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UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Responds to NRC 801010 request for analysis to demonstrate ability of facility to be safely controlled through total loss of AC power.Facility can endure loss of AC power for at least 3.5-4h.Analyses encl.

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FLORIDA POWER & LIGHT COMPANY

December 18, 1981
L-81-527

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED
DEC 23 1981

Dear Mr. Eisenhut:

Re: St. Lucie Unit 1
Docket No. 50-335
Loss of AC Power

Florida Power & Light has reviewed the NRC letter dated October 10, 1980 which requested that we submit an analysis to demonstrate the ability of St. Lucie Unit 1 to be safely controlled through a total loss of AC power. In our letter (L-80-384) dated November 18, 1980, we indicated that we would address this issue for St. Lucie Unit 1 by modifying the results of the St. Lucie Unit 2 study which was initiated as a result of the NRC Atomic Safety and Licensing Appeal Board decision (ALAB 603) issued July 30, 1980. This St. Lucie Unit 2 study of station blackout, which was recently submitted to the NRC as Appendix 15C-4 of the St. Lucie Unit 2 FSAR, is attached. (Enclosure 1)

The St. Lucie Unit 2 report has been reviewed as discussed above and the results of this review have shown that for St. Lucie Unit 1:

1. Natural circulation and core cooling can be maintained for at least 3 1/2 to 4 hours.

2. The reactor core remains in a subcritical condition.

3. There is no fuel damage.

4. The RCS coolant pressure remains within limits.

5. The resulting radiological doses are within limits.

The St. Lucie Unit 2 analysis and our review for St. Lucie Unit 1 have shown that St. Lucie 1 can endure a complete loss of AC power for at least 3 1/2 to 4 hours. This time period is more than sufficient because of the very low probability that a loss of all AC power will last longer than that time period. It is expected that AC power would be restored within 30 minutes as a result of either restoring offsite power or by starting one or both of the St. Lucie Unit 1 diesel generators. This time estimate is based on a statistical analysis of off site power and emergency diesel generator restoration times.

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This analysis was conducted for testimony for the Atomic Safety and Licensing Appeal Board's review of the grid stability issue for St. Lucie Unit 2. (ALAB 537) It was determined in the analysis that the mean restoration time for offsite power was 27 minutes and there was a 99.5% statistical confidence that the mean restoration time for offsite power will not be greater than 36.6 minutes. A blackout longer than three and one half hours is highly unlikely. We estimate that even with a loss of off site power frequency of 1.0 per year, the probability of not restoring AC power within 3.6 hours is about $1x10^{-7}$.

The St. Lucie Unit 2 report and the Unit 1 review were conducted for a power level of 2560 Mwt. We are reevaluating the results of the Unit 1 study for a power level of 2700 Mwt. We expect to complete this reevaluation and update this report by January 15, 1981 to address this higher power level.

A description of the differences between St. Lucie Unit 1 and Unit 2, as they affect the analysis is attached as Enclosure 2.

We will modify plant operating procedures and training programs to take into effect the results of this study. We expect this effort to be completed by April 15, 1982, and will inform you if it appears this schedule cannot be met.

The only physical change required as a result of this analysis is to place the pressurizer pressure low range (0-1600 psia) instrument on the inverters so it will be available when the emergency diesel generators are not available. The operator uses this instrument to determine subcooling margin. This modification is scheduled to be completed during our next refueling outage.

Very truly yours,

Robert E Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/PLP/ras

cc: Mr. J. P. O'Reilly, Region II
Harold F. Reis, Esquire

ENCLOSURE 2

Re: St. Lucie Unit 1
Docket No. 50-335
Loss of AC Power

Difference Between St. Lucie Unit 1 and Unit 2 as They Affect Station Blackout Analysis

(1) Unit 2 Atmospheric Dump Valves (ADVs) can be operated from the control room in order to maintain at least 10°F subcooling in the hot leg. Unit 1 ADVs will require local, manual operation (because of complete loss of AC) to maintain at least 10°F subcooling in the hot leg.

(2) The capacity of the Unit 1 ADVs is approximately one half that of the Unit 2 ADVs. The effect of this difference would be to require the Unit 1 ADVs to be opened for longer periods of time than indicated in the Unit 2 analysis.

(3) The Unit 2 analysis indicates that the Safety Injection Tanks (SITs) will release their contents to the RCS inventory when the RCS pressure has decayed to 583 psia (about 3.5 hours into the transient). The Unit 1 SITs are set to release at about 200 psig; therefore, credit cannot be taken for their increasing the RCS inventory and helping to maintain subcooling.

15C.4 STATION BLACKOUT ANALYSIS

The station blackout event is outside of the design basis for St. Lucie Unit 2. Nonetheless, an analysis was performed as requested by the NRC in response to the decision of ALAB-603. This analysis shows that St. Lucie Unit 2 can successfully endure a complete loss of ac power for at least 4 hours. However, it is expected that ac power would be restored within 30 minutes to one hour as a result of either one of the following corrective actions:

a) Offsite power is restored;

b) One of both of the St. Lucie Unit 2 diesel generators are started.

Operator action at 30 minutes is credited to open the atmospheric dump valves resulting in the closure of the main steam safety valves. Operator control of the atmospheric dump valves to assure natural circulation by maintaining subcooling in the RCS occurs after 2 hours.

The results of this analysis have shown that:

1) Natural circulation and core cooling can be maintained;

2) The reactor core remains in a subcritical condition;

3) There is no fuel failure; 7

4) The RCS coolant pressure remains within limits; and,

5) The resulting radiological doses are within limits.

Therefore, this analysis shows that St. Lucie Unit 2 can successfully endure station blackout event. Florida Power and Light will implement operator training and emergency procedures to ensure that plant operators would take appropriate actions to assure maintenance of natural circulation.

15C.4.1 IDENTIFICATION OF EVENT AND CAUSES

The Station Blackout event results from a loss of offsite power followed by failure of both standby diesel generators to start.

For Unit 2, this event results in a loss of all onsite ac power except that supplied by inverters from the two safeguards batteries. This provides power to the 120V ac (safeguards) instrument power and other required dc loads.

15C.4.2 SEQUENCE OF EVENTS AND SYSTEMS OPERATION

Table 15C.4-1 shows a chronological list of the timing of systems actions from the initiation of a station blackout event to the time that offsite power is restored (4 hours). A description of the sequence of events* is given below for each safety function:

*Those safety actions necessary to maintain the plant in hot shutdown.

Reactivity Control:

As a result of the loss of power to the reactor coolant pumps an automatic reactor trip signal is generated by the RPS on low reactor coolant system flow, as measured by steam generator delta-pressure (ΔP). The reactor trip signal interrupts power to the reactor trip switchgear which in turn releases the CEA's to drop into the core. The negative reactivity inserted by the CEAs is sufficient to maintain the core subcritical throughout the rest of the transient.

Reactor Heat Removal:

Following coastdown of the reactor coolant pumps, flow through the reactor is maintained by natural circulation. Heat is transferred to the secondary system through the steam generators.

Primary System Integrity:

A Power Operated Relief Valve (PORV) opens to limit the RCS pressure increase following turbine trip. Steam released from the PORV is contained in the quench tank. Letdown is isolated by the closing of the letdown control valve on loss of offsite power. Late in the transient, the Safety Injection Tanks provide borated water to the RCS increasing RCS inventory and helping to maintain subcooling in the hot leg.

Secondary System Integrity:

A turbine trip signal (TTS) is generated following the loss of offsite power and causes the turbine stop valves to close. The Main Steam Safety Valves (MSSVs) open to limit the pressure increase. 7

Auxiliary feedwater is automatically actuated on low steam generator level. Flow is provided by the turbine driven pump which derives all its control power from the station batteries. The operator opens the Atmospheric Dump Valves (ADVs) and regulates them from the control room to maintain steam generator pressure below the setpoint of the MSSVs and to reduce the primary system temperature to maintain subcooling in the hot leg.

Restoration of AC power:

Although the analysis which follows shows acceptable results assuming no ac power for 4 hours, in actuality ac power would be restored to the plant prior to this time (within 30 minutes to one hour) by either one of the following corrective actions.

a) Offsite power is restored and the onsite buses are manually connected to the startup transformers. Equipment is manually loaded on these buses, according to plant emergency procedures, or,

b) One (or both) Unit 2 diesel generators is started and safeguards loads are manually sequenced onto its 4.16 kV bus.

15C.4.3 ANALYSIS OF EFFECTS AND CONSEQUENCES

a). Mathematical Models

The NSSS response to a Station Blackout was simulated using the CESEC-III computer program.

b) Input Parameters and Initial Conditions

The initial conditions assumed for this event are contained in Table 15C.4-2. These conditions were chosen to provide the largest and most rapid depletion of RCS inventory and shutdown margin. The highest initial pressurizer pressure, least negative Doppler coefficient and most positive moderator temperature coefficient maximize the power and RCS pressure early in the transient resulting in inventory loss through the PORV. The major contributors to the RCS depressurization are the pressurizer heat losses and RCS leakage. Maximum values of these parameters were selected based on technical specifications, plant operating data and reactor coolant pump test results. The lowest initial pressurizer water volume minimizes the available RCS inventory. Initial core inlet temperature, core mass flow rate and pressurizer pressure have a negligible impact on the primary system depressurization. The evaluation of shutdown margin depletion was performed using the most negative moderator temperature coefficient and the least negative CEA worth for trip. This minimizes the shutdown margin remaining at the end of the transient. 7

The disposition of normally operating systems is given on Table 15C.4-3. The utilization of safety systems is given on Table 15C.4-4.

c) Results

The dynamic behavior of important NSSS parameters following a Station Blackout is presented in Figures 15C.4-1 to 15C.4-12. Table 15C.4-1 summarizes some of the important results of this event and the times at which the minimum and maximum parameter values discussed below occur. The loss of all ac electrical power initiates, among other things, a simultaneous loss of feedwater, loss of load, and loss of forced reactor coolant flow. As indicated in Figure 15C.4-1, the core power increases initially due to positive reactivity feedback and reaches a maximum value within a few seconds. Subsequent to loss of power to the reactor coolant pumps, the primary coolant flow decreases and a low flow reactor trip occurs as indicated in Table 15C.4-1. Reactor coolant flow vs. time is shown on Figure 15C.4-7. Subsequently, due to the insertion of large negative reactivity by the scram rods, the core power decreases very rapidly and approaches the decay heat value. Departure from nucleate boiling does not occur and therefore no fuel damage is predicted. See Figure 15C.4-8.

During the initial few seconds prior to reactor trip, the reduced steam generator heat rejection capability leads to a rapid increase in both the primary and secondary fluid temperatures. The volumetric expansion due to these increases in temperature produces sharp increases in primary and secondary pressures as well as an insurge of primary coolant into the pressurizer. The variations of the primary and secondary pressures are illustrated in Figures 15C.4-3, and 15C.4-9. The initial rapid increases in both pressures are terminated by the opening of the PORV and MSSVs. The primary relief valve closes rapidly, as the primary system pressure decreases below the setpoint value within a few seconds after opening of the valve. The secondary safety valves cycle open and closed until the operator opens the atmospheric dump valves. MSSV and ADV flow vs. time are shown on Figures 15C.4-11 and 15C.4-12, respectively.

The steam generator liquid level decreases during the transient and reaches a minimum value after auxiliary feedwater flow is automatically actuated using the steam-driven auxiliary feedwater pump. Steam generator level increases until normal water level is reached. The operator subsequently controls auxiliary feedwater to maintain normal level. See Figure 15C.4-6.

The RCS pressure and temperature gradually decrease at fairly constant rates in the long term as a result of pressurizer heat loss, RCS leakage, low heat transfer rates at the steam generators, and the operator manually reducing secondary side pressure. Since the RCS pressure decreases at a higher rate than the RCS temperature, the pressure approaches the saturation pressure.

Saturation occurs in the reactor vessel head. Continued primary pressure drop without a significant decrease in primary temperatures would result in saturated conditions in the hot leg. Credit is taken for operator action to maintain at least 10 F subcooling in the hot leg. This is accomplished by further opening the atmospheric dump valves to reduce the secondary system pressure and temperature. The increased heat removal in the steam generators caused by the larger ΔT across the steam generator tubes reduces the primary system temperatures. Voiding is restricted to the vessel head and natural circulation is not adversely impacted for more than 4 hours. 7

The Safety Injection Tanks (SITs) provide borated water to the RCS after RCS pressure is reduced below their discharge pressure. No credit is taken for the negative reactivity added as a result of this discharge.

At 4 hours, sufficient ac power is assumed to be restored to provide power to the charging pumps and pressurizer heaters. These will be used to pressurize the RCS and to continue hot leg subcooling.

Operability of the turbine driven auxiliary feedwater pump requires at least 50 psia secondary pressure. At 4 hours after the initiation of the event, the secondary pressure will be greater than 300 psia. Less than 100,000 gallons of auxiliary feedwater are used during the event. The condensate storage tank capacity is greater than 300,000 gallons.

15C.4.4 CONCLUSIONS

The maximum RCS pressure is 2541 psia (including reactor coolant pump and elevation heads). This is well below 110 percent of design pressure.

Natural circulation is maintained for at least the 4 hour period that offsite ac power and diesel generator power are assumed unavailable. During this time voids are restricted to the reactor vessel head and subcooling is maintained in the hot leg. 7

The radiological release due to a Station Blackout results in no more than a 0.4 rem 4 hour inhalation thyroid dose at the exclusion area boundary.

The average RCS temperature at 4 hours is above 430 F. This is above the temperature at which the shutdown margin would be depleted. Therefore, the core remains subcritical following reactor trip for the duration of the event.

No fuel damage occurs during this event.

TABLE 15C.4-1

SEQUENCE OF EVENTS, CORRESPONDING TIMES AND SUMMARY OF RESULTS FOR THE STATION BLACKOUT EVENT

Time (Sec)	Event	Setpoint or Value
0.0	Loss of all on- and offsite AC power	---
1.5	Low Primary Coolant Flow Reactor Trip, %	93
2.0	Auxiliary Feedwater Actuation Signal, % of Narrow Range Tap Span	5
2.4	Power Operated Relief Valve Opens, psig	2385
2.6	Maximum Core Power, %	104.8
5.5	Maximum RCS pressure, psia	2541
6.0	Maximum pressurizer pressure, psia	2460
6.3	Main Steam Safety Valves Open, psig	995
8.5	1. Power Operated Relief Valve Closes, psig	2361
	2. Total Primary Relief Valve Release, lbm	554
12.2	Maximum Secondary System Pressure, psia	1038
182.0	Auxiliary Feedwater reaches Steam Generators, gpm	500

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TABLE 15C.4-1 (Cont'd)

Time (Sec)	Event	Setpoint or Value	
1800.0	1. Operator Opens and Controls Atmospheric Dump Valves, psia	900	
	2. Main Steam Safety Valves close, psig	945	7
	3. Total Main Steam Safety Valve Release, lbm	116630	
2258.0	Voiding Occurs in Reactor Vessel Head	- - -	
8600.0	Operator Begins to Reduce Steam Generator Pressure to Maintain Hot Leg Subcooling	- - -	
11785.0	Main Steam Isolation Valves close, psig	435.0	
12540.0	Safety Injection Tanks actuated, psia	583.0	
14400.0	1. Operator Restores AC Power	- - -	
	2. Total Atmospheric Dump Valve Release, lbm	363300.0	

TABLE 15C.4-2

ASSUMED INITIAL CONDITIONS FOR STATION BLACKOUT ANALYSIS

PARAMETER	ASSUMED VALUE
Initial Core Power Level, MWt	2630
Core Inlet Coolant Temperature, ^{o}F	551
Core Mass Flow Rate, 10^6 lbm/hr	133.9
Pressurizer Pressure, psia	2350
Initial Pressurizer Water Volume, % Level	40
Steam Generator Water Level, % of Narrow Range Tap Span	70
Doppler Coefficient Multiplier	0.85
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho/^{o}F$	
To determine initial power transient, 0-10 seconds	+0.4
To determine degree of shutdown margin depletion	-2.7
CEA Worth for Trip, 10^{-2} $\Delta\rho$	6.68
Pressurizer Heat Loss, 10^6 BTU/hr	0.546
Primary Coolant Leakage, gpm:	16
Identified Leakage, gpm	
a) Technical Specification Steam Generator Tube Leakage	1
b) Primary Safety Valve Leakage	3
c) Other Identified Leakage	6
Unidentified Leakage	1
RCP Controlled Bleedoff	4
RCP Seal Leakage	1
	16

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DISPOSITION OF NORMALLY OPERATING SYSTEM FOR STATION BLACKOUT

System	Normal Automatic Mode Throughout Transient	Manual Mode Throughout Transient	Normal Automatic Mode Inoperative on Loss of AC	Manual Mode Inoperative on Loss of AC	Failure Assumed Within System	Associated Notes
1. Main Feedwater System			X			
2. Turbine-Generator Control System	X					
3. Steam Bypass Control System				X		
4. Pressurizer Pressure Control System				X		2
5. Pressurizer Level Control System				X		
6. Control Element Drive Mechanism Control System	X					
7. Reactor Regulating System				X		
8. Reactor Coolant Pumps				X		1
9. Chemical and Volume Control System				X		2
10. Condenser Evacuation System				X		
11. Turbine Gland Sealing System				X		
12. Component Cooling Water System						2
13. Turbine Cooling Water System				X		
14. Intake Cooling Water System						2
15. Condensate Transfer System				X		
16. Circulating Water System				X		
17. Spent Fuel Pool Cooling System				X		2
18. AC Power (Nonsafety)			X			
19. AC Power (Safety)					X	3
20. DC Power		X				
21. Power-Operated Relief Valves	X					
22. Instrument Air System			X			2
23. Waste Management-Liquid				X		

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NOTES:
1. RCP bleedoff is not isolated during this event.
2. Portions of these systems, powered by the safety bus on loss of AC, are not available due to the failure of both diesel generators.
3. Only the AC power supplied through the inverters is available.

TABLE 15C.4-4

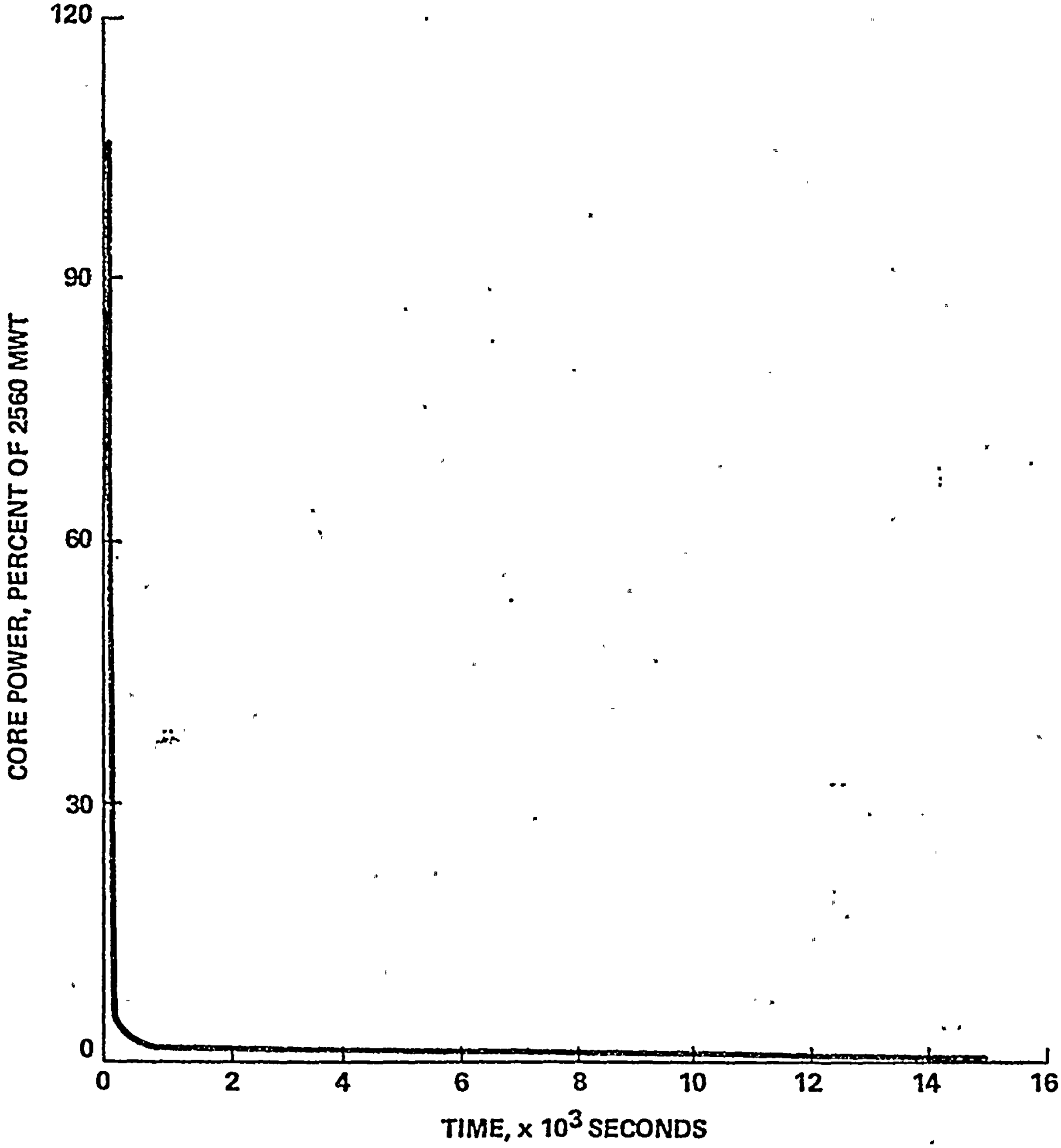
UTILIZATION OF SAFETY SYSTEMS FOR STATION BLACKOUT

	Actuated and Required	Actuated But Not Required	Safety Grade Backup to Nonsafety Grade System	Failure Assumed Within System (See Notes)	Associated Notes
1. Reactor Protection System	X				
2. Engineered Safety Features Actuation Systems					2
3. Diesel Generators and Support Systems				X	1
4. Reactor Trip Switchgear	X				
5. Main Steam Safety Valves	X				
6. Pressurizer Safety Valves			X		
7. Main Steam Isolation Valves		X			2
8. Main Feedwater Isolation Valves			X		
9. Auxiliary Feedwater System	X				2,4
10. Safety Injection System		X			3
11. Shutdown Cooling System (CCW & ICW)					
12. Atmospheric Dump Valve System	X				5
13. Containment Isolation System		X			6
14. Containment Spray System					
15. Iodine Removal System					
16. Containment Combustible Gas Control System					
17. Containment Cooling System					

NOTES:
1. Both diesel generators fail for this event.
2. Only those portions powered from the safeguard batteries are available.
3. Safety Injection Tanks are available.
4. Auxiliary Feedwater is automatically actuated. Only the turbine driven pump is available.
5. ADVs can be manually operated from the control room.
6. Portions of this system are actuated on loss of instrument air.

Systems not checked are not utilized during this event.

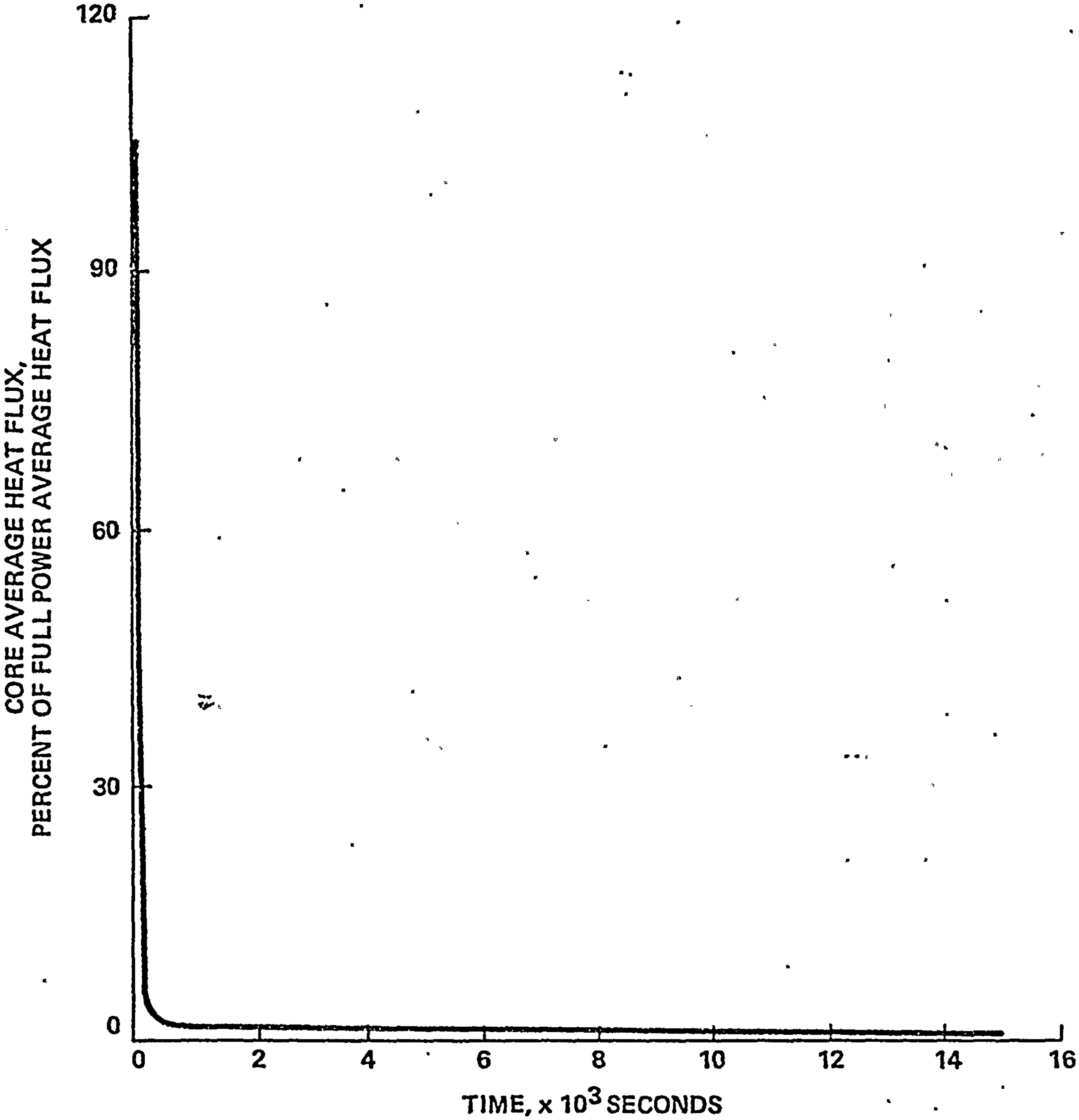
Amendment No 7, (10/81)



AMENDMENT NO. 7, (10/81)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CORE POWER VS TIME
FIGURE 15C.4-1



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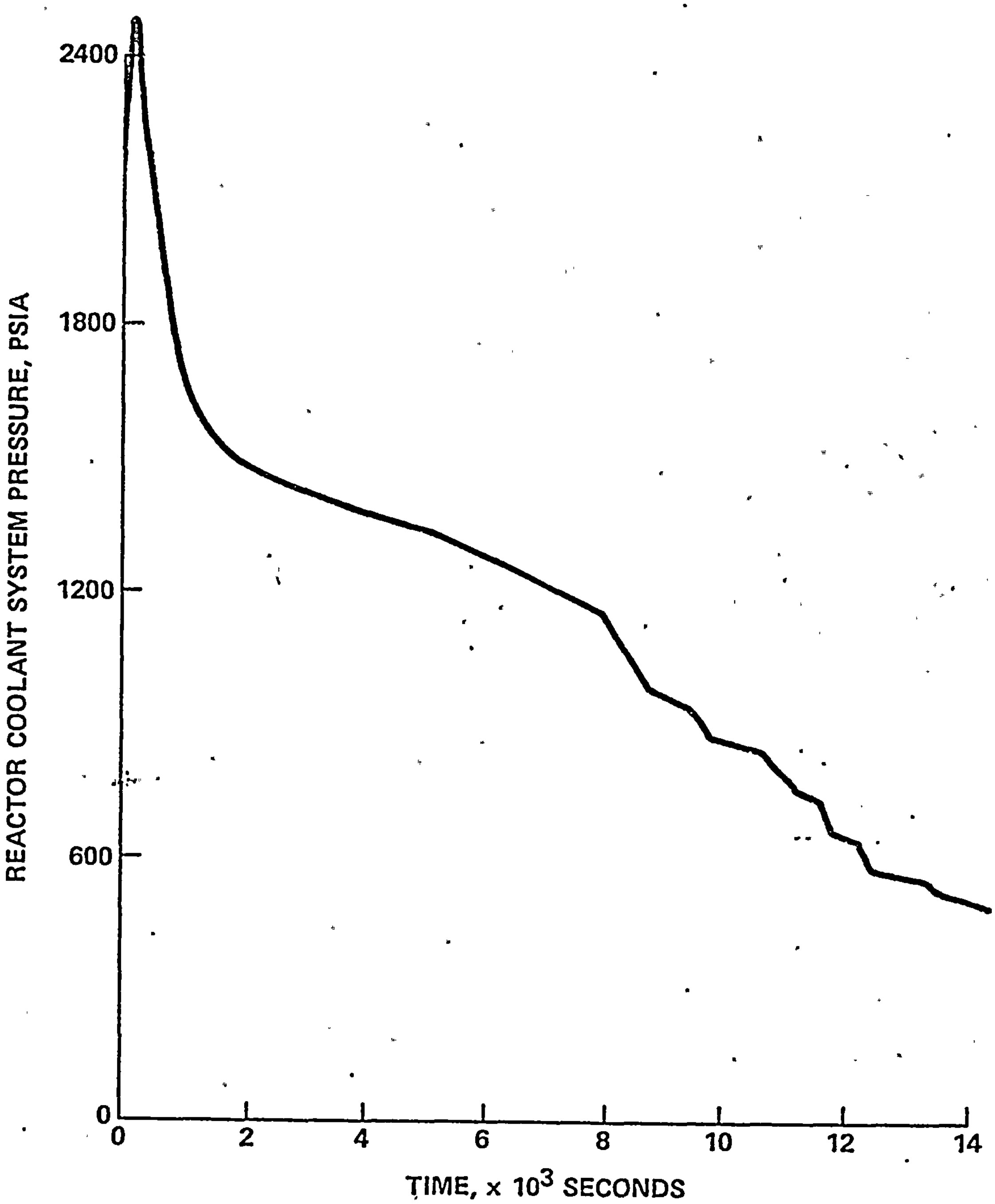
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 2
CORE AVG. HEAT FLUX VS TIME FIGURE 15C.4-2



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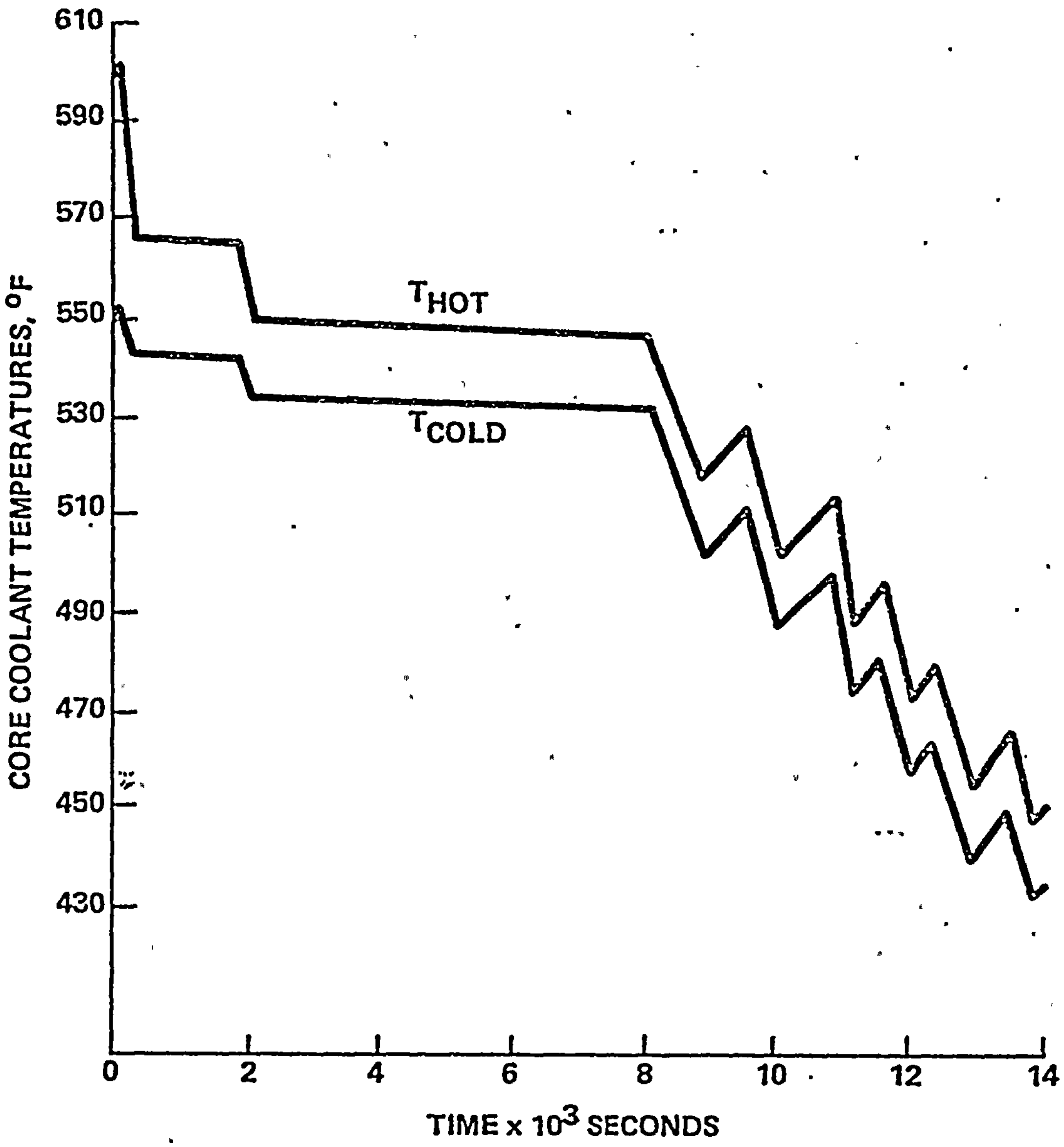
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT SYSTEM
PRESSURE VS. TIME
FIGURE 15C.4-3



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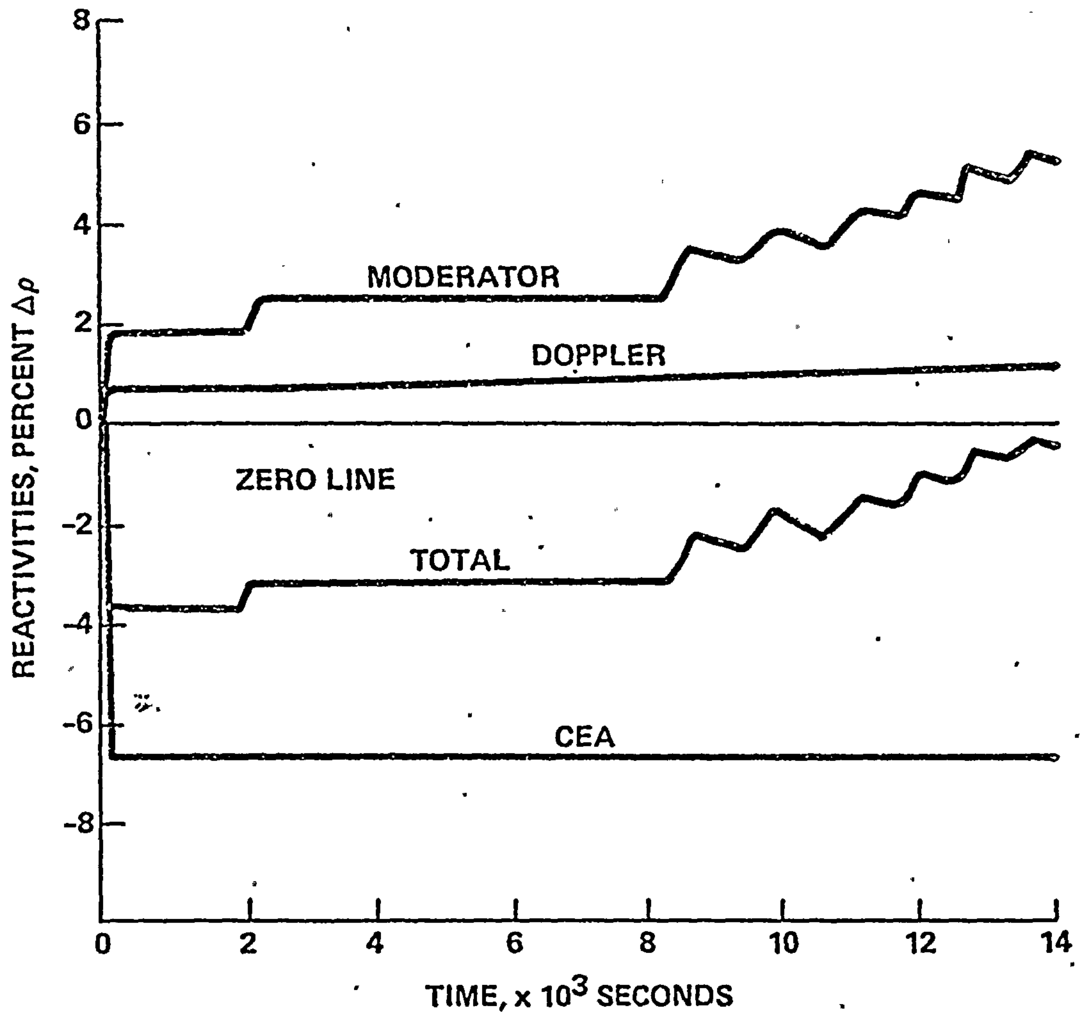
ST. LUCIE PLANT UNIT 2

CORE COOLANT TEMPS VS TIME
FIGURE 15C.4-4



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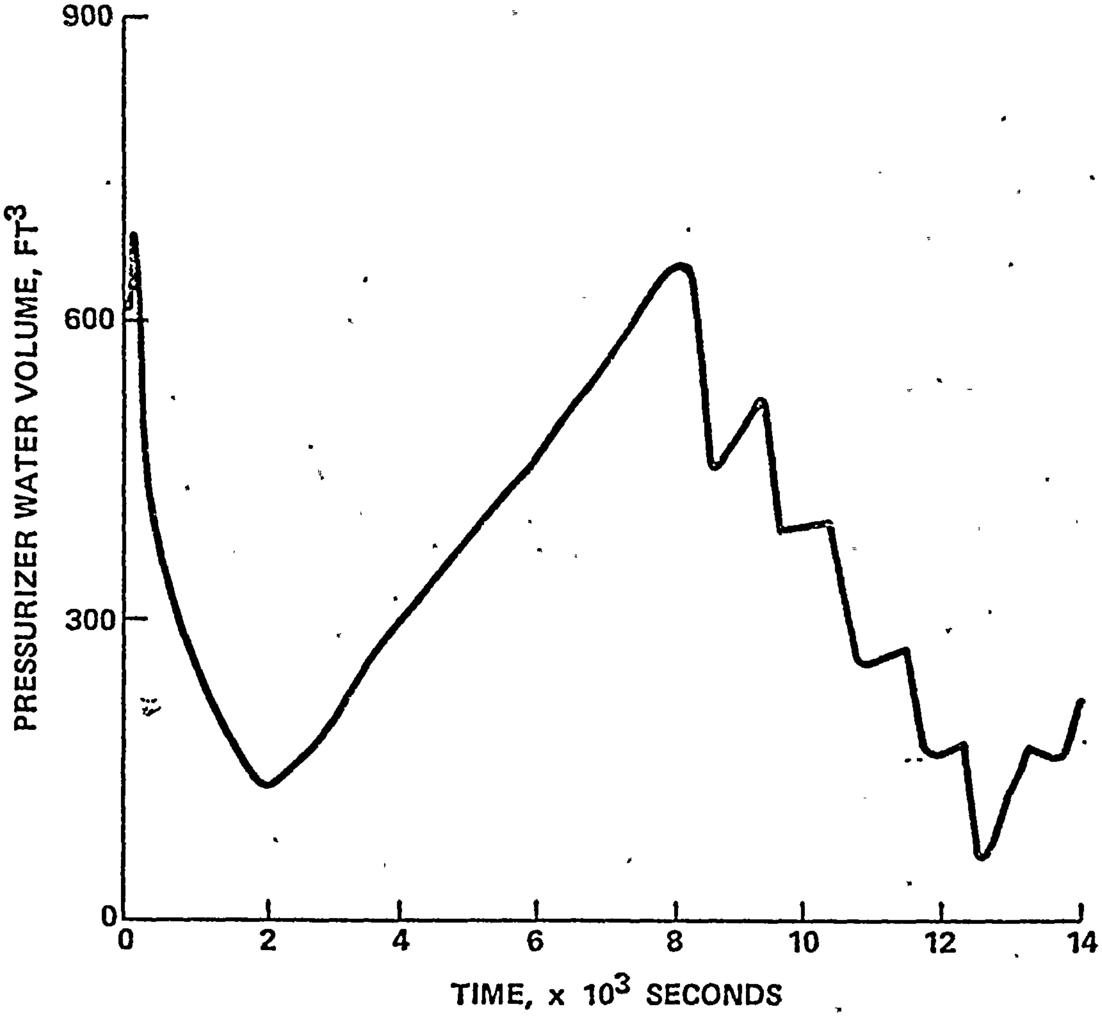
ST. LUCIE PLANT UNIT 2

REACTIVITIES VS. TIME
FIGURE 15C.4-5



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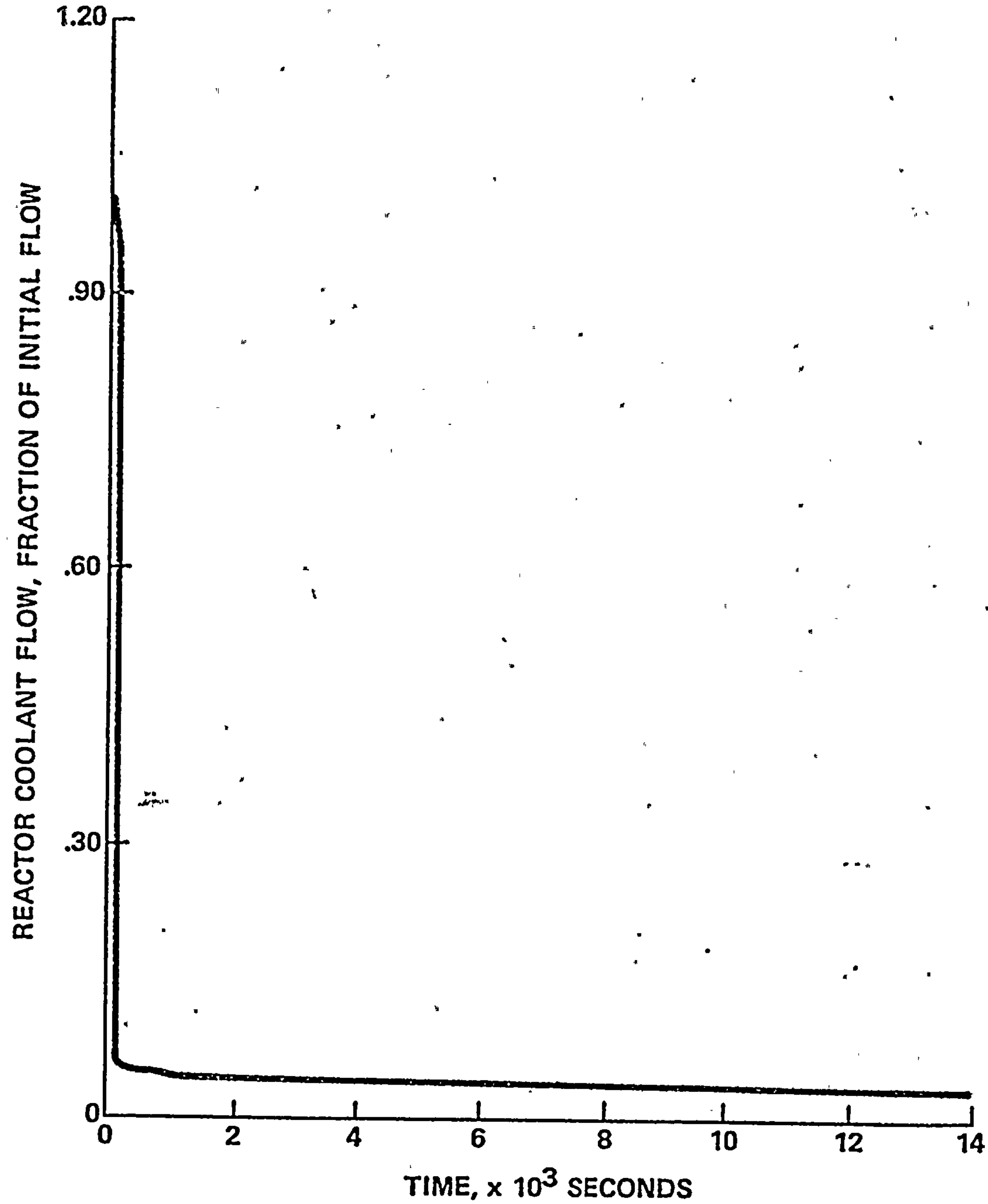
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 2
PRESSURIZER WATER VOL. VS. TIME FIGURE 15C.4–6



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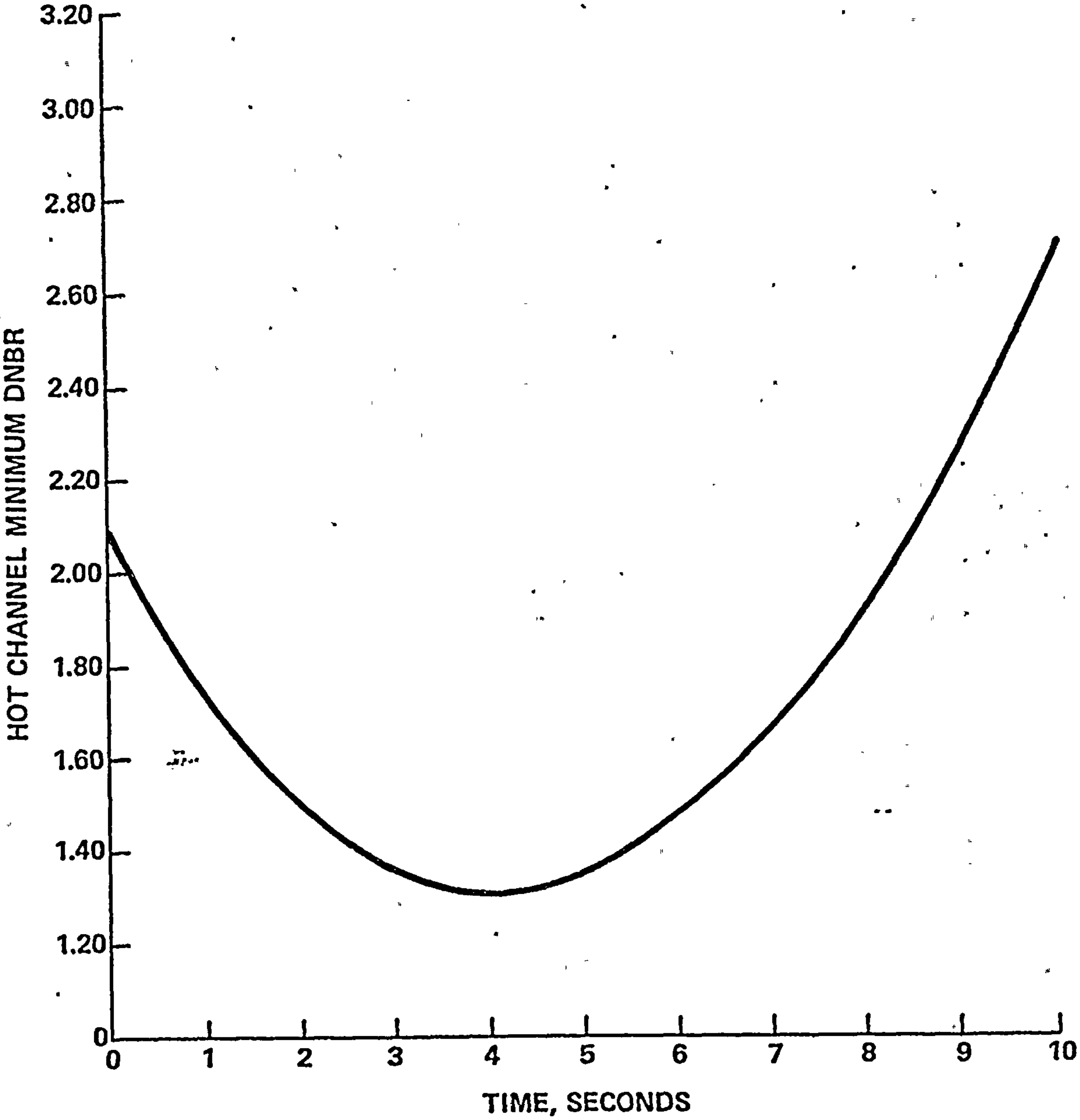
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT FLOW VS. TIME
FIGURE 15C.4-7



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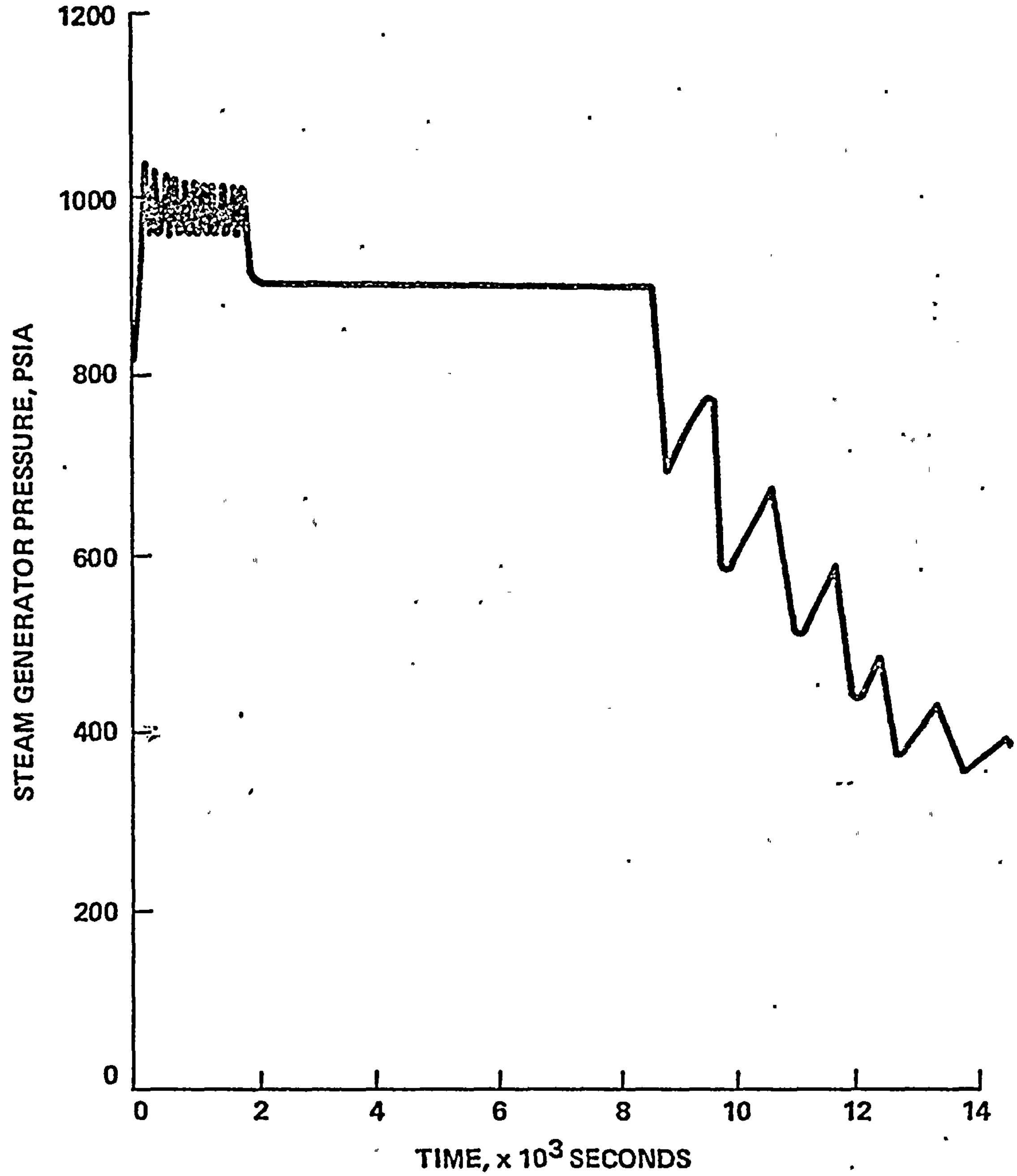
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HOT CHANNEL MINIMUM DNBR VS. TIME
FIGURE 15C.4-8



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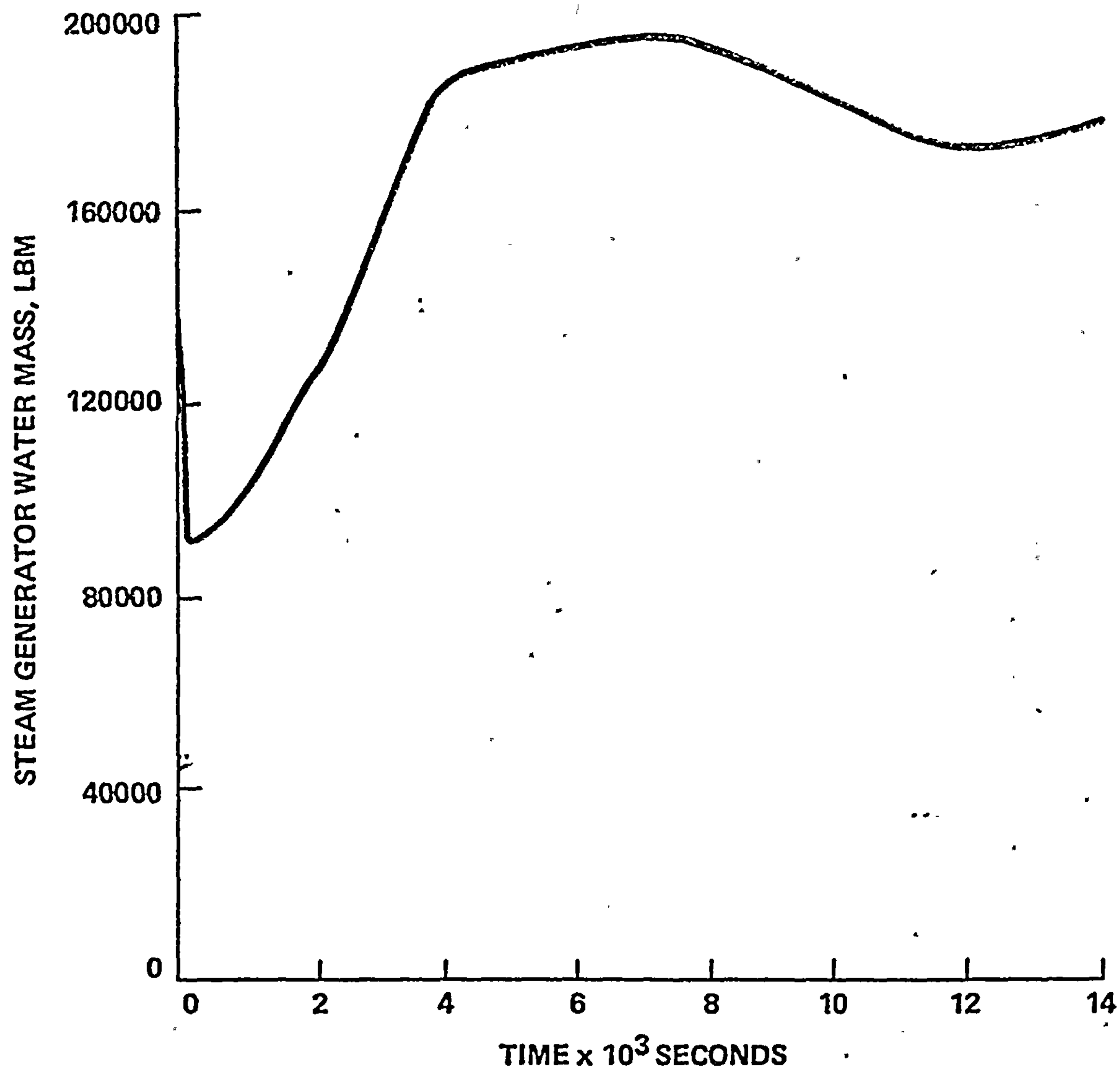
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 2
STEAM GENERATOR PRESSURE VS TIME FIGURE 15C.4–9



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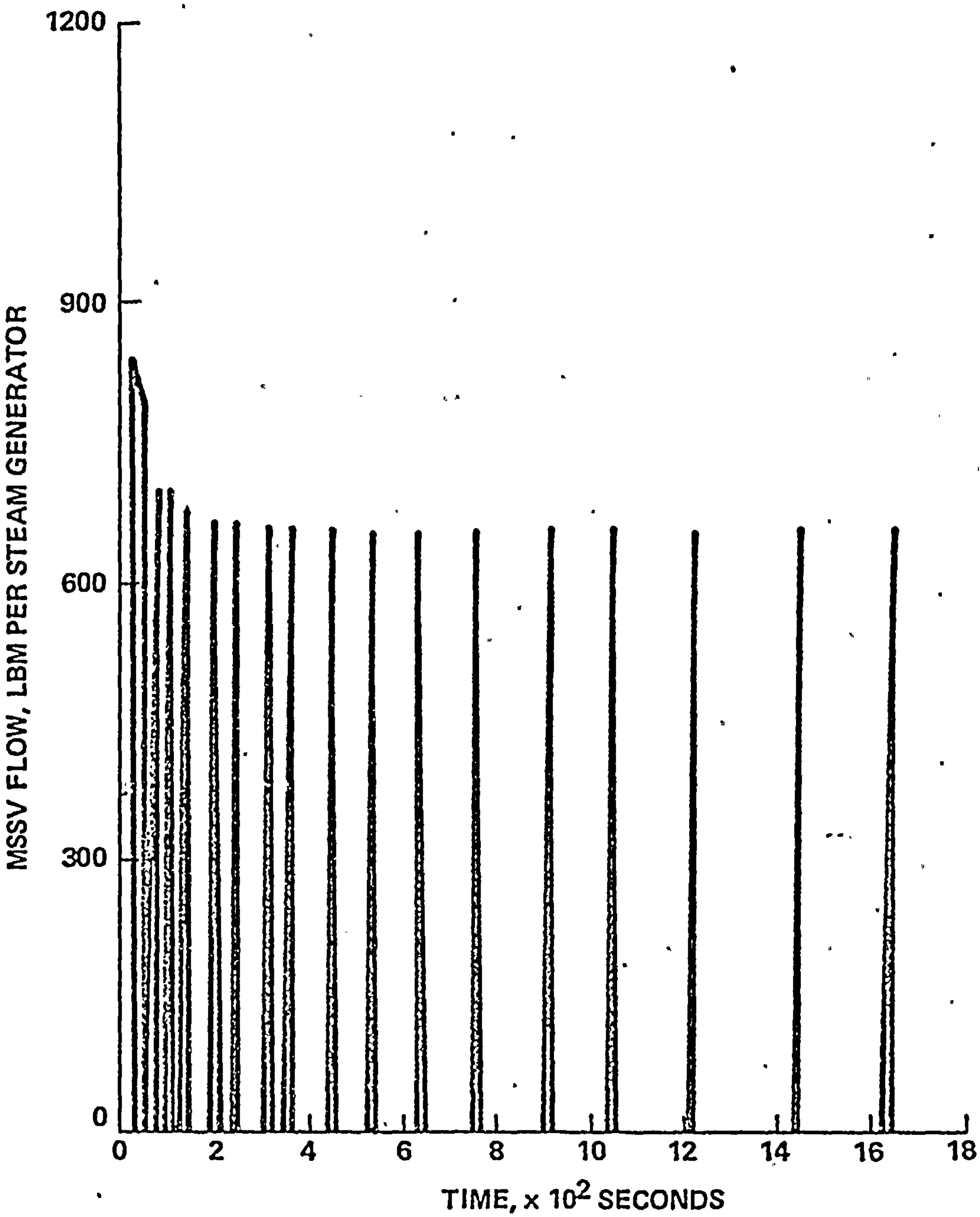
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 2
STEAM GEN. WATER MASS VS TIME FIGURE 15C.4-10



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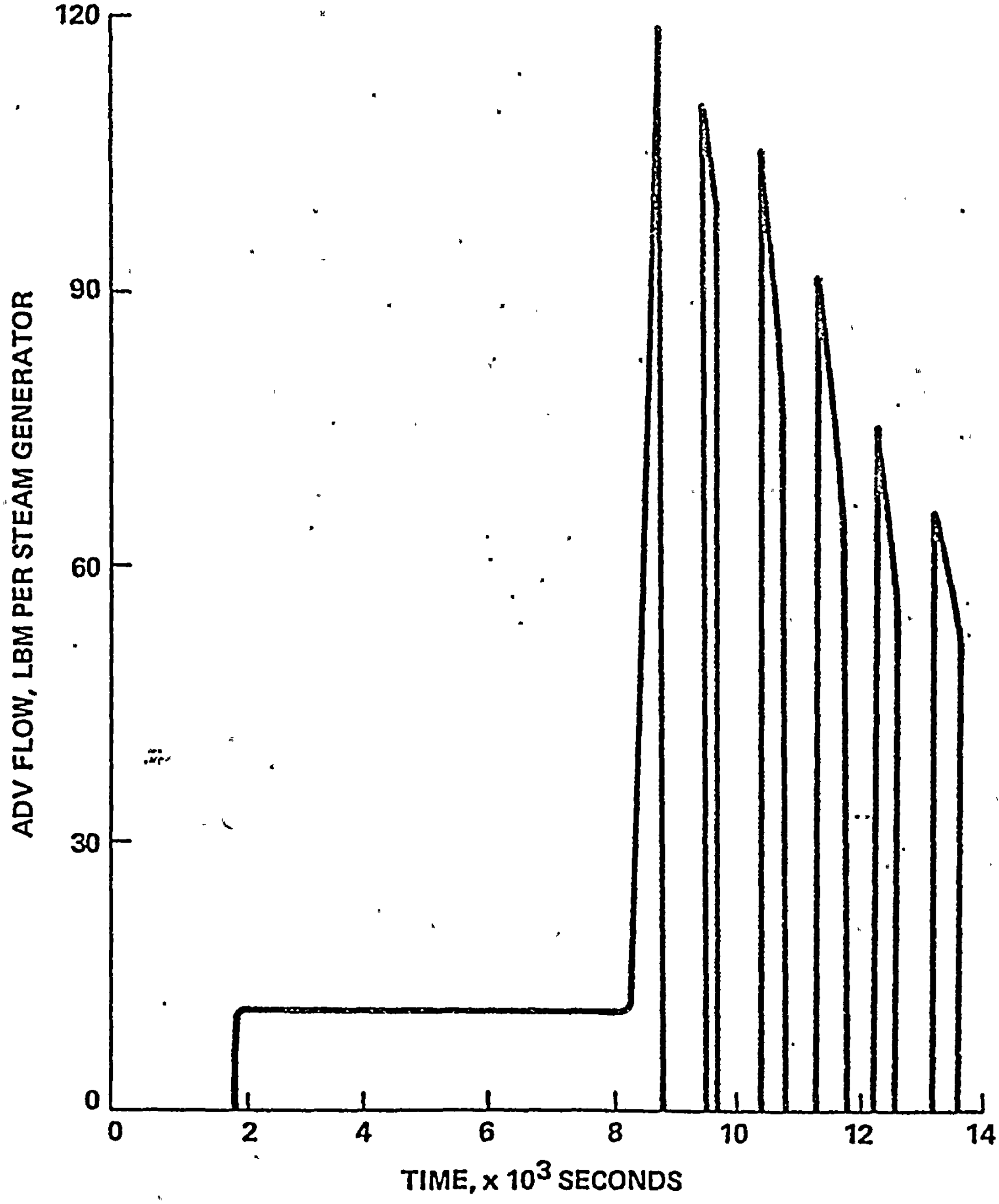
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 2
MSSV FLOW VS. TIME FIGURE 15C.4-11



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FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ADV FLOW VS. TIME
FIGURE 15C.4–12

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is Vice President of Florida Power & Light Company, the herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said

Robert E. Uhrig

Robert E. Uhrig

Subscribed and sworn to before me this

18 day of December, 19 81

Cheryl L. Fredrick

NOTARY PUBLIC, in and for the County of Dade, State of Florida

My commission expires: Notary Public, State of Florida at Large
My Commission Expires October 30, 1983
~~Bonded thru Maynard Bonding Agency~~