle 2 using an MTC of 0.0×10^{-4} $\Delta\rho$ /^{o}F and an assumption that 200 tubes were plugged in each generator. To support the new MTC limit and the new steam generator plugged tube limit, computer runs were made at full power with an MTC of $+0.3 \times 10^{-4}$ $\Delta\rho$ /^{o}F and 1500 plugged tubes per steam generator (instead of $+0.2 \times 10^{-4}$ $\Delta\rho$ /^{o}F). The analysis results were acceptable; that is, both the peak RCS and steam generator did not exceed their respective pressure upset limits of 2750 psia and 1100 psia. A comparison of the more important analysis inputs and final results for this event against those reported in the original Cycle 2 license submittal is provided below.

Analysis Input	Units	New Value	Old Cycle 2 Value	Comment
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774	2774	----
Initial Core Coolant Inlet Temperature	^{o}F	535	535	minimum T_{inlet} used to reduce initial S.G. pressure and delay MSSV opening
Initial Reactor Coolant System Pressure	psia	2180	2180	----
Initial RCS Vessel Flow Rate	gmp	363,000	363,000	minimum guaranteed flow assumed available after tube plugging
Moderator Temperature Coefficient	$\times 10^{-4}$ $\Delta\rho$/^{o}F	0.3	0.0	----
Doppler Coefficient Multiplier	----	.85	.85	----
CEA Worth at Trip	%$\Delta\rho$	-5.5	-5.5	----
Number of Plugged Tubes	#/S.G.	1500	200	new value has a 200 plugged tube asymmetry limit
Results				
Peak RCS Pressure	psia	2745*	2710	<2750 psia reported
Peak S.G. Pressure	psia	1029	1025	<1100 psia reported
Initial S.G. Pressure	psia	688	687	815 psia design value

*Includes extra 15 psia for conservatism.

ATTACHMENT 2

3.2.2.6 FEEDWATER LINE BREAK EVENT

3.2.2.6.1 Identification of Causes

The Feedwater Line Break Event with a Loss of AC was re-analyzed to ensure that the DNBR limit is not exceeded, the site boundary doses would not exceed a small fraction of the 10CFR100 guidelines, and the peak RCS pressure does not exceed the upset pressure limit when (a) the MTC limit above 70% power is 0.3×10^{-4} $\Delta\rho$ /^{o}F and (b) as many as 1500 tubes are plugged in each steam generator.

A Feedwater Line Break Event is defined as the failure of a main feedwater system pipe during plant operation. A rupture in the main feedwater system rapidly reduces the steam generator secondary inventory causing a partial loss of the secondary heat sink, thereby allowing heat up of the Reactor Coolant System (RCS). The RCS is protected from over-pressurization by the high pressurizer pressure trip. A loss of AC (LOAC) power concurrent with the time of trip exacerbates the RCS pressure increase during the transient.

The limiting location for a feedwater line rupture occurs between the steam generator and the feedwater line check valves, since blowdown of the affected steam generator continues until the steam generator pressure equals the containment back pressure. An adequate secondary heat sink is provided by auxiliary feedwater to the unaffected steam generator.

3.2.2.6.2 Analysis of Effects and Consequences

The analysis of the Feedwater Line Break Event was performed using the initial conditions shown in Table 3.2.2.6-1. A new parametric study showed that a break area of 0.375 ft^2 along with the above initial conditions maximized the RCS pressure spike following reactor trip. This is a larger limiting break size than previously reported. The change in break size as well as the increased peak RCS pressure reported herein are both attributable to the new (more positive) MTC value assumed and to the assumption that 1500 tubes per steam generator are plugged. The following additional assumptions were made for this transient:

A) At time zero:

1) The steam generator feedwater flow to both steam generators is instantaneously reduced to zero.

2) Pressurizer Pressure and Level Programs are set to manual control to maximize the RCS pressure spike.

B) At the time of reactor trip:

1) A loss of AC occurs concurrent with trip and the reactor coolant pumps begin to coast down. Following coastdown, the coolant flow necessary to remove decay heat is maintained by natural circulation.

2) Emergency diesel generators start automatically after the loss of all non-emergency AC power.

3) Auxiliary feedwater is not available until 420 seconds after initiation of AFAS on low steam generator level. AFAS on low steam generator level signal from the unaffected steam generator is conservatively assumed to occur at 30 seconds.

During the event, two sources of radioactivity contribute to the site boundary dose. The initial Technical Specification (Tech. Spec.) activity in the steam generator and the activity associated with the maximum Tech. Spec. allowed primary to secondary leak rate. The analysis assumed that all of the initial activity in both steam generators and the activity added due to the Tech. Spec. allowed primary to secondary leak rate tube leakage are released to the atmosphere with a decontamination factor of 1.0, resulting in the maximum site boundary dose during the transient. This makes the dose calculation independent of the number of tubes plugged in each steam generator.

Table 3.2.0-8 shows the key parameters assumed for the radiological evaluation.

To determine the maximum possible radioactivity release associated with a Feedwater Line Break Event concurrent with a loss of offsite power, the following additional assumptions were made:

1) Offsite power is not restored and action is initiated to bring the plant to a cold shutdown condition;

2) The reactor coolant system specific activity equals the Technical Specification limit of 1.0 μCi/gm (I-131 Dose Equivalent Curies);

3) The secondary system specific activity equals the Technical Specification limit of 0.1 μCi/gm (I-131 Dose Equivalent Curies);

4) The primary to secondary leak rate is the Technical Specification Limit of 1 gpm (0.5 gpm per steam generator).

5) Assumed lower feedwater nozzle level increases leak out of break.

These assumptions increase the calculated steam and activity release, thus maximizing the predicted doses. In addition, the concentration of radionuclides in the steam generators was based on the minimum liquid mass occurring during the transient. This also maximizes the final predicted doses.

3.2.2.6.3 Results

Table 3.2.2.6-3 represents the sequence of events for the Feedwater Line Break Event with a Loss of AC.

Figures 3.2.2.6-1 through 3.2.2.6-7 show the NSSS response for power, heat flux, RCS temperature, RCS pressure, steam generator pressure, pressurizer liquid level and integrated feedwater flow during the transient. The peak RCS pressure calculated during the Feedwater Line Break Event with a Loss of AC, including the pump and elevation head was below the 2750 psia upset pressure limit. The minimum DNBR calculated during the event was greater than the 1.28 DNBR SAFDL. The event doses presented in Table 3.2.2.6-2 for both thyroid and whole body are unchanged from previously analyzed values and therefore continue to be within the acceptance guidelines.

3.2.2.6.4 Conclusions

From this analysis, it is concluded that the Feedwater Line Break Event with the new MTC limit and 1500 plugged tubes per steam generator will not lead to a DNBR that is less than the design limit of 1.28 during the transient, the radiological consequences for this event remain a small fraction of 10CFR100 guidelines, and the RCS peak pressure does not exceed the upset pressure limit of 2750 psia.

TABLE 3.2.2.6-1

KEY PARAMETERS ASSUMED FOR THE FEEDWATER LINE BREAK EVENT

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	^{o}F	552
Initial RCS Vessel Flow Rate	gpm	363,000
Initial Reactor Coolant System Pressure	psia	2180
Moderator Temperature Coefficient	$x10^{-4}\ \Delta\rho/^{o}F$	0.3
Doppler Coefficient Multiplier	----	.85
CEA Worth at Trip	$\%\ \Delta\rho$	-7.0
Number of Plugged Tubes	#/S.G.	1500

TABLE 3.2.2.6-2

RADIOLOGICAL EXPOSURES AS A RESULT OF A FEEDWATER LINE BREAK EVENT

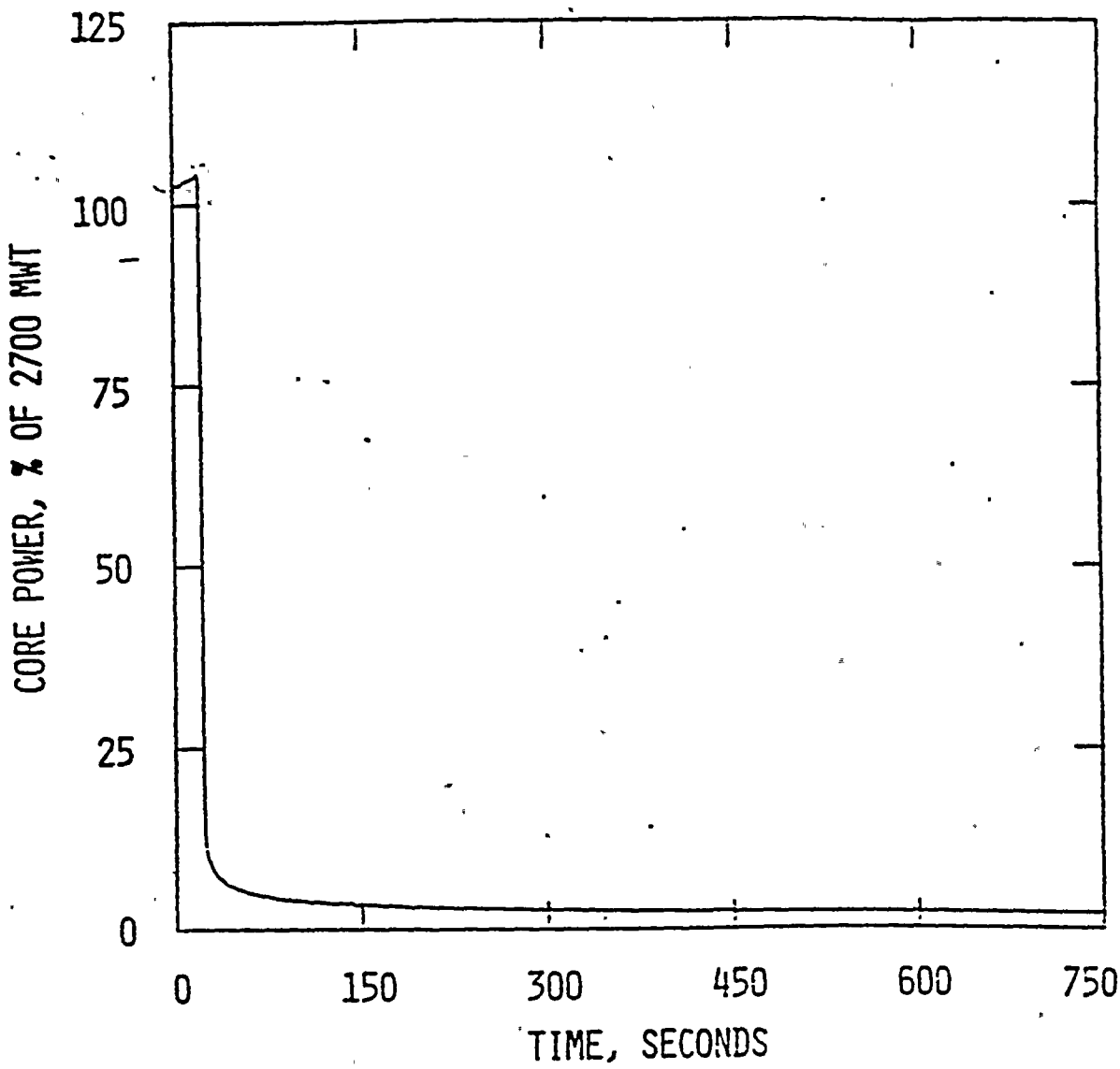
Parameter	Dose (rem)	
	EAB	LPZ
Thyroid	2.05	0.90
Whole Body	.0014	.0055

TABLE 3.2.2.6-3

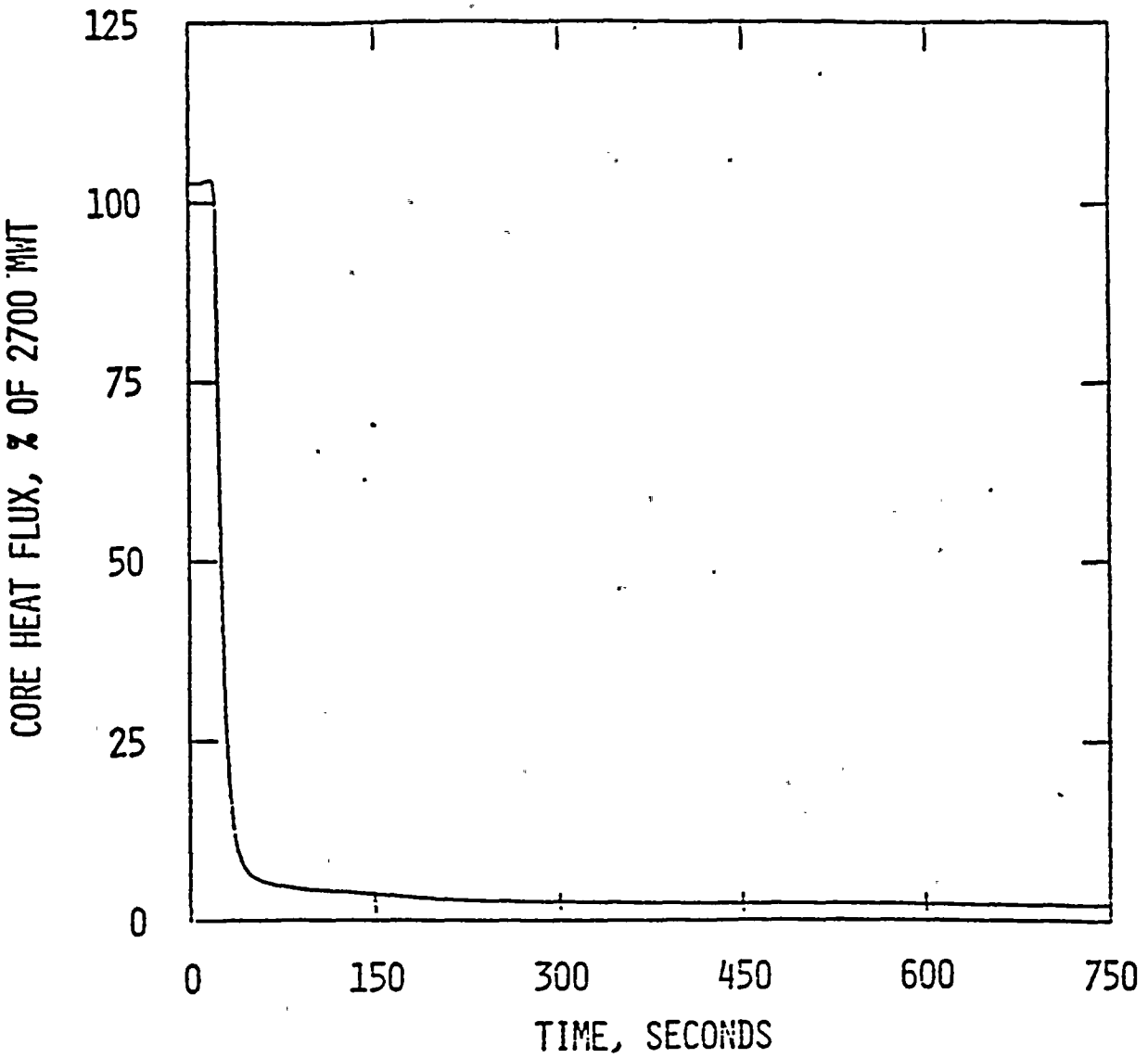
SEQUENCE OF EVENTS FOR FEEDLINE BREAK EVENT

Time (sec)	Event	Setpoint or Value
0.0	Break in the Main Feedwater Line Instantaneous Loss of Feedwater Flow to Both Steam Generators	0.375 ft^2
15.2	Heat Transfer Rampdown Begins in Affected Steam Generator	26000 lbm
18.83	High Pressurizer Pressure Trip Setpoint is Reached	2465 psia
19.5	Primary Safety Valves Open	2525 psia
19.8	Level in Affected Steam Generator Goes Below the Assumed Nozzle Level Steam Would Now Be Blown Out of Break	5000 lbm
19.98	CEAs Begin to Drop Loss of AC Power	
23.3	Maximum RCS Pressure*	<2750 psia
26.9	Unaffected Steam Generator Safety Valves Open	1010 psia
30.0	Low Level Signal in Unaffected Generator Initiating AFAS is Assumed	Setpoint = 26 ft (5% of Narrow Range Tap Span)
78.8	Heat Transfer Rampdown Begins in Unaffected Steam Generator	26000 lbm
145.0	Low Steam Generator Pressure Setpoint Reached, MSIS is Actuated	460 psia
195.0	Level in Unaffected Steam Generator Goes Below the Assumed Nozzle Level	5000 lbm
298.6	Primary Safety Valves Reopen	2525 psia
450.0	Auxiliary Feedwater Begins to be Delivered to Unaffected Steam Generator	
664.0	Maximum Pressurizer Liquid Volume	1206 ft^3

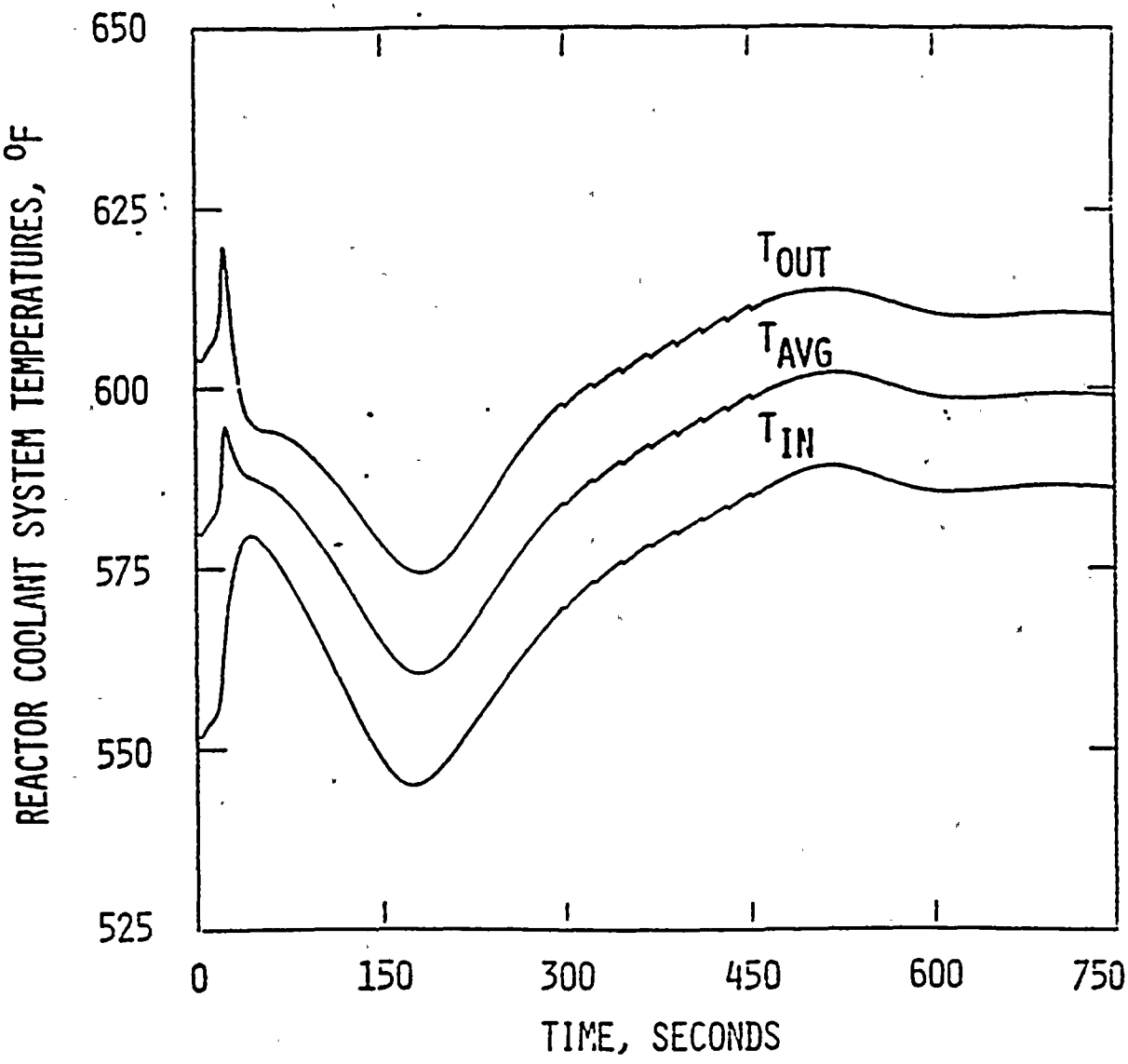
* Includes pump and elevation head



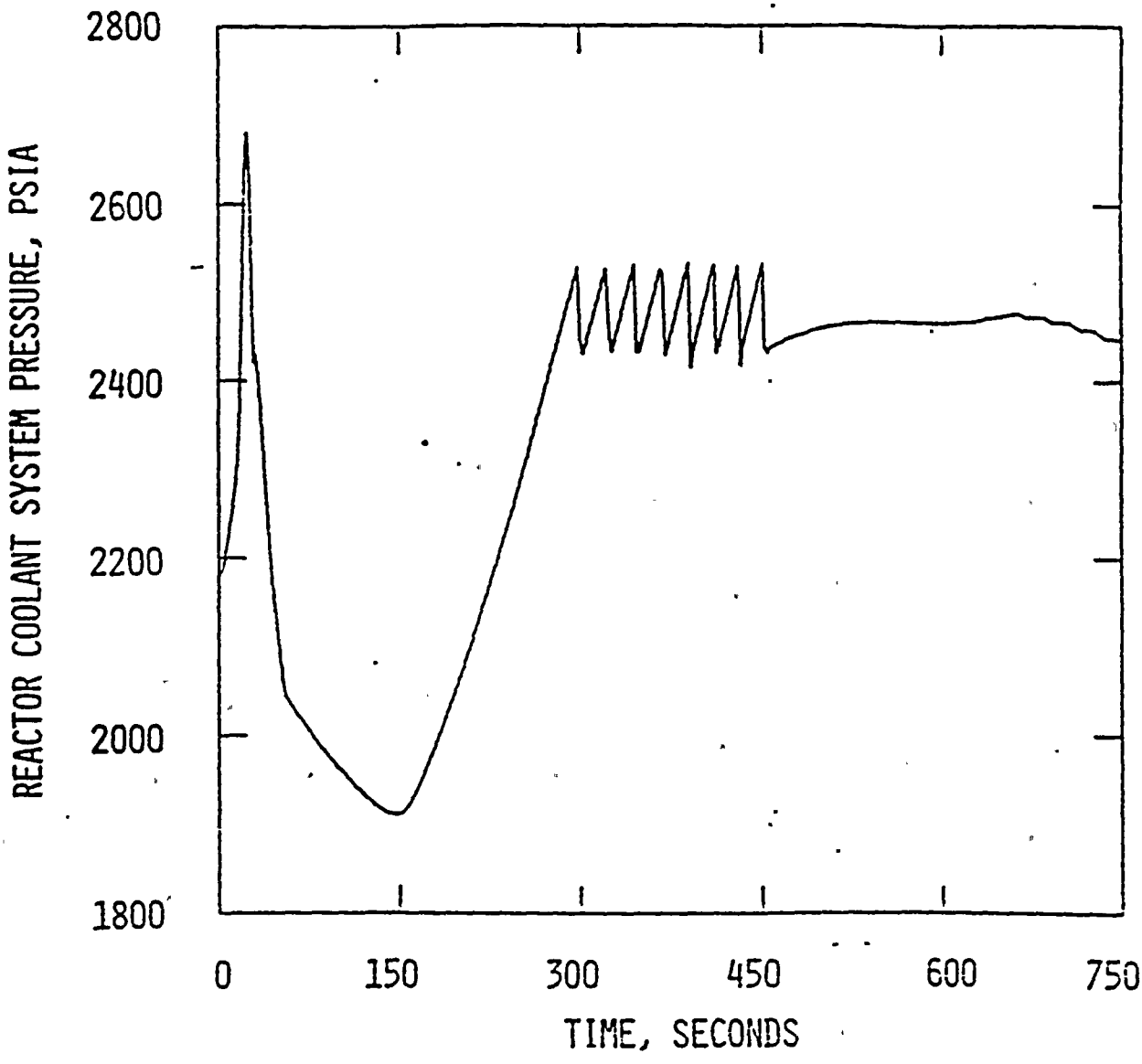
FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT CORE POWER VS TIME	FIGURE 3.2.2.6-



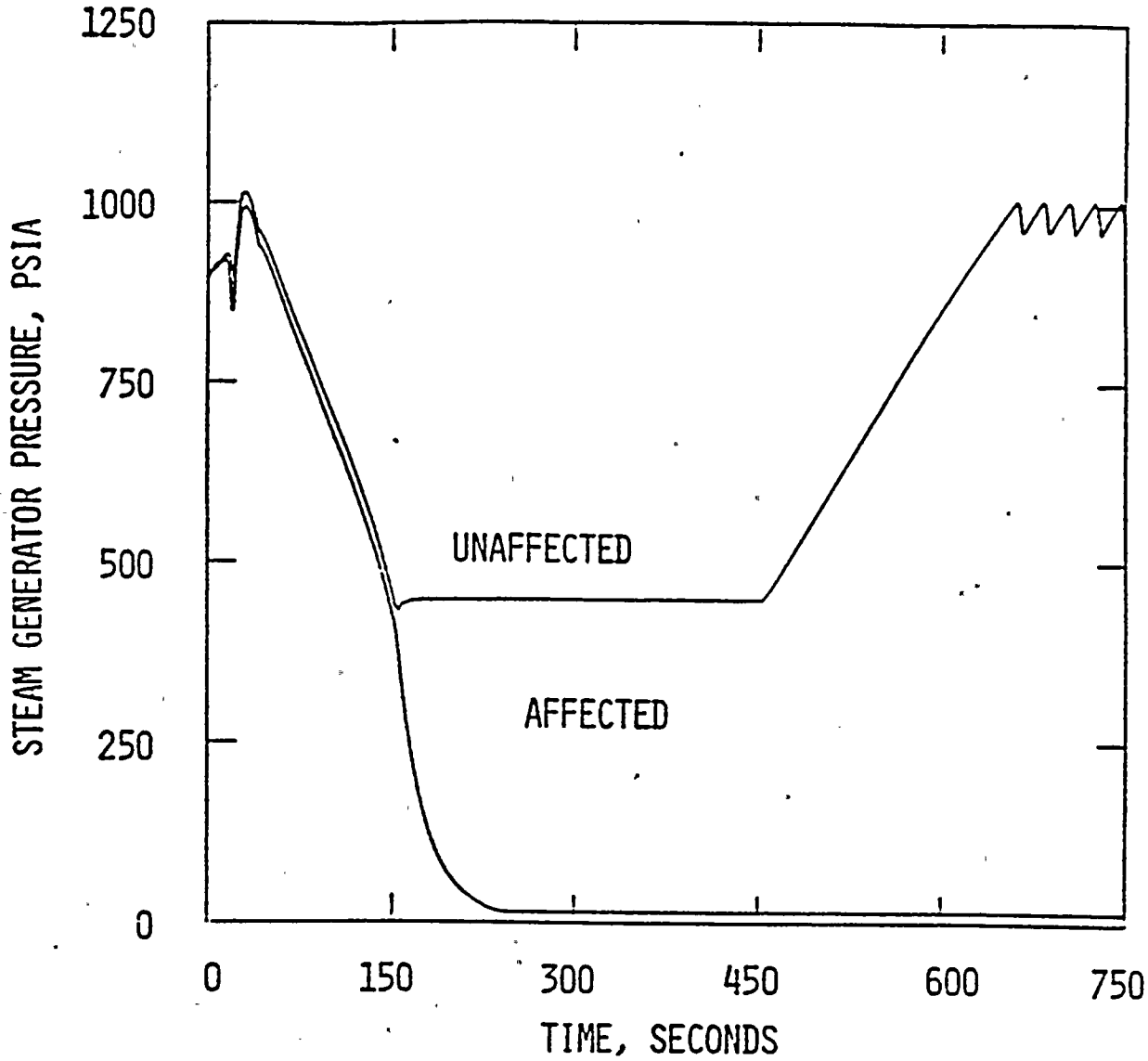
FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT CORE HEAT FLUX VS TIME	FIGURE 3.2.2.6-



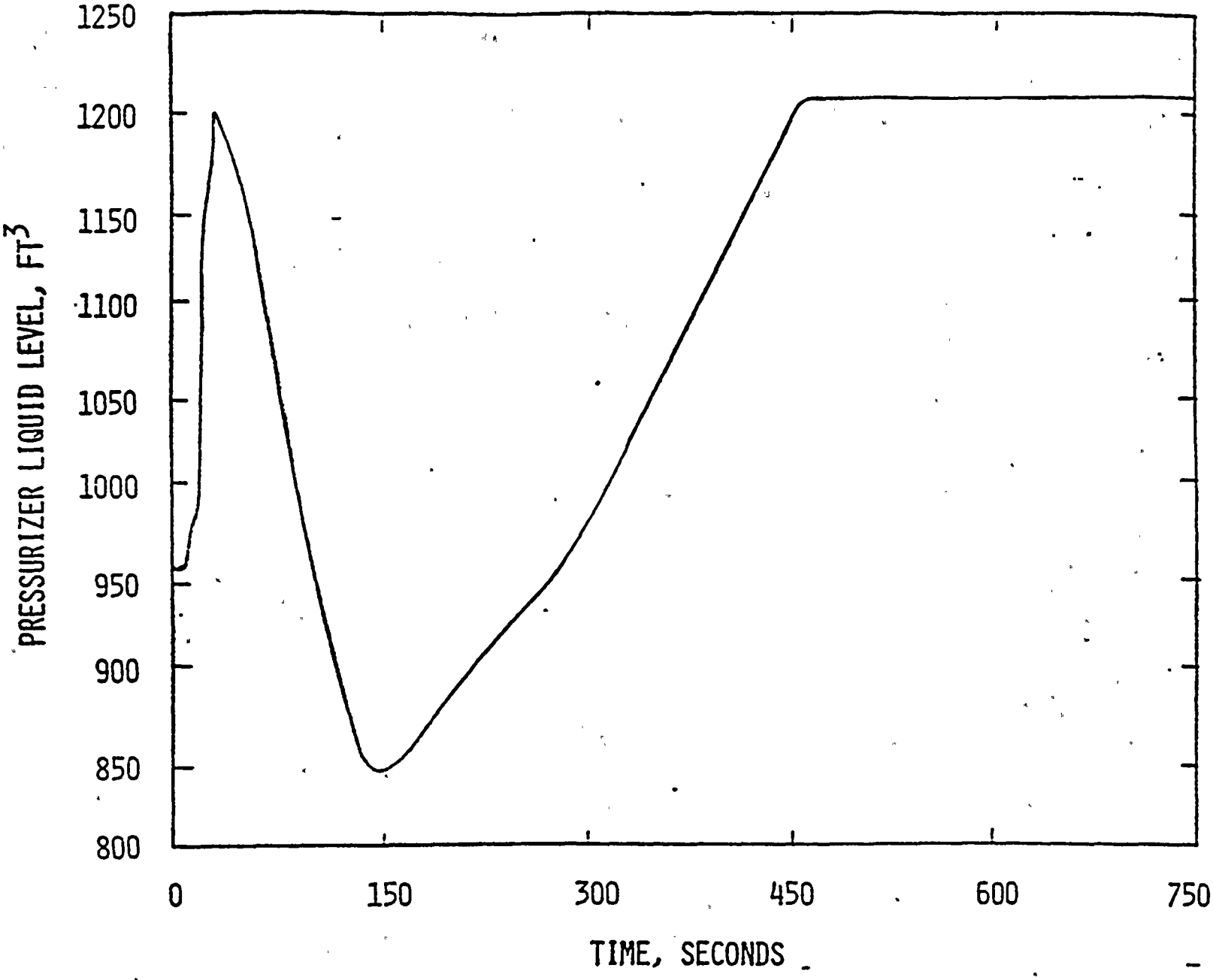
FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 3.2.2.6-3



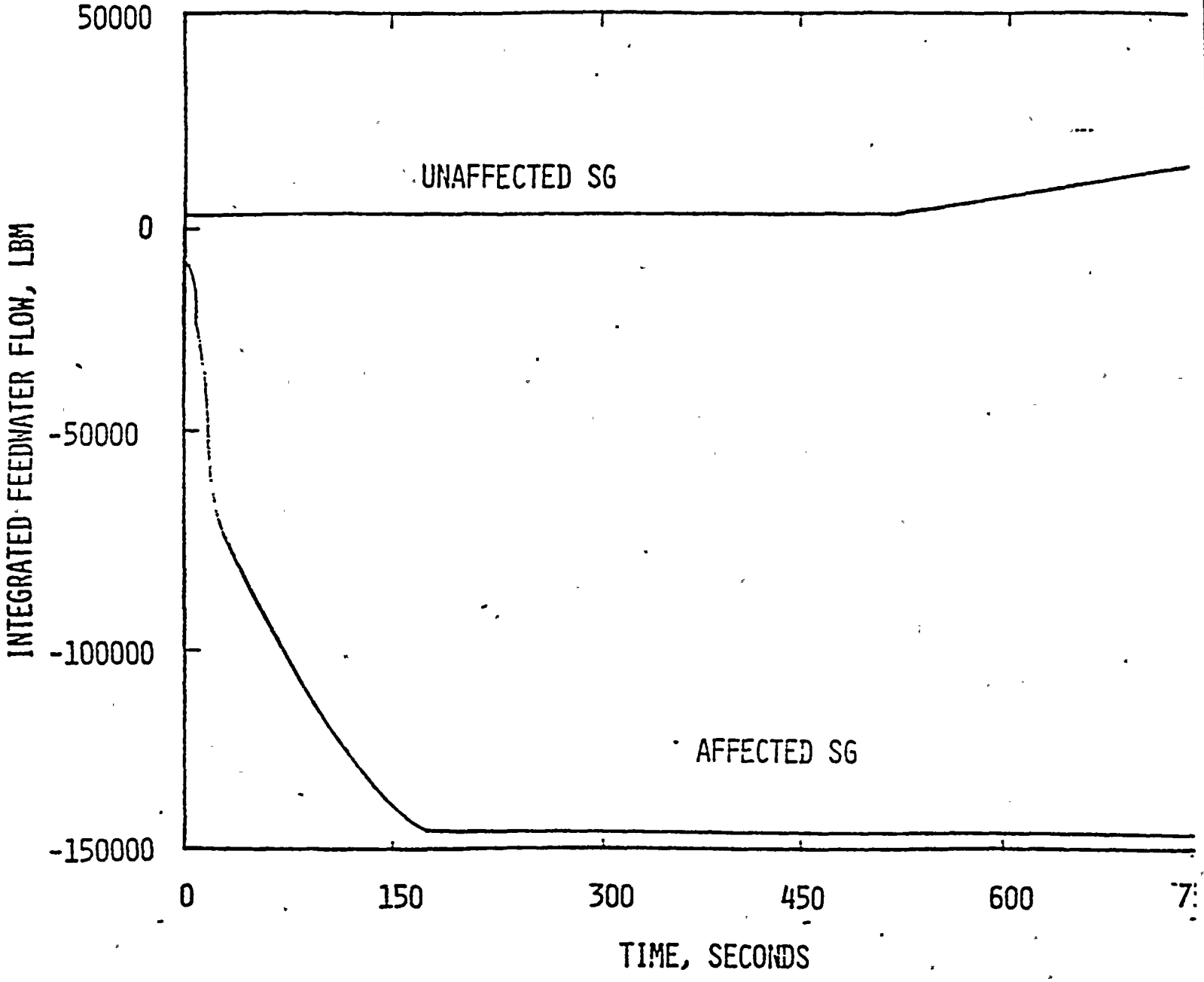
FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 3.2.2.6-



FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 3.2.2.6-5



FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT PRESSURIZER LIQUID LEVEL VS TIME	FIGURE 3.2.2.6-6



FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	FEEDWATER LINE BREAK EVENT INTEGRATED FEEDWATER FLOW VS TIME	FIGURE 3.2.2.6-7

3.2.2.3 LOSS OF CONDENSER VACUUM EVENT

3.2.2.3.1 Identification of Causes

The Loss of Condenser Vacuum event was re-analyzed to demonstrate that the RCS and main steam system pressures do not exceed 110% of design values (i.e., 2750 psia and 1100 psia, respectively) when (a) the MTC limit above 70% power is 0.3×10^{-4} $\Delta\rho$ /^{o}F and (b) as many as 1500 tubes are plugged in each steam generator.

The Loss of Condenser Vacuum (LOCV) will cause the turbine stop valves to close, the main feedwater pumps to trip, and the steam bypass valves to be unavailable. No credit was assumed for a simultaneous reactor trip on turbine trip (due to closure of turbine stop valves). The resulting loss of load and loss of main feedwater cause the steam generator pressure to increase and open the main steam safety valves. The transient is terminated by a reactor trip on high pressurizer pressure.

3.2.2.3.2 Analysis of Effects and Consequences

The LOCV event was initiated at the conditions shown in Table 3.2.2.3-1. The combination of parameters shown in this table maximizes the calculated peak RCS pressure. The most important parameters for this event, other than MTC and the number of plugged tubes, are the initial primary inlet temperature and the fuel temperature coefficients of reactivity. The methods used to analyze this event are consistent with those described in the FSAR.

The initial core average axial power distribution for this analysis was chosen to be a bottom peaked shape. This distribution is assumed because it minimize the negative reactivity inserted during the initial portion of the scram following a reactor trip. This will delay reactor shutdown and hence maximizes the pressure transient.

A Moderator Temperature Coefficient (MTC) of 0.3×10^{-4} $\Delta\rho$ /^{o}F (the most positive value allowed at Hot Full Power (HFP) by the proposed Technical Specification change) was used in the analysis. This MTC caused the greatest amount of positive reactivity feedback to exacerbate the transient increase in pressure. A Fuel Temperature Coefficient (FTC) corresponding to beginning of cycle conditions was used in the analysis. This FTC causes the least amount of negative reactivity feedback to mitigate the transient increase in pressure. The uncertainty on the FTC used in the analysis is shown in Table 3.2.2.3-1.

A minimum allowable initial RCS pressure is used to maximize the rate of change of pressure. This will maximize peak pressure following trip. The lower RCS inlet temperature lowers the initial steam generator pressure resulting in a more severe secondary transient due to the delay in opening of the main steam safety valves. In addition, a loss of offsite power was assumed to occur such that the high pressurizer pressure trip and the low coolant flow trip occur simultaneously. This maximizes the peak RCS pressure following trip.

An assumption that 1500 tubes were plugged in each steam generator was made. This decreases the primary-to-secondary heat transfer and therefore, increases the primary temperatures and also the maximum RCS pressure attained during the transient.

3.2.2.3.3 Results

The LOCV event, initiated from the conditions given in Table 3.2.2.3-1, results in a high pressurizer pressure trip condition at 5.6 seconds. At 9.6 seconds, the primary pressure reaches its maximum value which does not exceed the 2750 psia limit. The increase in secondary pressure is limited by the opening of the main steam safety valves, which open at 9.3 seconds. The secondary pressure reaches its maximum value which does not exceed the 1100 psia limit at 16.9 seconds, after the initiation of the event.

Table 3.2.2.3-2 presents the sequence of events for this event. Figures 3.2.2.3-1 to 3.2.2.3-5 show the NSSS response to power, heat flux, the RCS pressure, RCS coolant temperatures, and steam generator pressure.

The core performance following a LOCV would be no more adverse than those following a Loss of Normal AC Power, which is decribes in Section 3.2.2.4. The radiological consequences due to steam releases from the secondary system would be less severe than the consequences of the Inadvertent Opening of an Atmospheric Dump Valve Event discussed in Section 3.2.1.4.

3.2.2.3.4 Conclusions

The results of this re-analysis demonstrate that the Loss of Condenser Vacuum event with the new MTC limit and 1500 plugged tubes per steam generator will not result in peak RCS pressure or main steam pressure in excess of their respective upset pressure limits.

TABLE 3.2.2.3-1

KEY PARAMETERS ASSUMED FOR THE LOSS OF CONDENSER VACUUM EVENT

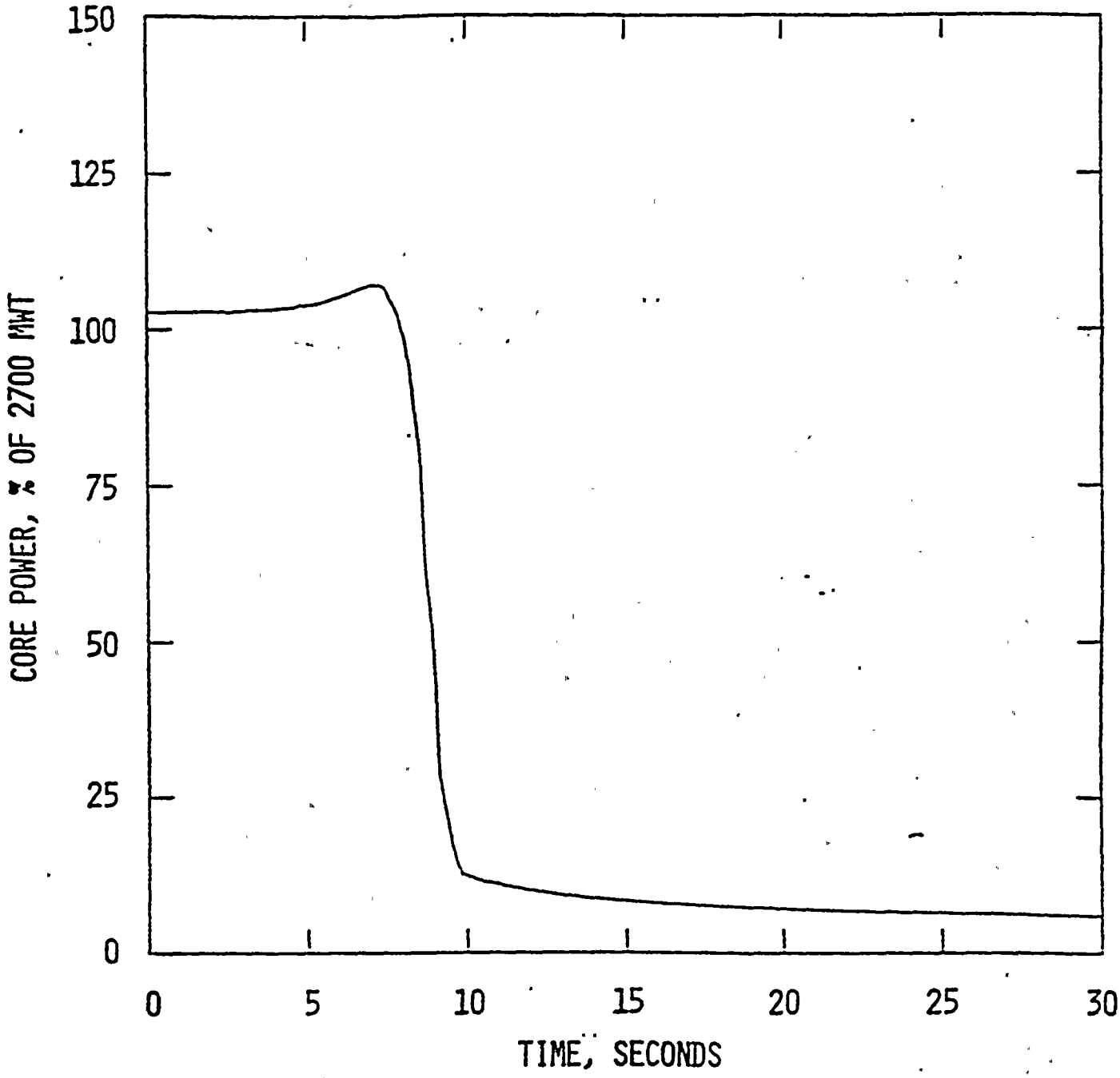
Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	^{o}F	535
Initial Reactor Coolant System Pressure	psia	2180
Initial RCS Vessel Flow Rate	gpm	363,000
Moderator Temperature Coefficient	$X10^{-4} \Delta\rho/^{o}F$	0.3
Doppler Coefficient Multiplier	----	.85
CEA Worth at Trip	$\%\Delta\rho$	-5.5
Number of Plugged Tubes	#/S.G.	1500

TABLE 3.2.2.3-2

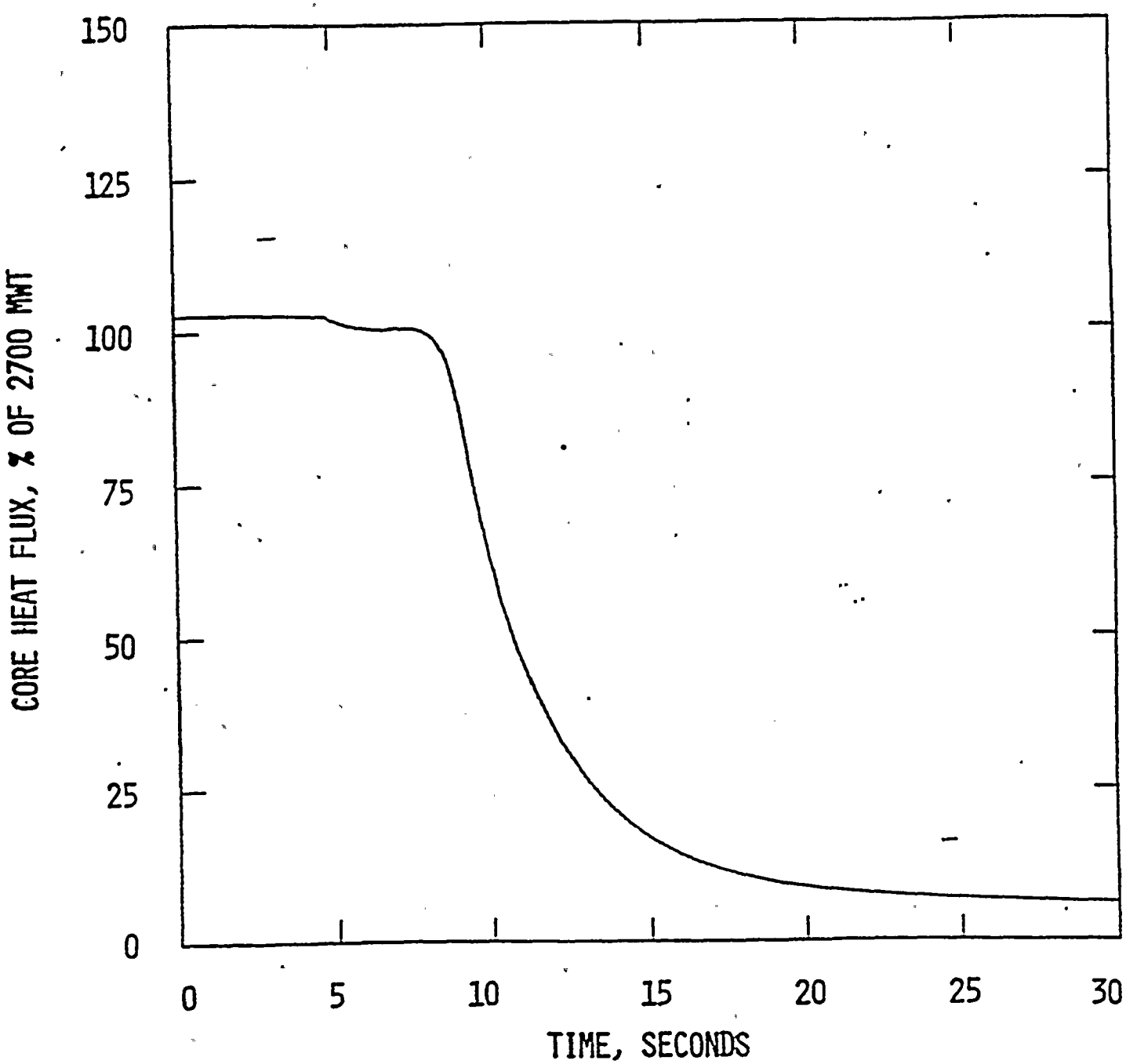
SEQUENCE OF EVENTS FOR THE LOSS OF CONDENSER VACUUM EVENT

Time (sec)	Event	Setpoint or Value
0.0	Closure of Turbine Stop Valves on Turbine Trip due to Loss of Condenser Vacuum	----
4.7	Loss of Offsite Power	----
5.6	High Pressurizer Pressure Trip/ Low Flow Trip Analysis Setpoint Reached	2428 psia/93% of 363,000 gpm
6.5	Pressurizer Safety Valves Open	2525 psia
6.75	Trip Breakers Open	----
7.09	CEAs Begin to Drop Into Core	- - - - -
7.3	Maximum Core Power	108.0% of 2700 MWt
9.3	Steam Generator Safety Valves Open	1000 psia
9.6	Maximum RCS Pressure*	≤2750 psia
13.7	Pressurizer Safety Valves Close	2424 psia
----	Total PSV Release	1094 lbm
16.9	Maximum Steam Generator Pressure	≤1100 psia

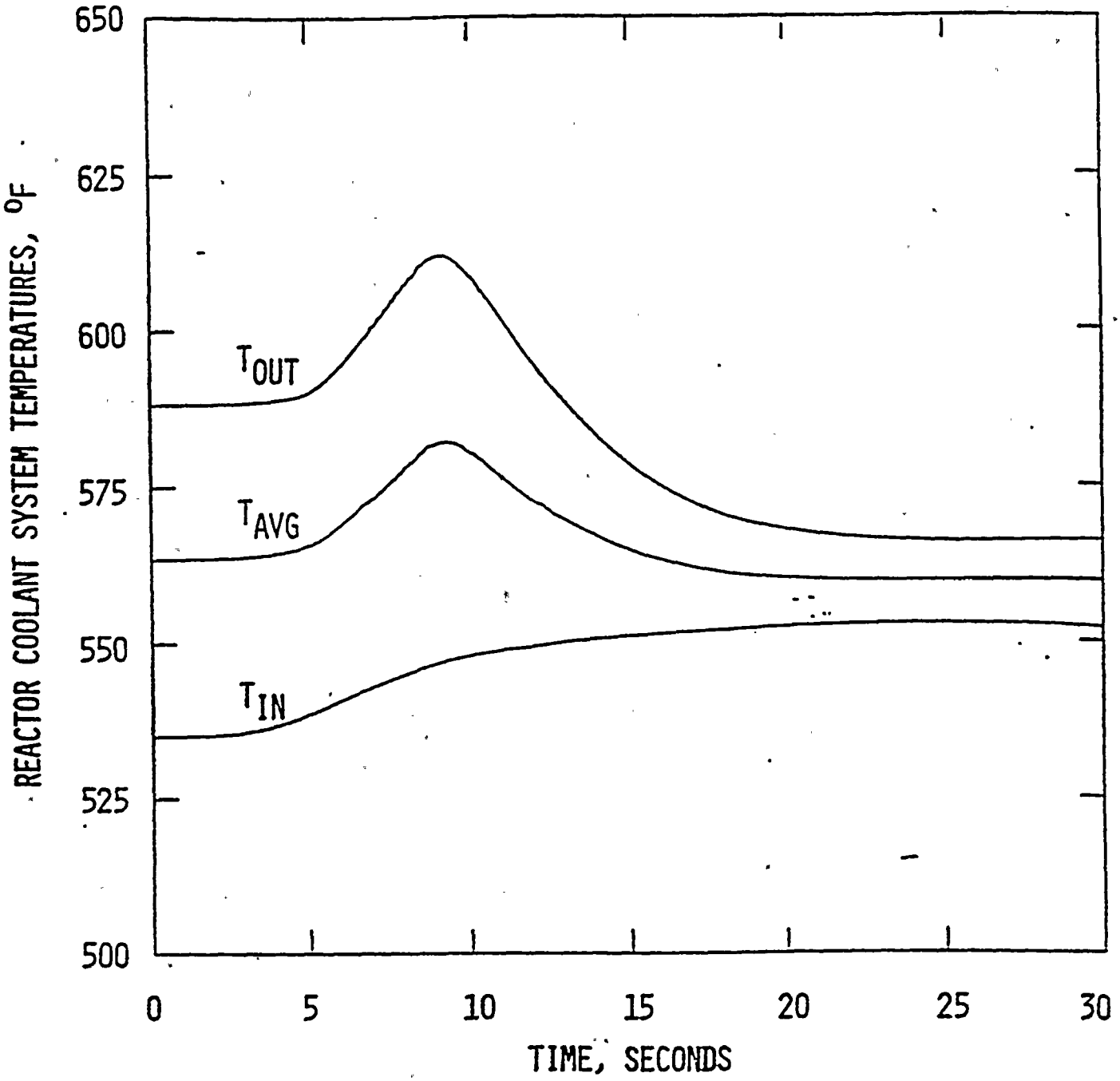
* Including pump and elevation head



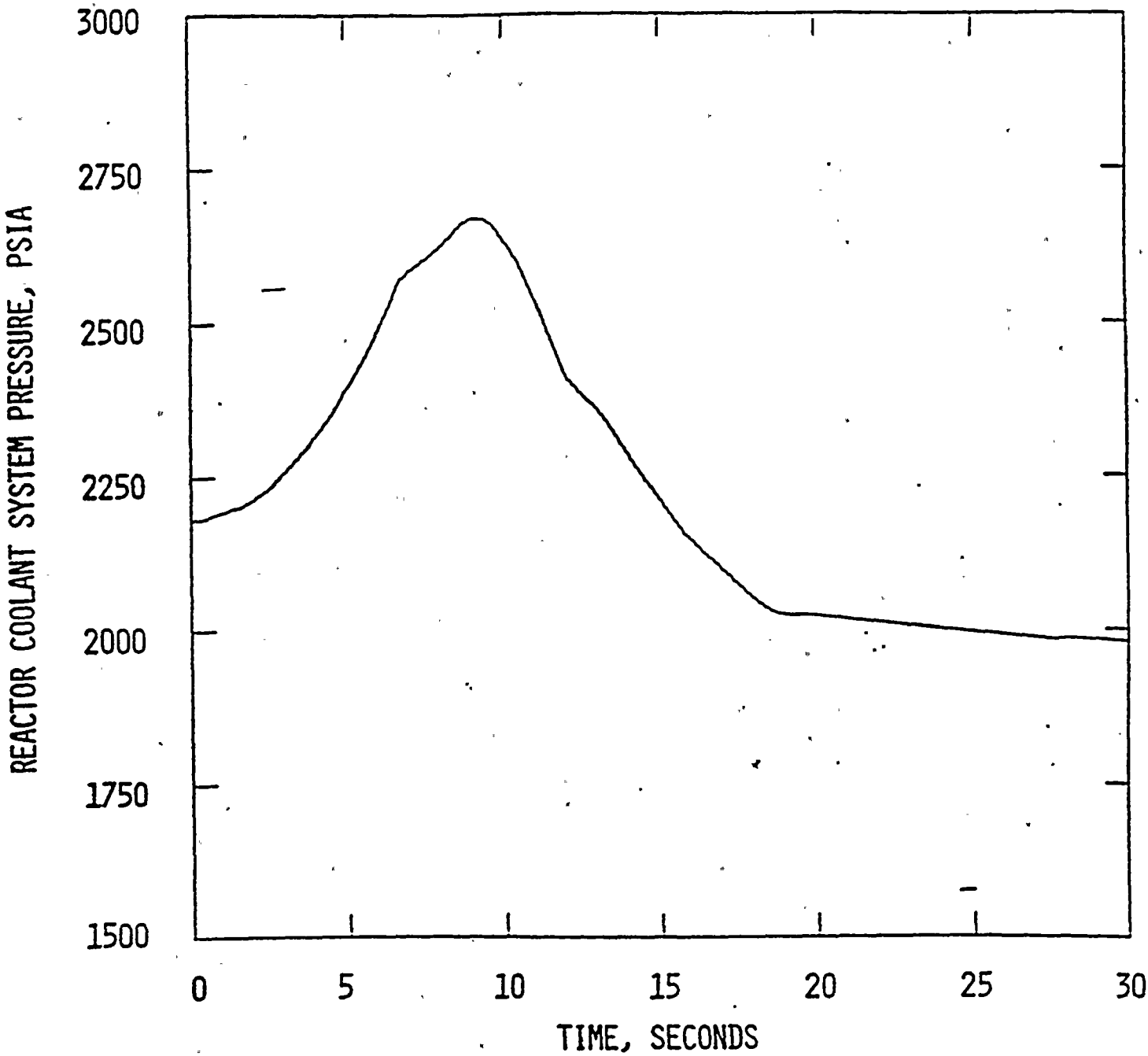
FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	LOSS OF CONDENSER VACUUM EVENT CORE POWER VS TIME	FIGURE 3.2.2.3-.



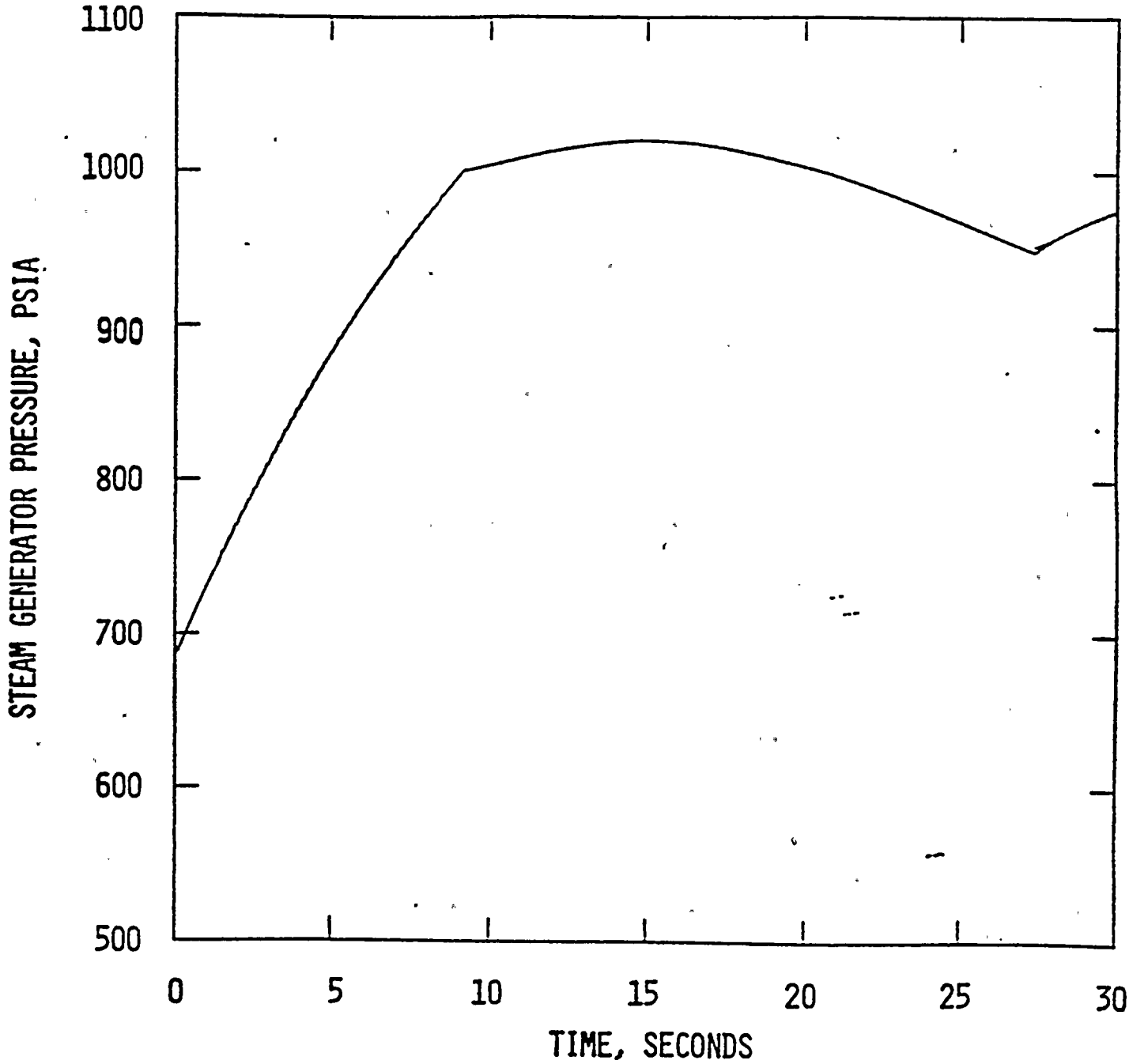
FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	LOSS OF CONDENSER VACUUM EVENT CORE HEAT FLUX VS TIME	FIGURE 3.2.2.3-



FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	LOSS OF CONDENSER VACUUM EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 3.2.2.3-



FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	LOSS OF CONDENSER VACUUM EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 3.2.2.3-1



FLORIDA POWER & LIGHT CO. St. Lucie 2 Nuclear Power Plant	LOSS OF CONDENSER VACUUM EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 3.2.2.3-5