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CLARK,R.A.	Operating Reactors Branch 3

SUBJECT: Forwards corrected safety assessment as result of thermal shield removal,submitted w/830623 ltr.

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FLORIDA POWER & LIGHT COMPANY

July 27, 1983
L-83-429

Office of Nuclear Reactor Regulation
Attention: Mr. Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Clark:

Re: St. Lucie Unit I
Docket No. 50-335
Reactor Vessel Internals and Thermal Shield;
Plant Recovery Program

In our letter of April 19, 1983 (L-83-230), Florida Power & Light Company committed to provide a safety assessment as a result of thermal shield removal. This assessment was subsequently transmitted to you on June 23, 1983 (L-83-369).

It has been determined that our June 23rd letter contained three minor errors in the text, that do not change the conclusions made therein. These changes are indicated in Attachment I of this letter. This attachment should replace the attachment of our June 23rd letter in its entirety.

No other changes have been made to our original submittal of June 23, 1983.

Very truly yours,

J a De Mastry
for

Robert E. Uhrig
Vice President
Advanced Systems and Technology

REU/DAC/cab

cc: Harold F. Reis, Esquire

Enclosure

A001
1/1

ASSESSMENT OF SAFETY IMPACTS

OF REMOVAL OF THERMAL

SHIELD FROM

ST. LUCIE UNIT 1

SUMMARY

The following attachments provide an assessment of the safety impacts of removing the thermal shield from St. Lucie Unit 1. Specifically, Attachment 1 addresses the effect on transients, ex-core detector decalibration, hydraulic liftoff forces and core physics; Attachment 2 presents the results of a reanalysis of the Loss of Coolant Accident (LOCA) without the thermal shield performed by Exxon.

In addition to the material contained herein, both Exxon and Combustion Engineering have performed a review of current plant Technical Specifications. Both have concluded that no fuel related technical specifications need be changed in support of thermal shield removal.

Based on the material presented, no major impact to safety due to thermal shield removal is seen.

ATTACHMENT 1

IMPACT OF REMOVAL OF THERMAL SHIELD ON TRANSIENTS

The removal of the thermal shield in St. Lucie Unit 1 will result in an increased flow (about 1/2%) and an increase in downcomer liquid volume (about 20%). The increased flow will result in slightly more rapid transients in the primary loop. In particular the excess load, the loss-of-load and the steam line break will tend to respond more rapidly (by less than 1/2 second) at design flows. However, the increased volume delays cold leg transients by about 1/3 second. The net impact on transient timing will not be greater than a few tenths of a second at design flows, nor a few seconds at natural circulation flows. Hence no significant impact could be expected for temperature transients.

Flow transients are conservatively estimated in transient analysis. Any impact on flow coastdown is undetectable. Even the most rapid flow transient, the seized rotor, should not be noticeably affected by the increased flow. The increased volume serves to increase the stored energy in the fluid and offsets, to some extent, the tendency of the pump to coast down more rapidly from higher flows.

The major effect of increased flow, for which there are no offsetting effects, is the increase in MDNBR caused by the increased flow. Thus, the net effect on transient analyses of removal of the thermal shield should be negligible and probably beneficial.

EX-CORE DECALIBRATION DUE TO REMOVAL OF THE THERMAL SHIELD

The ex-core neutron flux detectors are distal to the thermal shield and the downcomer. Absolute calibration of the detectors is based on the in-core monitoring system and depends on the amount of attenuation present between the core and the detectors. Decalibration of the ex-core detectors during large system cooldowns is an expected phenomenon due to the density increase associated with the cooldown. The current safety analysis addressed this effect with the thermal shield present. Without the thermal shield, the larger volume of water between the core and the ex-core detectors will show a slightly larger decalibration effect. This impact of the removal of the thermal shield could produce a minor penalty on the analyses in that it would reduce the pressure margin in the TM/LP trip by 10 PSI or so and the margin in the LPD LSSS by 0.6% for the worst cooldown events. Neither of these values are significant when compared to the available margin.

HYDRAULIC LIFTOFF

Increased flow through the reactor core will result in increased levitational forces on the bundles. Based on a quadratic flow dependence for pressure losses, it is anticipated that the lift-off forces will be about 1% greater with 1/2% more flow. For a dynamic pressure drop of approximately 40 PSID, which conservatively bounds the actual pressure drop, the increased levitation forces on a 9"X9" assembly would be 32 pounds. Since the margin with the thermal shield exceeds 400 lbs, this increase represents a loss of about 8% of the margin and thus lift-off is not a concern.

CORE PHYSICS

The impact of the thermal shield removal on the core physics is negligible. To quantitatively assess the impact of the thermal shield removal upon the core reactivity and upon the core power distribution a two dimensional pin by pin PDQ calculation was made with and without the thermal shield. The core reactivity and core power distribution were identical between the two calculations.

performed with bounding ENC 14 X 14 fuel parameters and power history corresponding to: beginning-of-life stored energy, end-of-life fission gas release, and end-of-life actinide decay power. The fuel design parameters are shown in Table 1. Calculated event times for this break are given in Table 2, and final temperature and metal-water reaction results are shown in Table 3. Results from the previous analysis are compared in Table 3 with those calculations in this study. The system nodalization is identical to that used in the previous analysis and is shown in Figure 1.

The result of the analysis show a decrease of 59°F in the Peak Cladding Temperature (PCT) as compared to previous case. The reduction in PCT is due primarly to improved heat transfer calculated during the blowdown phase of the analysis. The improved heat transfer results from reduced coolant temperatures in the new analysis from about 10 seconds to the end of blowdown. The new calculated containment pressure is 0.35 PSIA higher than the previous case for the first 110 seconds of the transients. The increase in the containment pressure is caused by an increase in the mass flow to the containment building. The increased containment pressure results in reduced steam binding which improves reflood rates. The larger downcomer volume produced by the removal of the thermal shield, slightly retards the reflood rate during the early portion of the reflood transient due to reduced downcomer liquid head. Sufficient water however exists to fill the downcomer, within 10 seconds after the start of reflood. The improved blowdown heat transfer and higher containment pressure more than offset the effect on PCT that reduced downcomer liquid heads have on reflood rates during the early portion of the reflood transient.

System blowdown results for the 0.4 DECLG break are given in Figures 2-10, hot channel results in Figures 11-14, containment pressure and normalized power results in Figures 15-16, reflood results in Figures 17-19, and TOODEE2 heatup result in Figure 20.

The calculated peak cladding temperature for the 0.4 DECLG break is 2000^{o}F with less than 3.3% maximum local metal-water reaction, and less that 1% core-wide metal-water reaction. These results show that for St. Lucie Unit 1 operating at 2700 MWth and maximum LHGR of 15.30 KW/ft, including 2% power uncertainty, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46 with the thermal shield removed.

Table 1 CE 2x4 PWR Data

Primary Heat Output, MWt	2700*
Primary Coolant Flow, lbm/hr	1.394×10^8
Primary Coolant Volume, ft^3	19,214**
Operating Pressure, psia	2250
Inlet Coolant Temperature, ^oF	549
Reactor Vessel Volume, ft^3	4402
Pressurizer Volume, Total, ft^3	1500
Pressurizer Volume, Liquid, ft^3	800
Accumulator Volume, Total, ft^3 (one of four)	2020
Accumulator Volume, Liquid, ft^3	1090
Accumulator Pressure, psia	230
Steam Generator Heat Transfer Area, ft^2 (one of two)	74,722
Steam Generator Secondary Flow, lbm/hr	5.899×10^6
Steam Generator Secondary Pressure, psia	885
Reactor Coolant Pump Head, ft	280
Reactor Coolant Pump Speed, rpm	886
Moment of Inertia, lbm-ft^2/rad	101,900
Cold Leg Pipe, I.D., in	30
Hot Leg Pipe, I.D., in	42
Pump Suction Pipe, I.D., in	30

* Primary Heat Output used in RELAP4-EM Model - 1.02 x 2700 = 2754 MWt.
**Includes total accumulator and pressurizer volume.

Table 1 (Continued)

Fuel Assembly Rod Diameter, in*	.440
Fuel Assembly Rod Pitch, in*	.580
Fuel Assembly Pitch, in*	8.180
Fueled (Core) Height, in*	136.7
Fuel Heat Transfer Area, ft^2	50,117
Fuel Total Flow Area, ft^2	53.19
Steam Generator Tube Plugging (Assumed Uniform)	5%

* ENC Fuel Parameters

Table 2 Large Break Results Time Sequence of Events

Event	Time of Event (seconds) DECLG ($C_D = 0.4$)
Start	0.0
Initiate Break	0.05
Safety Injection Signal	0.92
Pressurizer Empties	8.8
Accumulator Injection, Broken Loop	20.98
Accumulator Injection, Single Intact Loop	22.95
Accumulator Injection, Double Intact Loop	22.95
End-of-Bypass	27.81
Safety Injection Flow, SIS	30.96
Start of Reflood	46.05
Accumulators Empty, Single Intact Loop	73.81
Accumulators Empty, Double Intact Loop	74.11
Peak Clad Temperature is Reached	165.0

Table 3 Analysis Results for 0.4 DECLG Break

Analysis Results	w/Shield[1] DECLG (C_D0.4)	w/o Shield DECLG (C_D=0.4)
Peak Clad Temperature, °F	2059	2000
Peak Clad Temperature Location, ft from bottom	9.22	9.47
Local Zr/H_2O·Reaction (max), % (at 395 sec)	4.0	3.3
Local Zr/H_2O Location, ft from bottom	8.7	9.22
Total H_2 Generation, % of Total Zr Reacted	<1.0	<1.0
Hot Rod Burst Time, sec	39.8	41.41
Hot Rod Burst Location, ft from bottom	8.0	7.47
Peak Linear Heat Generation Rate, BOCREC, kW/ft	.729	.728
Analysis Input		
License Core Power MWt	2700	
Power Use for Analysis, MWt	2754	
Peak Core Linear Power kW/ft	15.00*	
Total Allowable Peaking Factor	2.42	

* Does not include 2% power uncertainty.

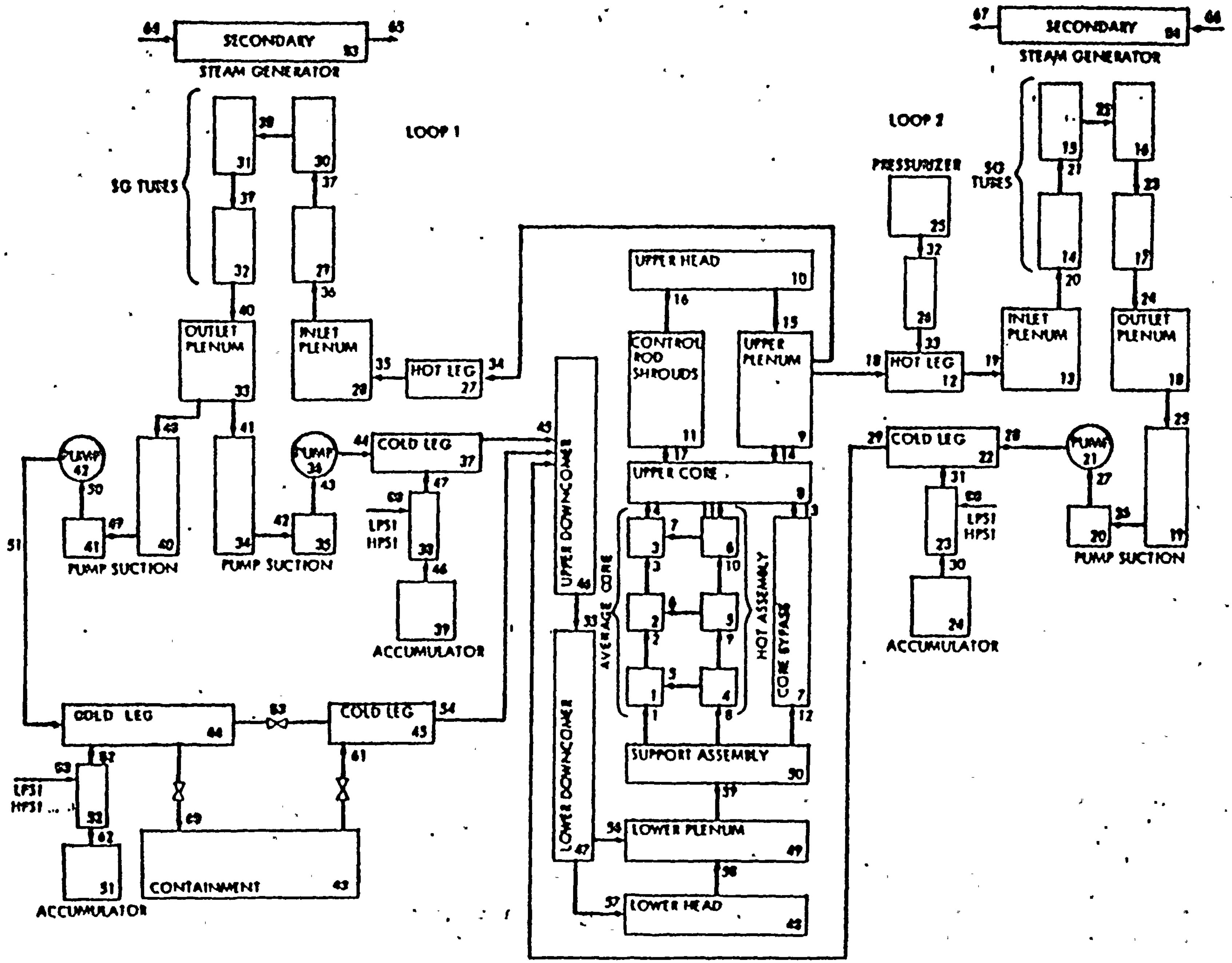

Figure 1 RELAP4-EM Blowdown System Nodalization For St. Lucie Unit 1

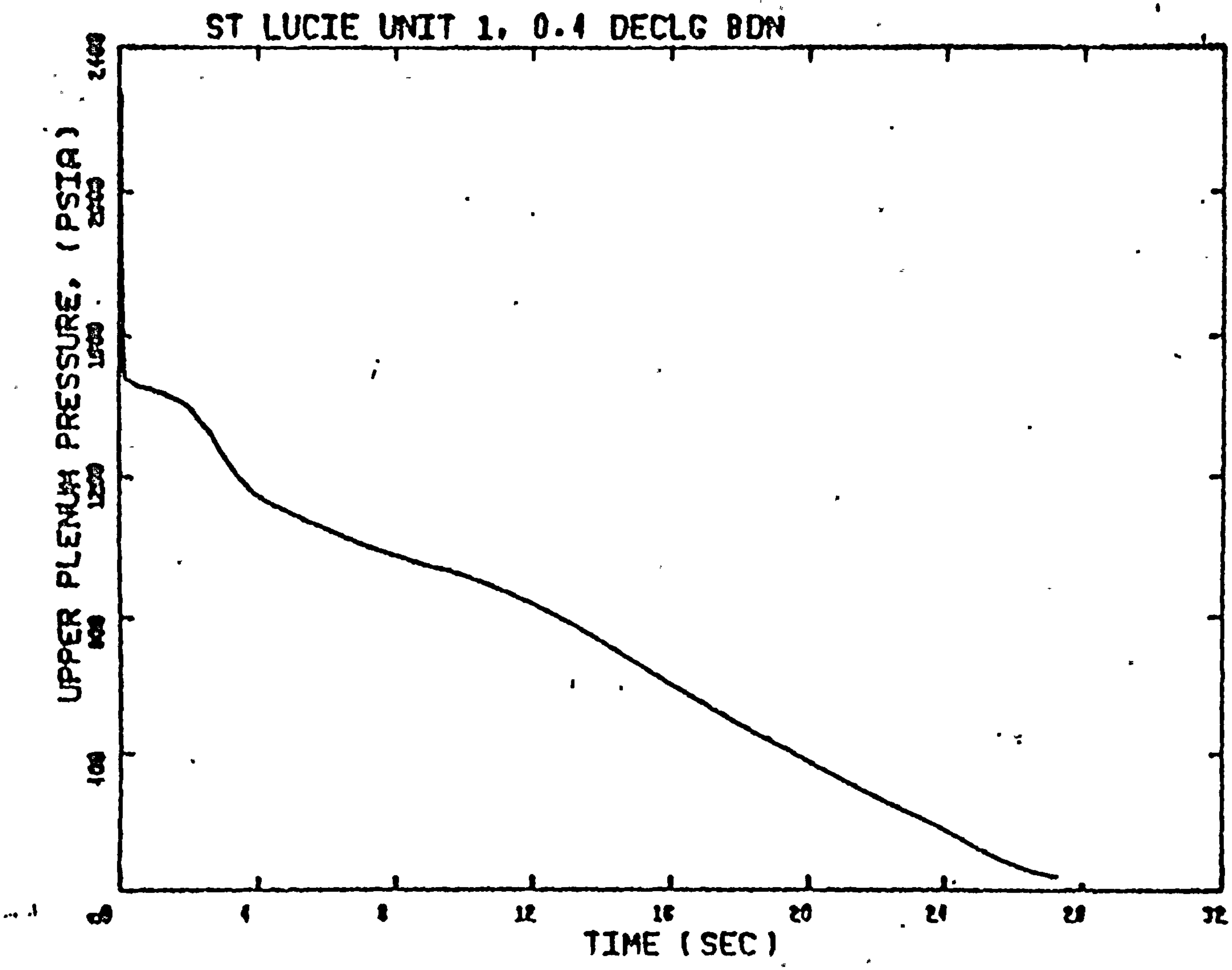

Figure 2 Blowdown System Pressure, 0.4 DECLG Break

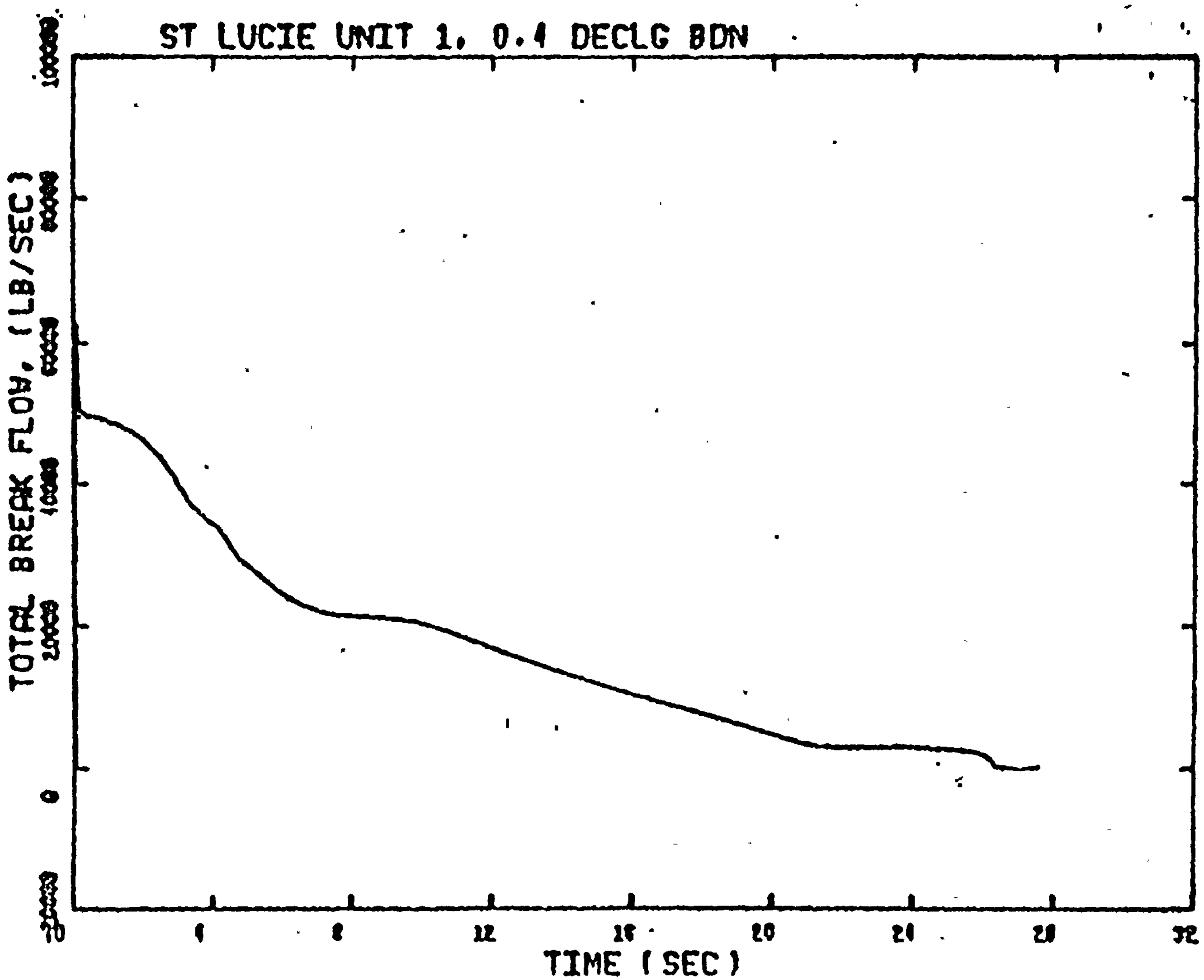

Figure 3 Blowdown Total Break Junction Flow Rate, 0.4 DECLG Break

ST. LUCIE UNIT 1 LOCA-ECCS ANALYSIS
WITH THERMAL SHIELD REMOVED

Prepared by

T. Tahvili

This Attachment presents results of a LOCA-ECCS analysis performed for the St. Lucie Unit 1 Nuclear Reactor utilizing Exxon Nuclear Company (ENC) fuel. This analysis differs from previous ENC analyses[1] in that the thermal shield was removed from the Reactor Vessel. The analysis was performed using the current NRC approved ENC evaluation models[2,3,4,5,6]. The results of this analysis show that the St. Lucie Unit 1 reactor can operate in conformance with 10 CFR 50.46 Appendix K criteria[7] with an allowed linear heat generation rate, including 2% for power uncertainty, of 15.30 KW/ft. The total allowable peaking (F_Q^T) remains at the value justified in the previous analysis of 2.42.

The methods used to perform the LOCA-ECCS analysis with the thermal shield removed are the same as those used in the previous analysis[1]. The analysis assumed no increase in reactor vessel flow from the value established in previous ENC analysis. Revisions were made to the input to reflect changes in the lower downcomer flow area, volume, and heat transfer area resulting from the thermal shield removal. Pressure distributions were revised to re-establish initial steady-state conditions.

Calculations were made for the limiting break identified in the previous analysis[1], i.e., the double-ended cold leg guillotine break with a discharge coefficient of 0.4 (0.4 DECLG). The analysis was

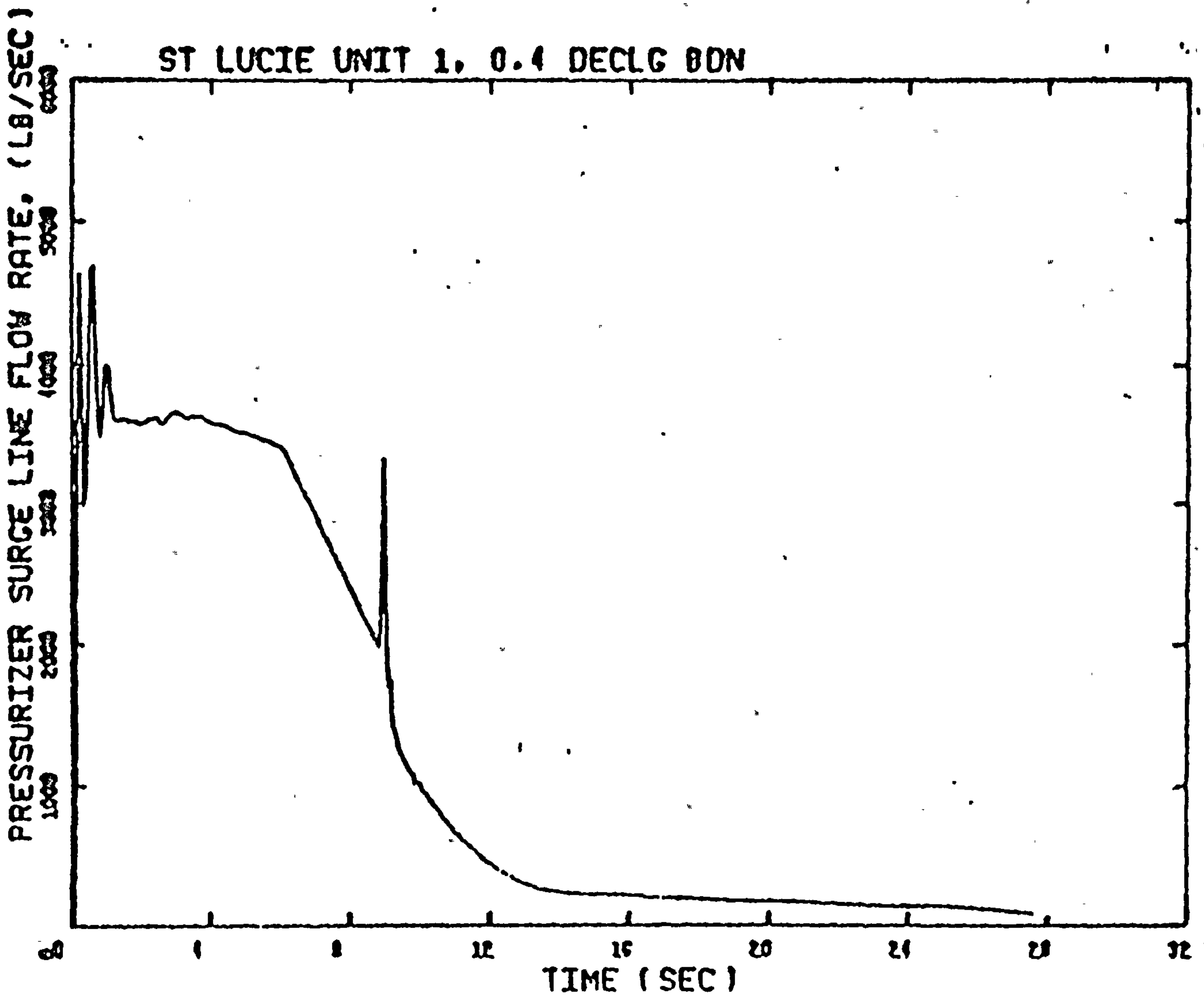

Figure 4 Blowdown Pressurizer Surge Line Flow Rate, 0.4 DECLG Break

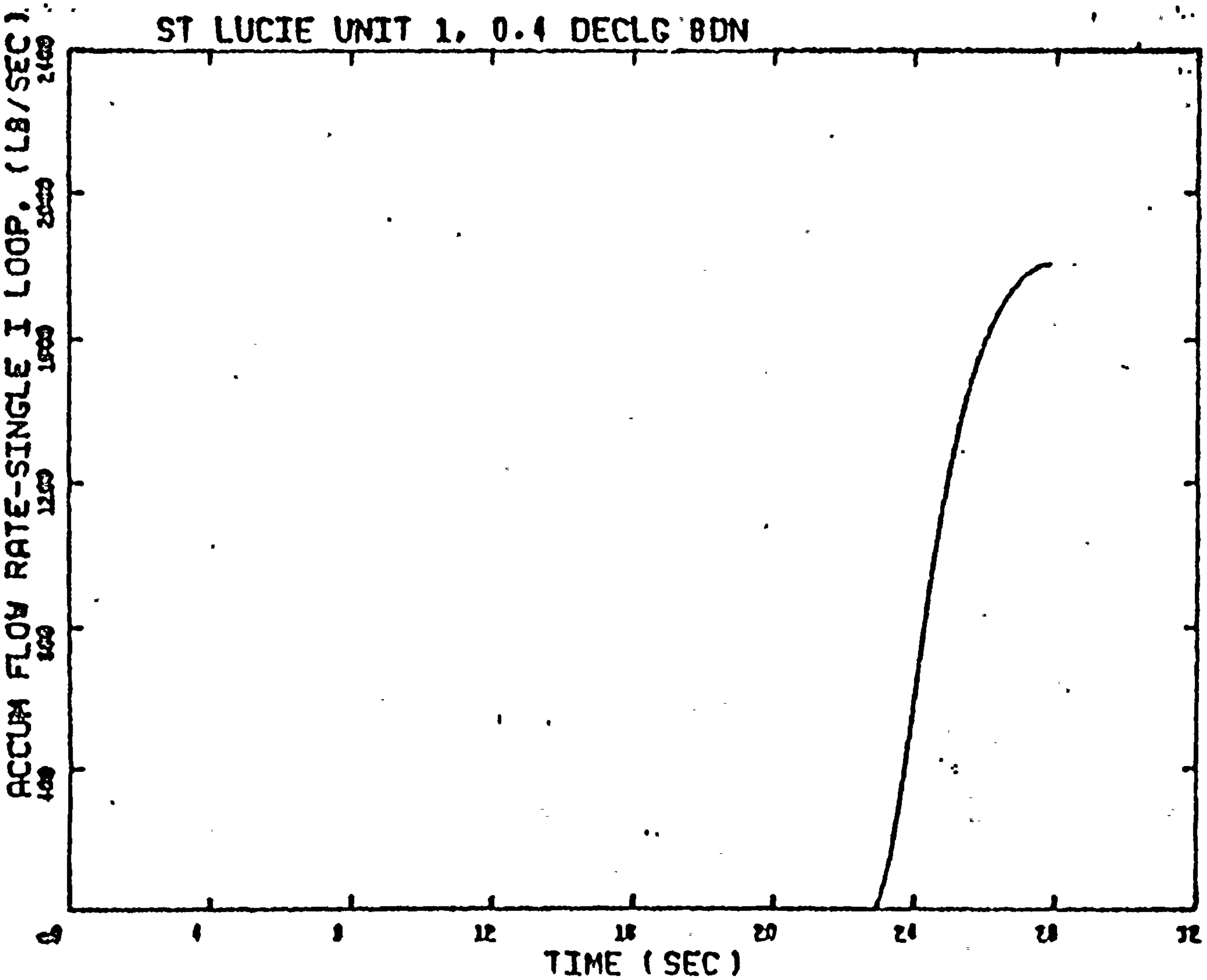

Figure 5 Single Intact Loop Accumulator Flow Rate, 0.4 DECLG Break

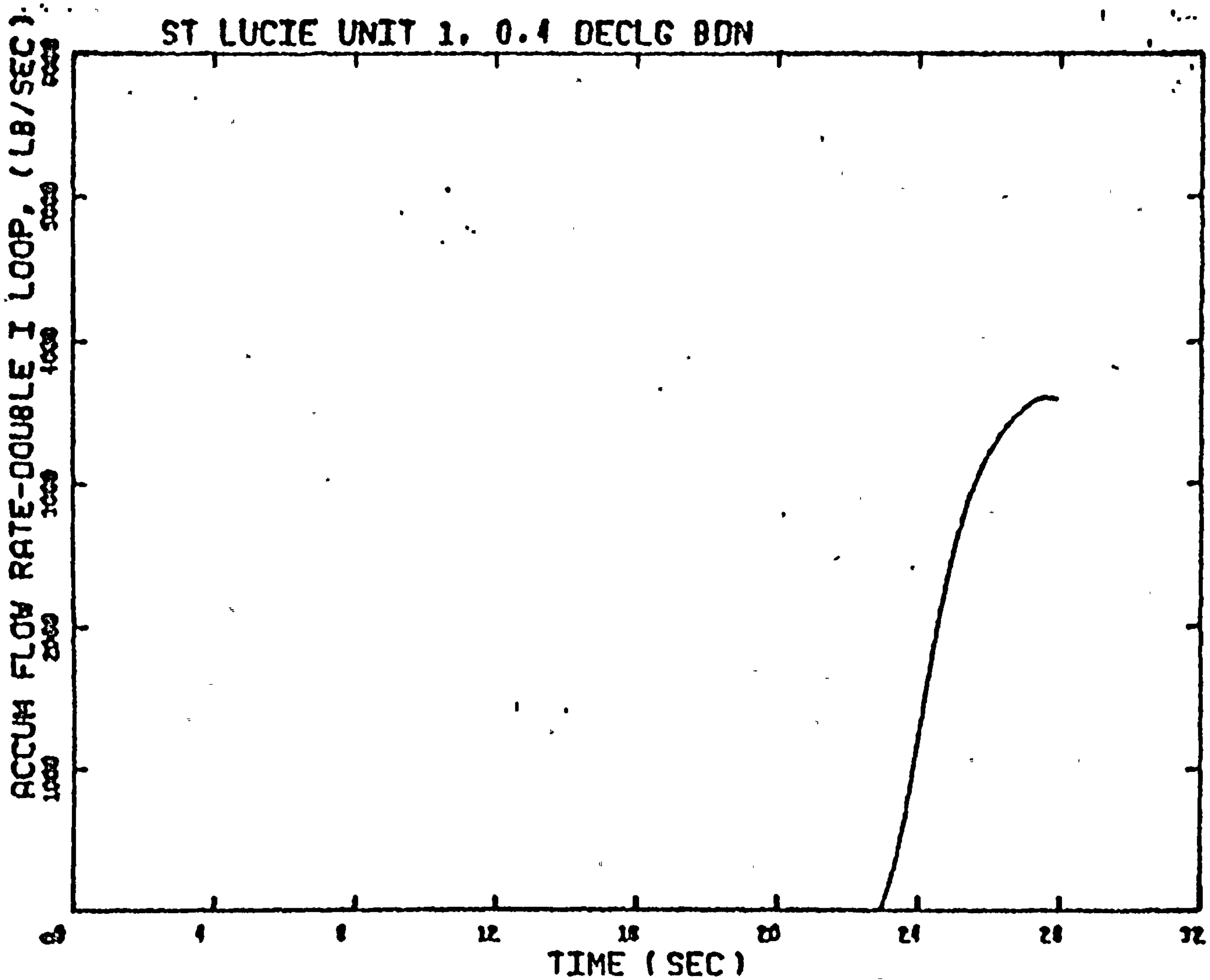

Figure 6 Double Intact Loop Accumulator Flow Rate, 0.4 DECLG Break

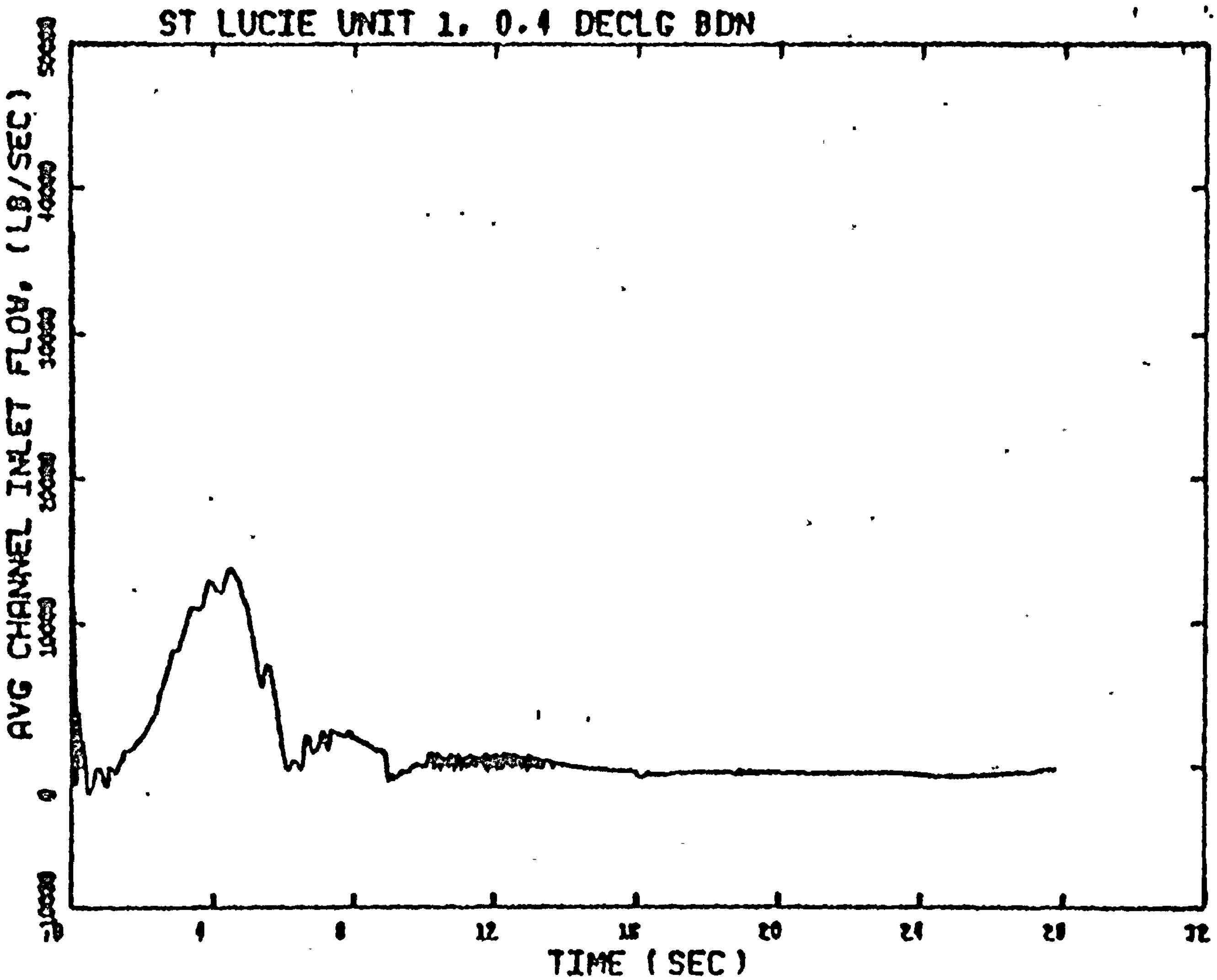

Figure 7 Blowdown Average Channel Inlet Flow Rate, 0.4 DECLG Break

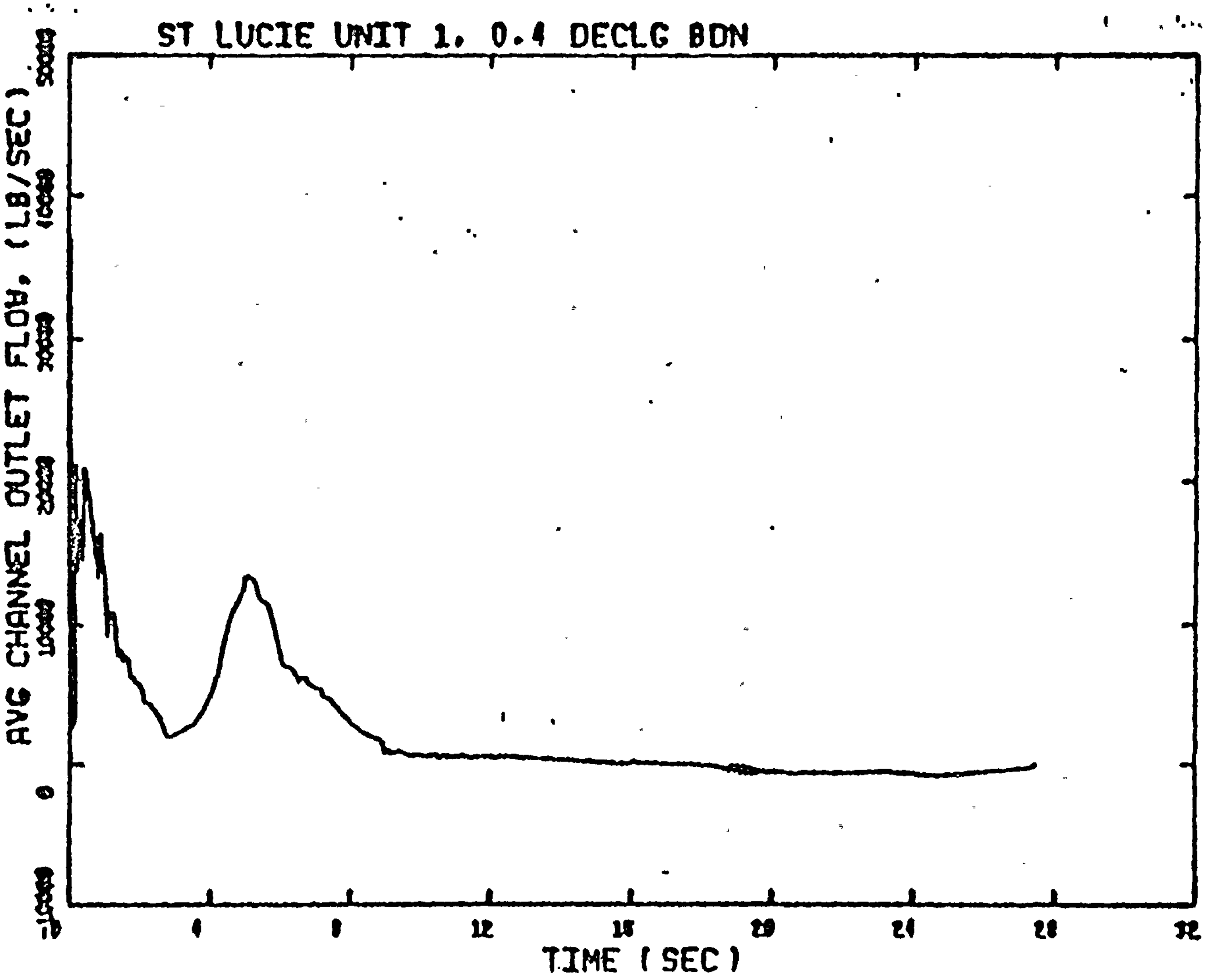

Figure 8 Blowdown Average Channel Outlet Flow Rate, 0.4 DECLG Break

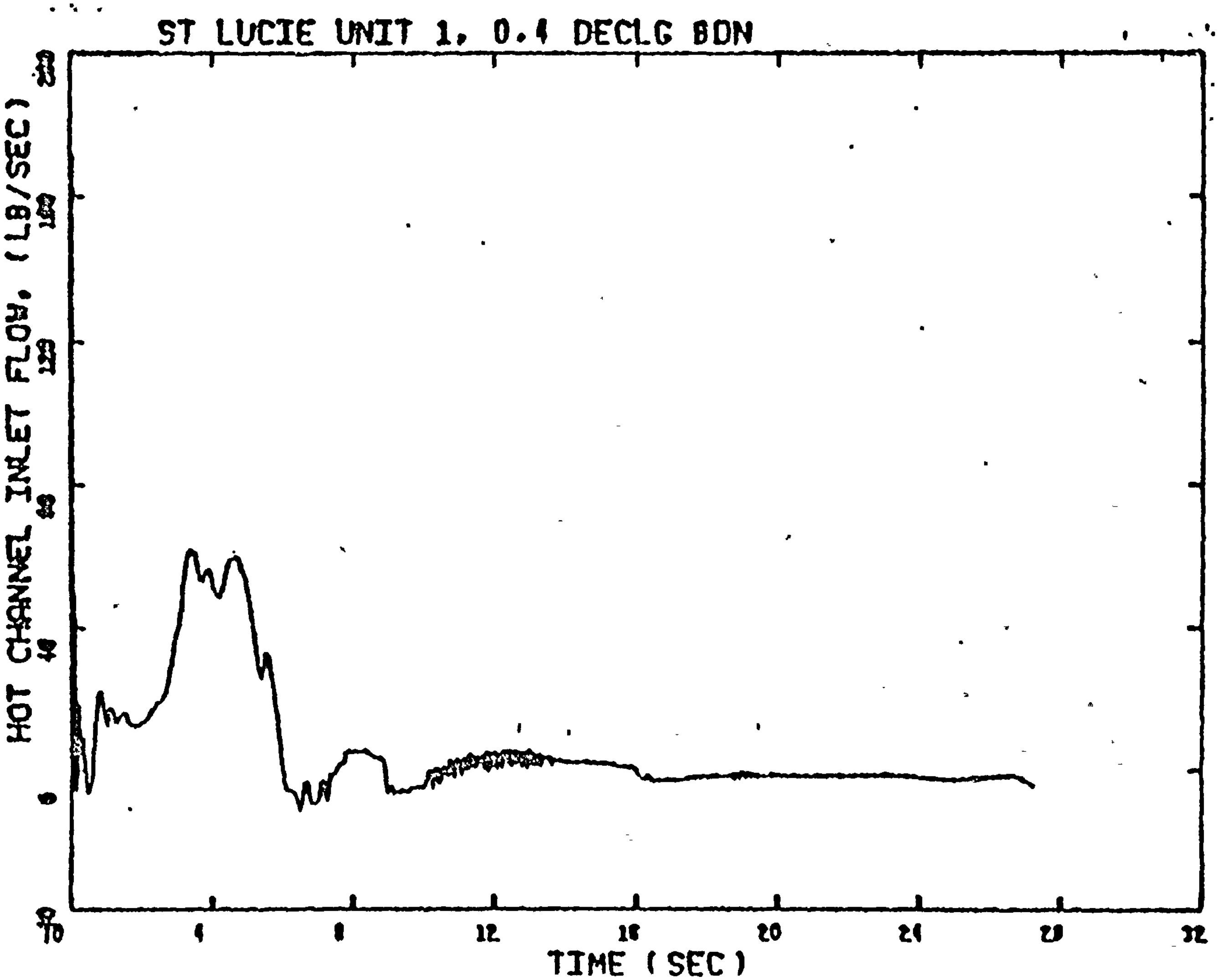

Figure 9 Blowdown Hot Channel Inlet Flow Rate, 0.4 DECLG Break

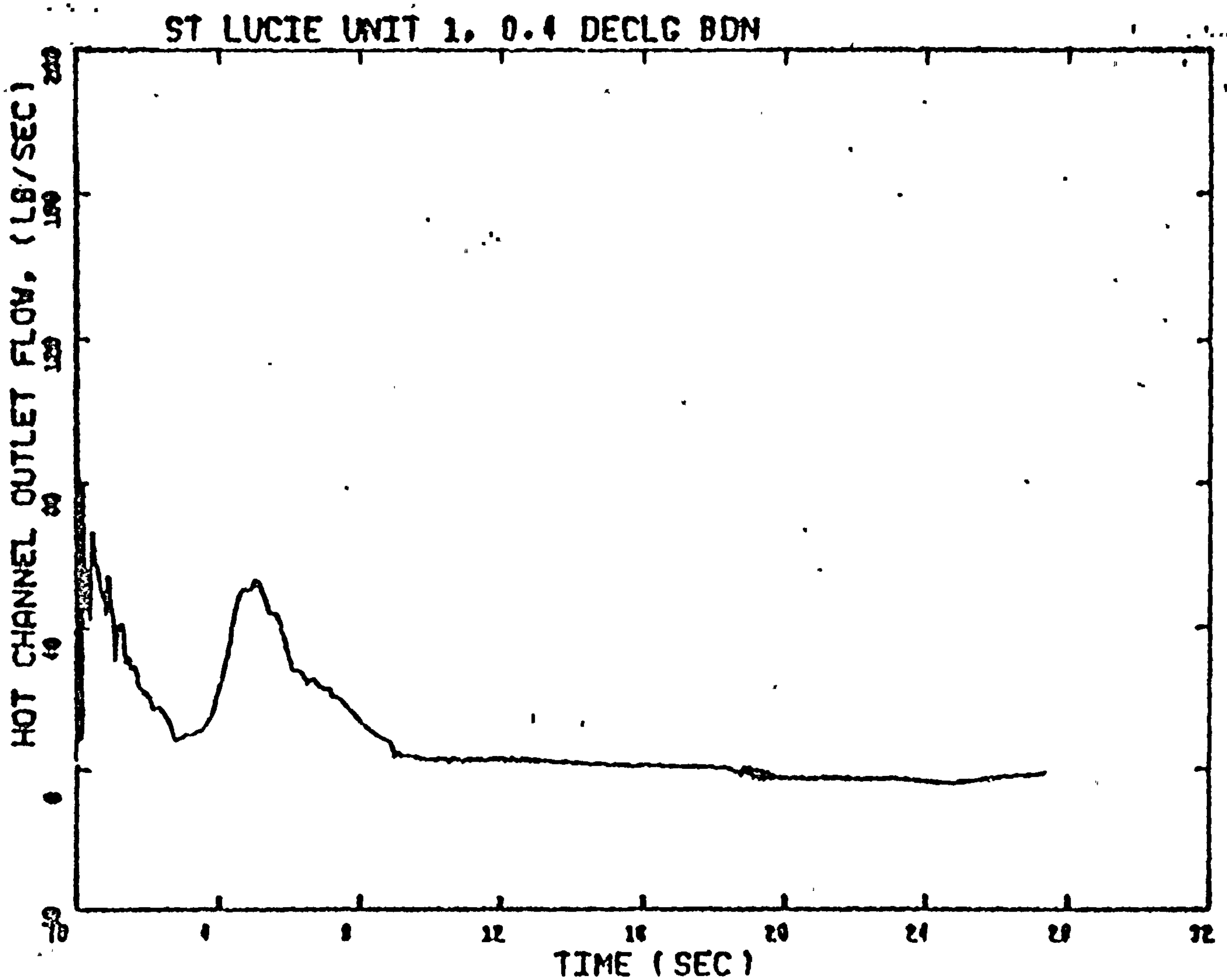

Figure 10 Blowdown Hot Channel Outlet Flow Rate, 0.4 DECLG Break

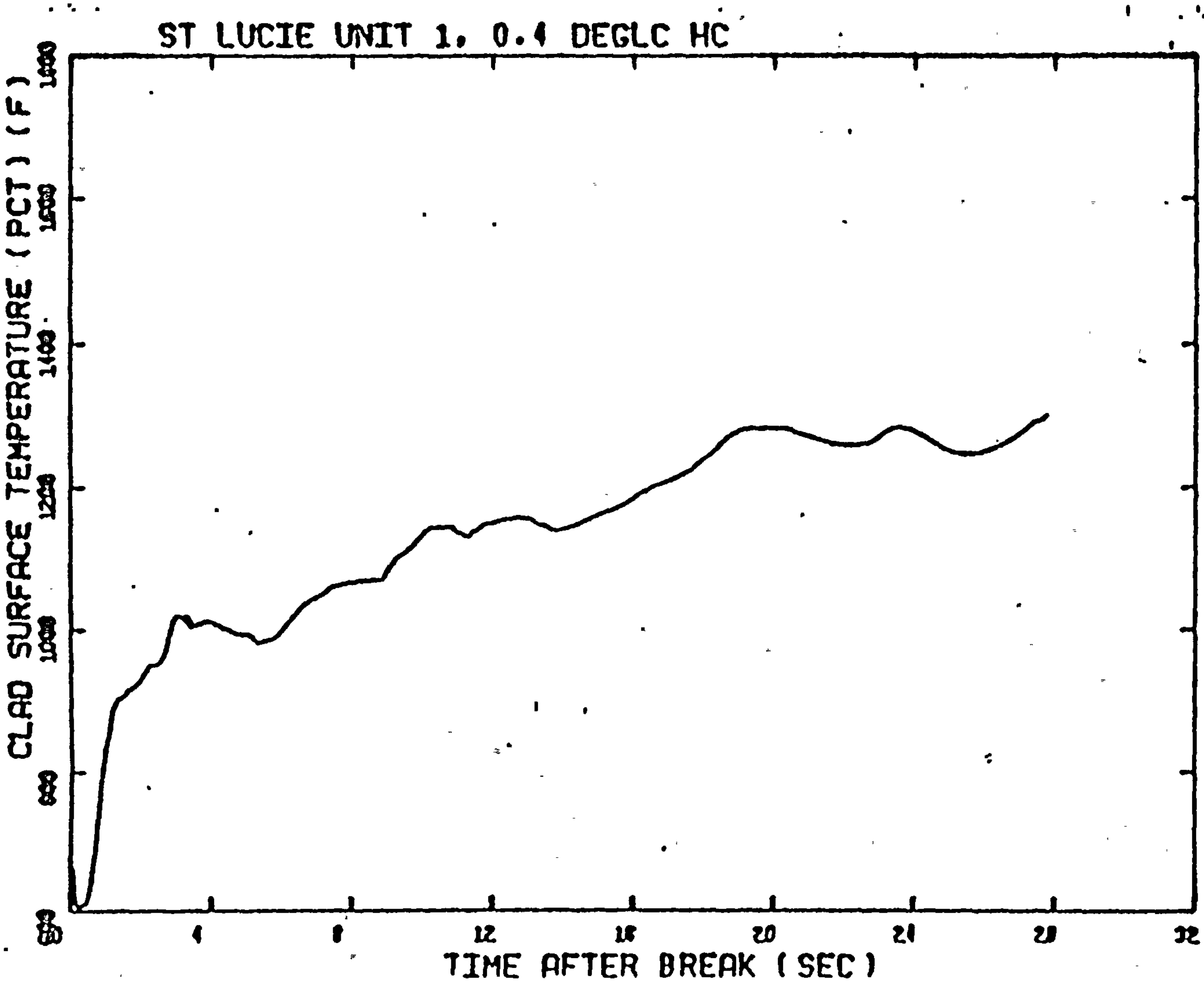

Figure 11 Blowdown Hot Rod Cladding Surface Temperature, 0.4 DECLG Break

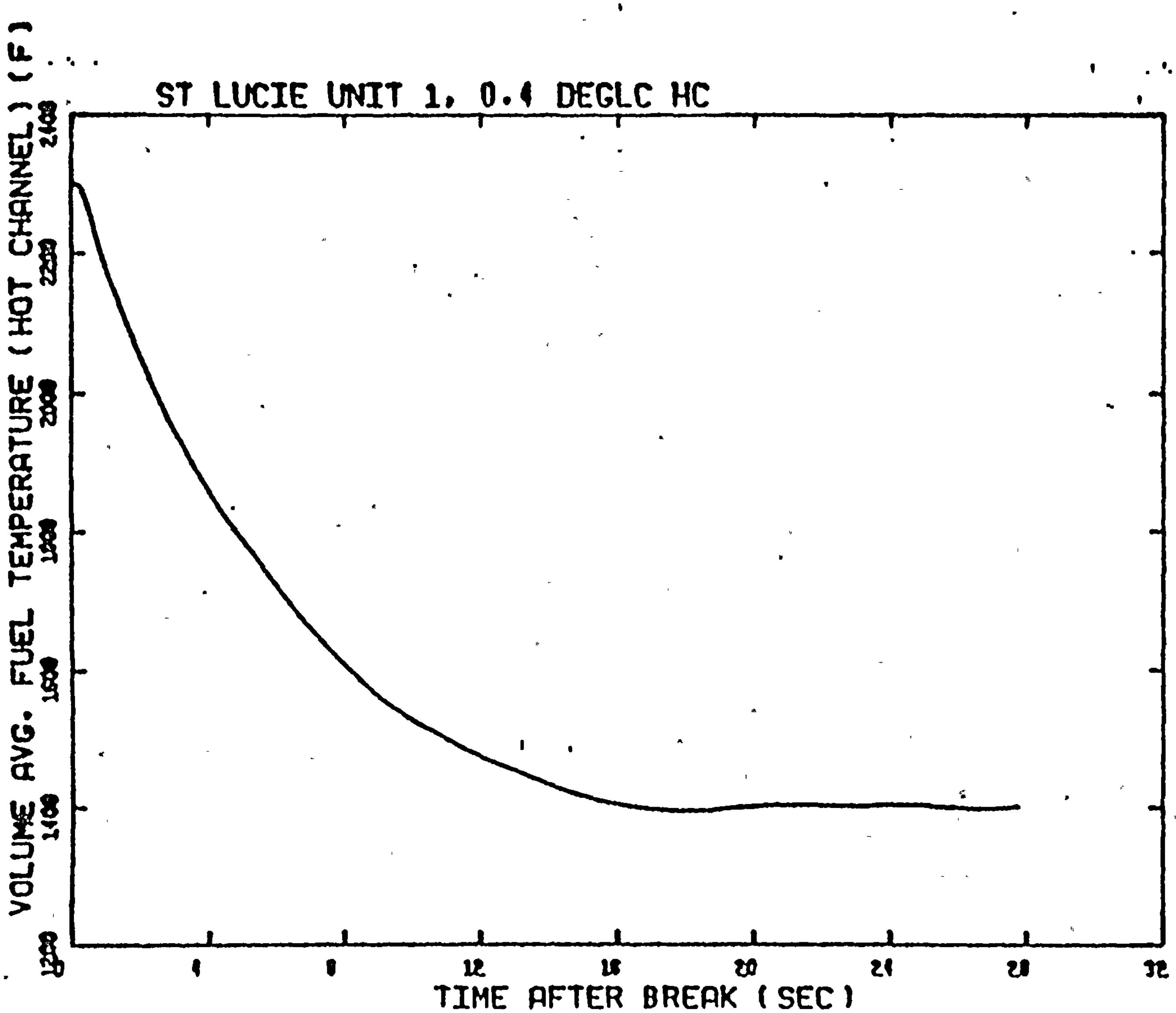

Figure 12 Blowdown Hot Rod Volumetric Average Temperature, 0.4 DECLG Break

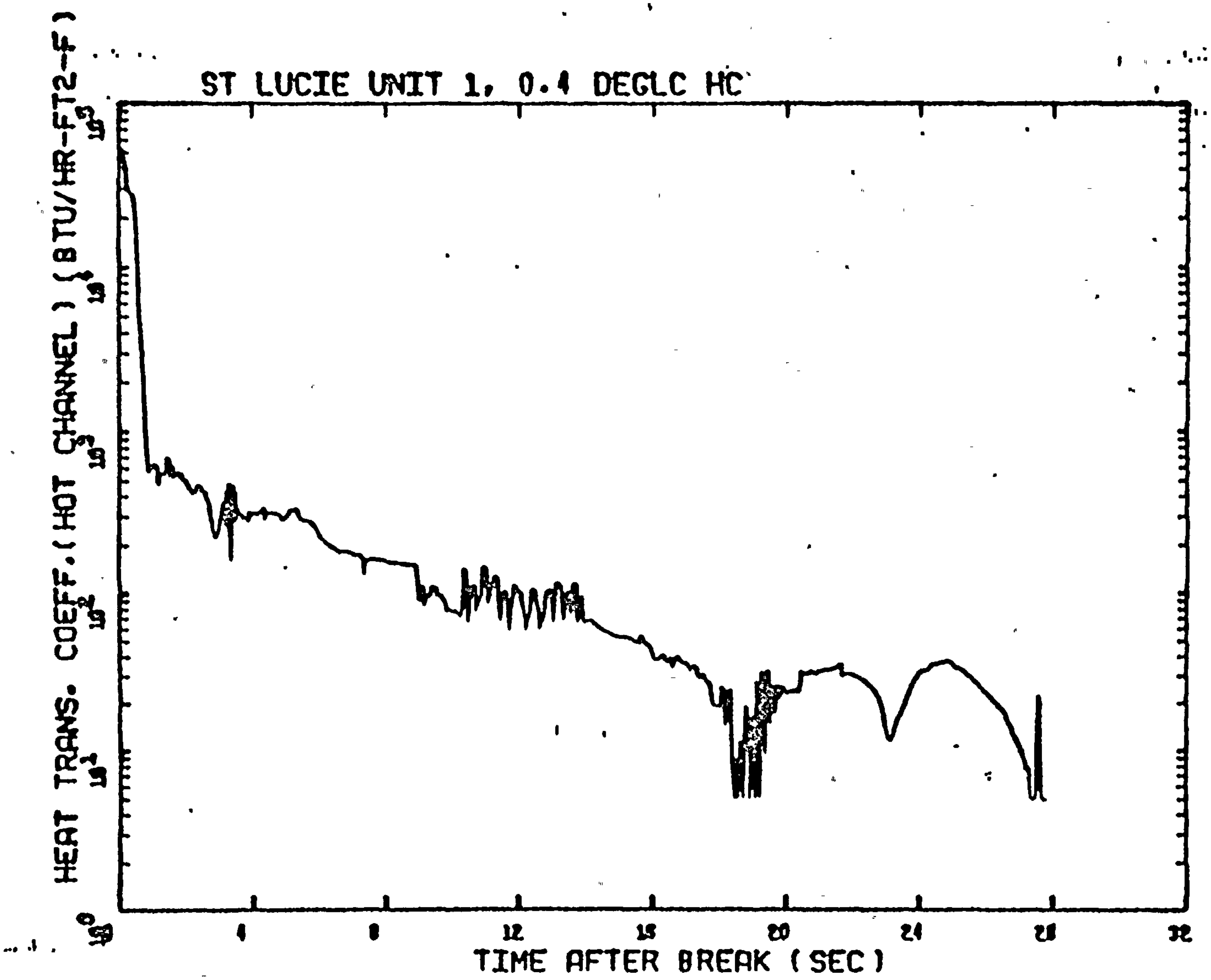

Figure 13 Hot Rod Blowdown Heat Transfer Coefficient, 0.4 DECLG Break

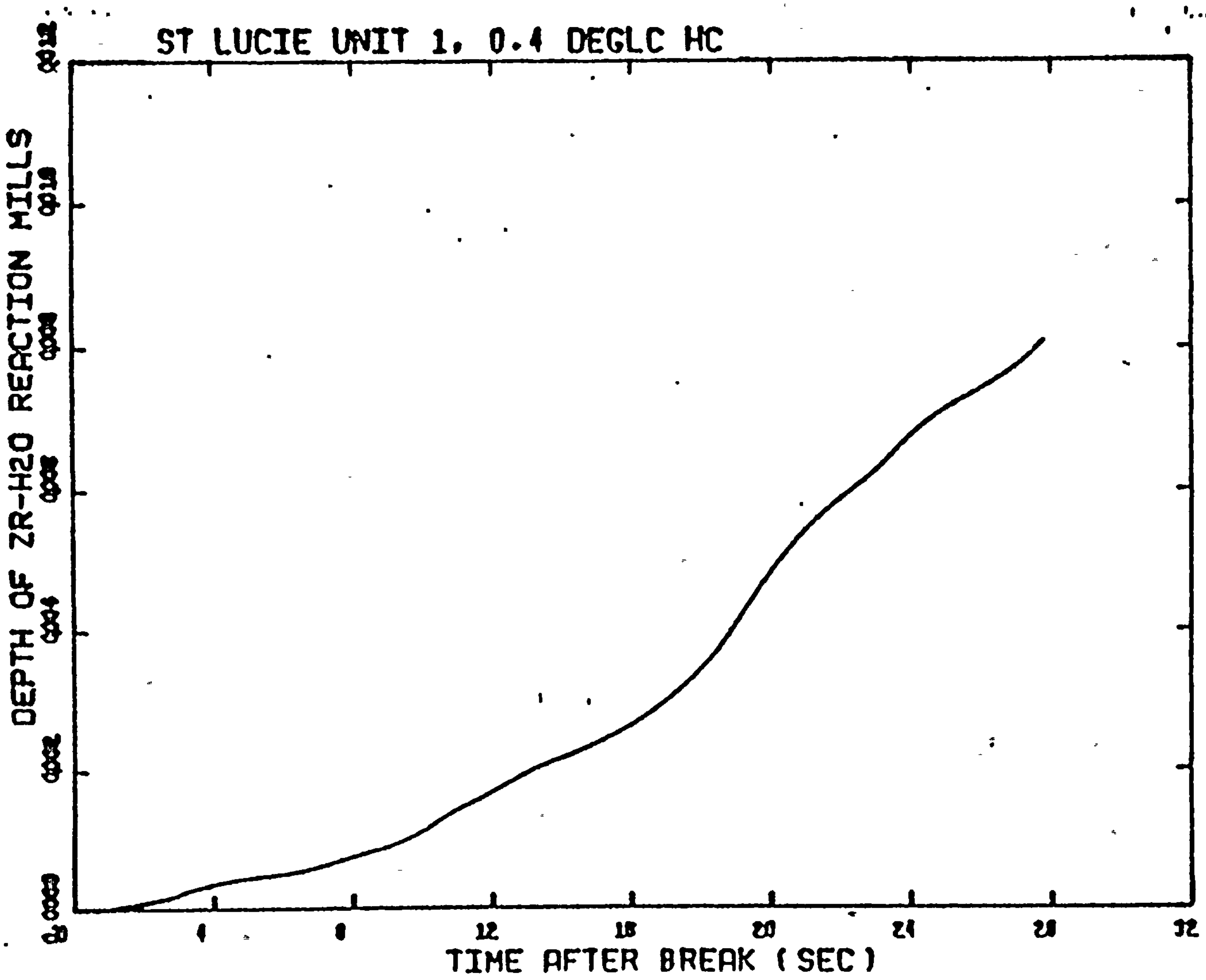

Figure 14 Hot Rod Blowdown Depth of Zirconium-Water Reaction, 0.4 DECLG Break

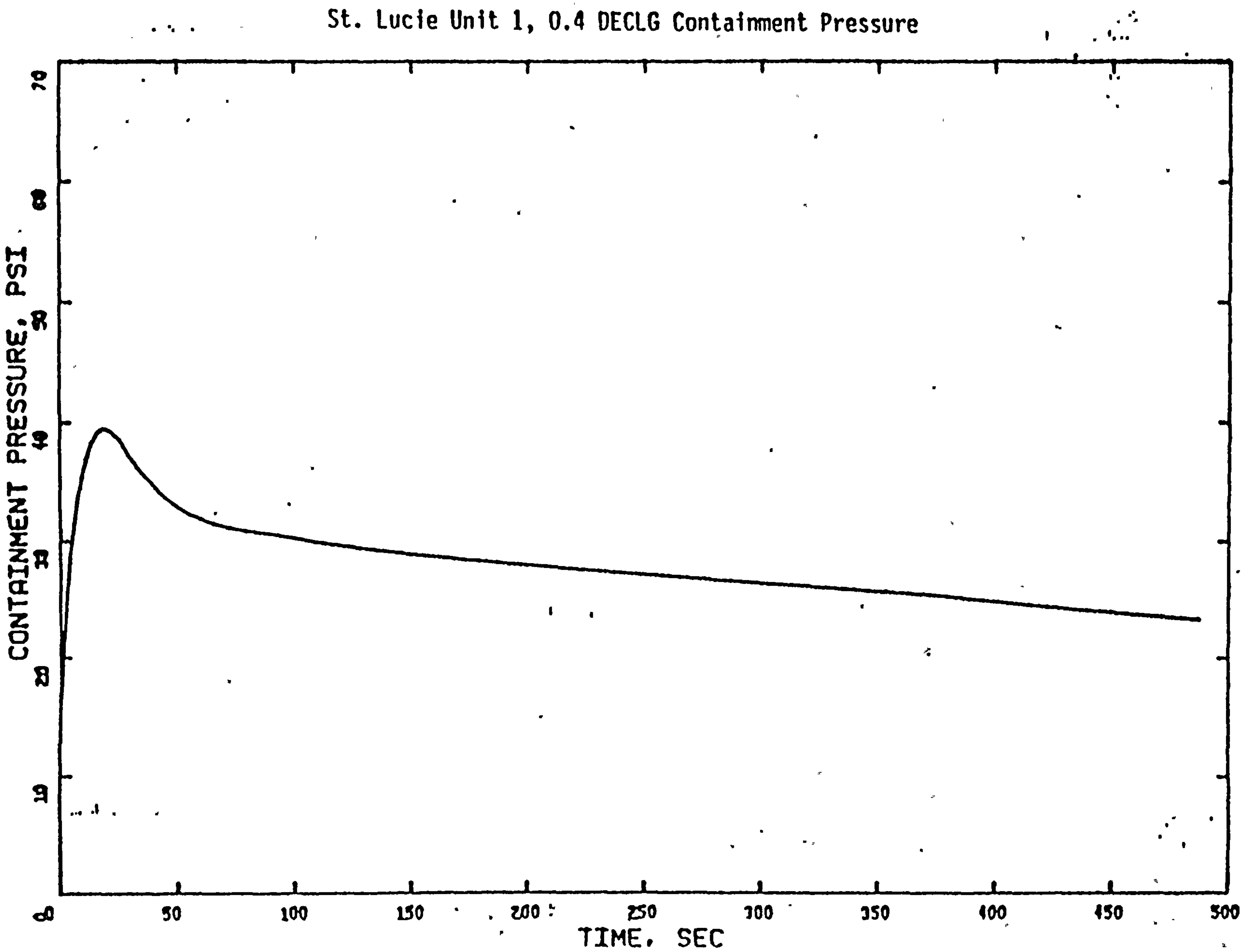

Figure 15 Containment Back Pressure versus Time, 0.4 DECLG Break

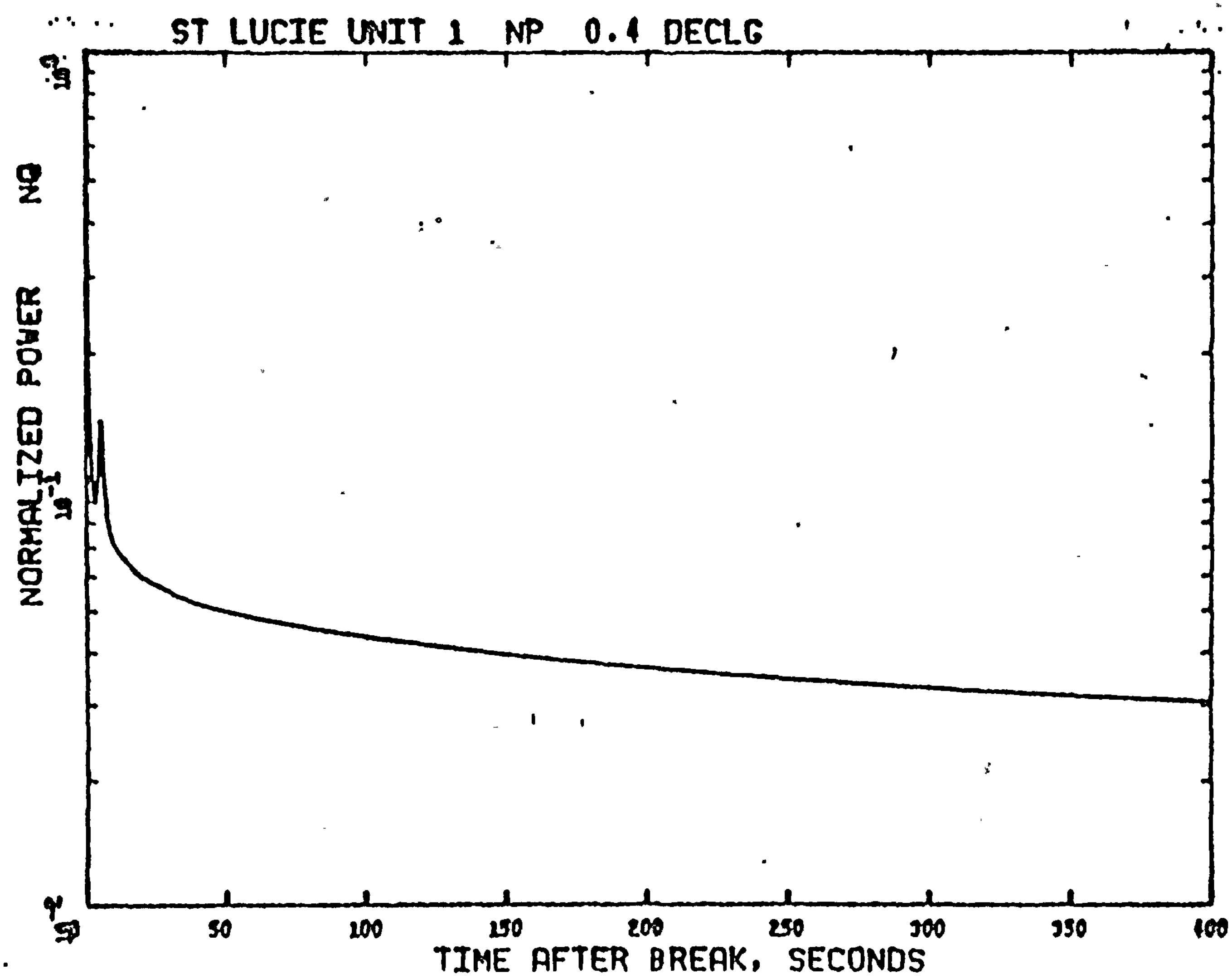

Figure 16 Normalized Power versus Time, 0.4 DECLG Break

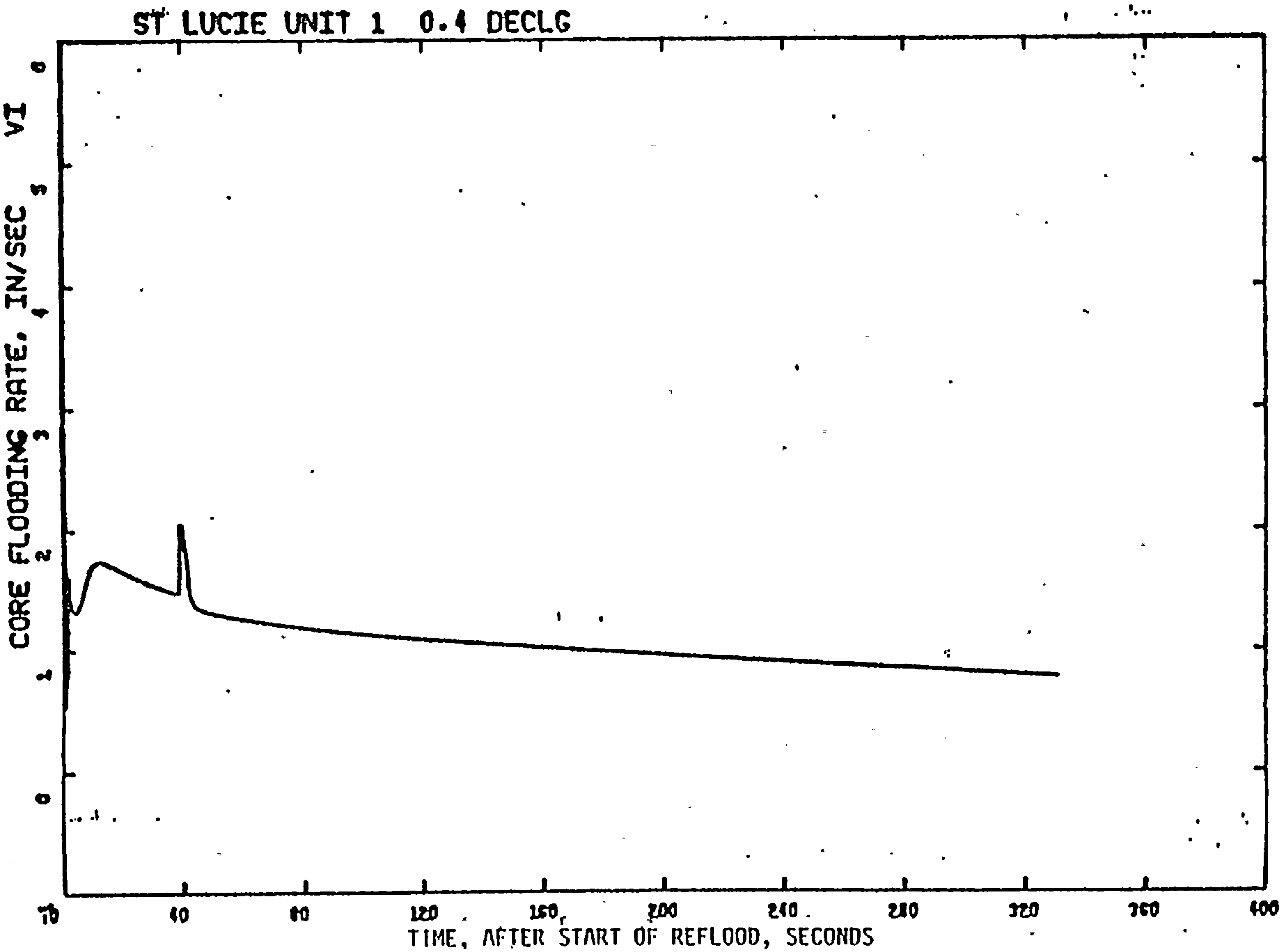

Figure 17 Core Flooding Rate, 0.4 DECLG Break

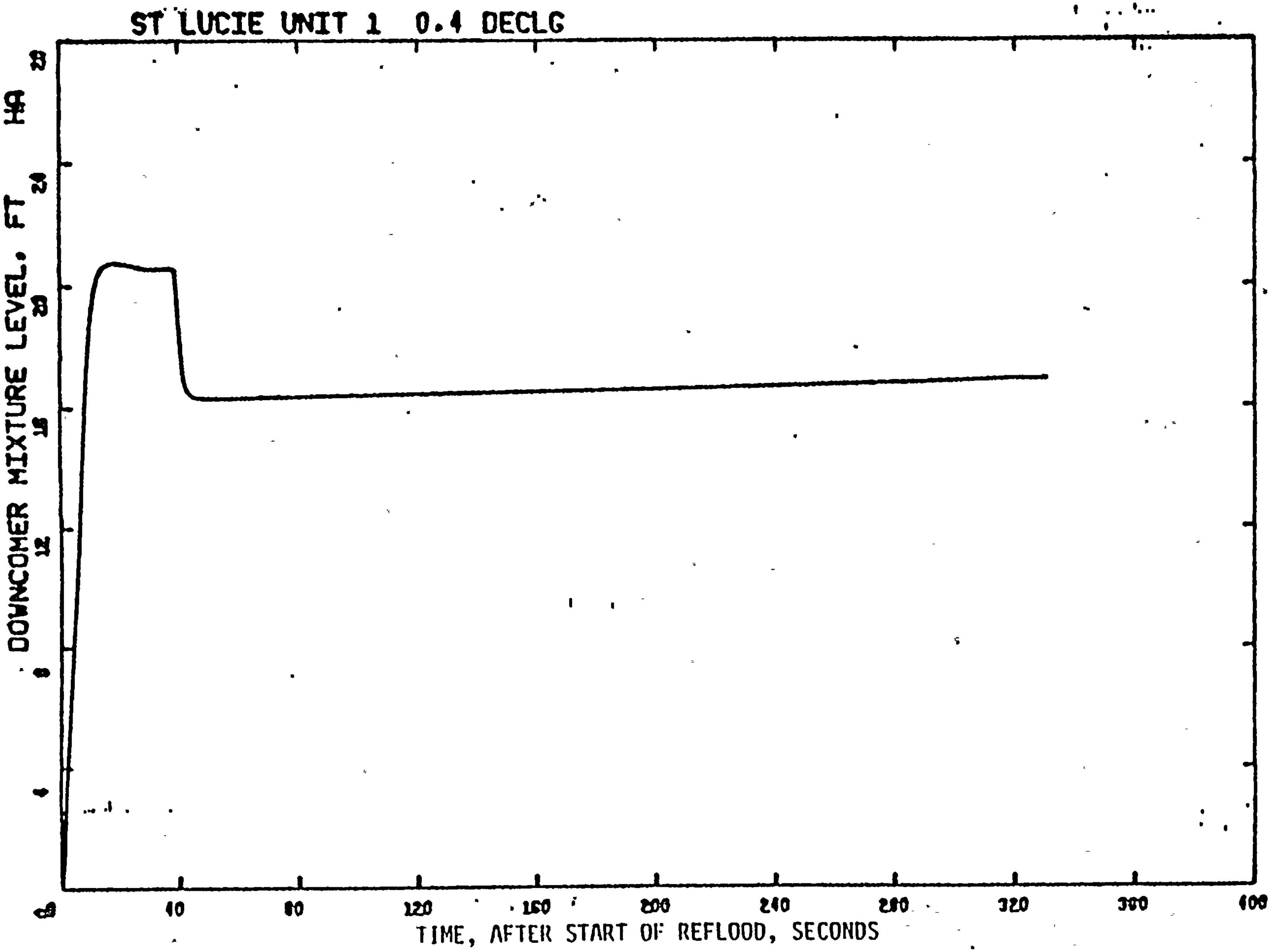

Figure 18 Downcome Mixture Level, 0.4 DECLG Break

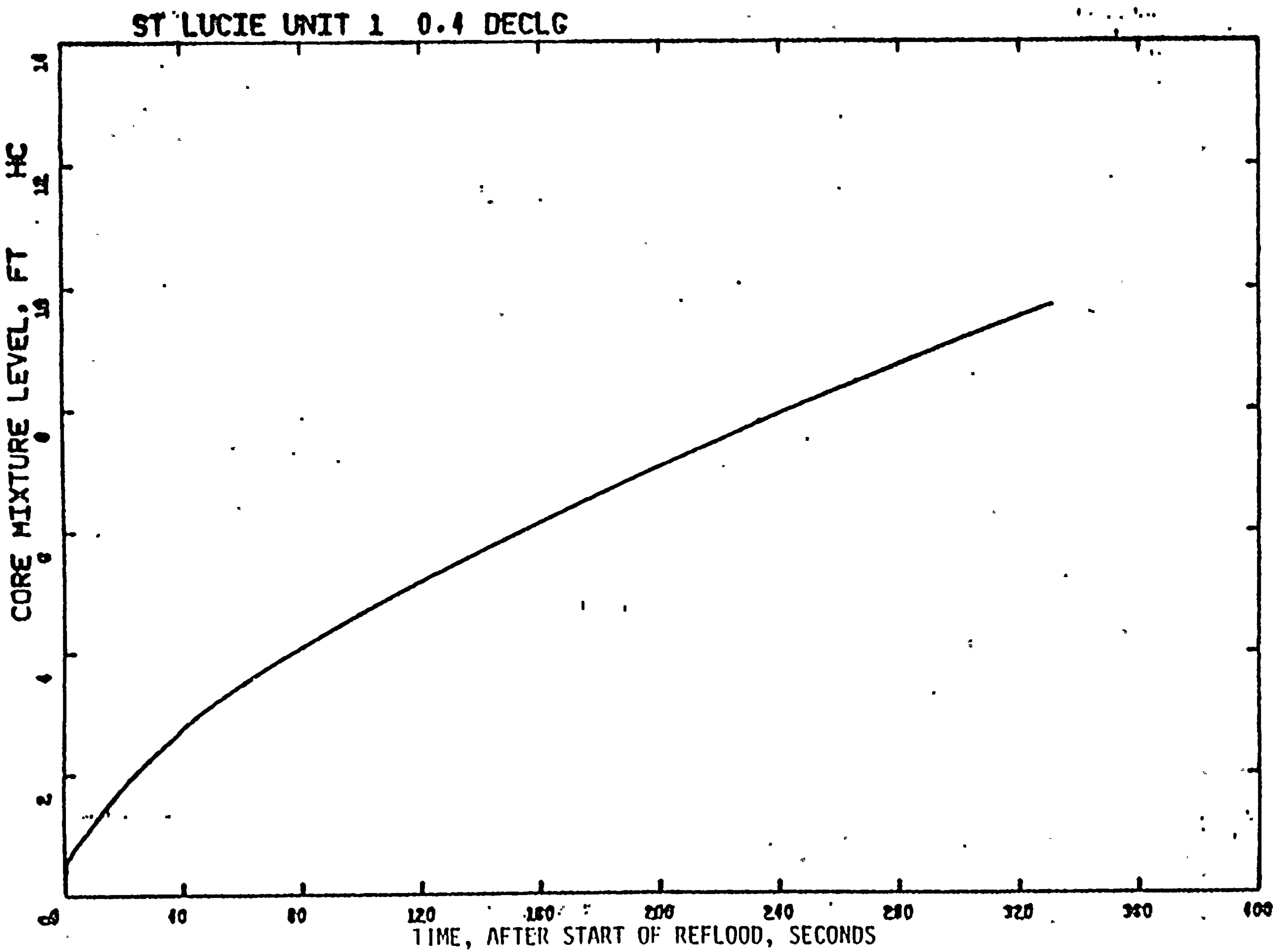

Figure 19 Core Mixture Level, 0.4 DECLG Break

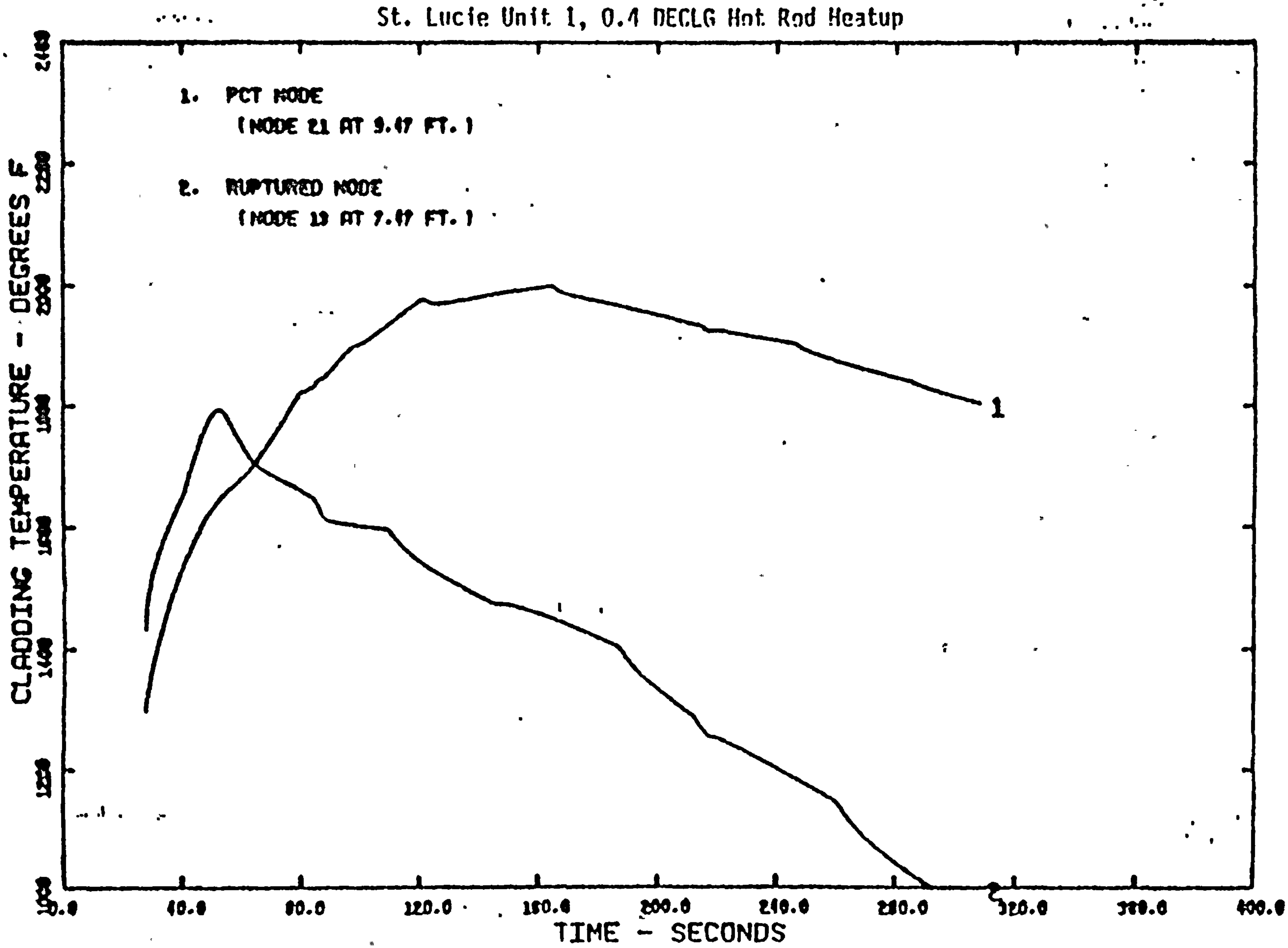

Figure 20 TOODEE2 Calculated Cladding Surface Temperature (PCT), 0.4 DECLG Break

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   c. Volume III, Revision 2, August 1975
   d. Supplement 1, August 1975
   e. Supplement 2, August 1975
   f. Supplement 3, August 1975
   g. Supplement 4, August 1975
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