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FROM:
FLORIDA POWER & LIGHT COMPANY
MIAMI, FLORIDA
MR. ROBERT E. UHRIG

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LTR. NOTORIZED 5/14/76 RE THEIR LTRS. OF 4/27/76 & 4/30/76 TRANS THE FOLLOWING:

PLANT NAME:
ST. LUCIE #1

ENCLOSURE

APPENDIX A : CONTAINS ADDITIONAL ECCS INFO.

APPENDIX B : CONTAINS CALORIMETRIC INFO.

ACKNOWLEDGED

DO NOT REMOVE

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FLORIDA POWER & LIGHT COMPANY

May 14, 1976
L-76-190

RECEIVED
MAY 19 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Director of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington DC 20555

Dear Mr. Stello:

Re: St. Lucie Plant Unit 1
Docket No. 50-335
Proposed Amendment to Facility
Operating License DPR-67

This letter provides additional information to supplement our letters of April 27 and April 30, 1976.

Appendix A to this letter contains additional ECCS information. Appendix B contains calorimetric information.

The detailed methodology for determining reactor coolant flow is in the public record in the "Reactor Coolant Pump (RCP) Flow Test Report," which is a part of the Millstone Nuclear Power Station, Unit No. 2, Docket No. 50-336. This is applicable to the St. Lucie Plant.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU:tg

Attachments

cc: Mr. Norman C. Moseley
Jack R. Newman, Esquire

Regulatory Docket File

DOCKETED
USNRC
MAY 19 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

5054

STATE OF FLORIDA)
) SS.
COUNTY OF DADE)

ROBERT E. UHRIG, being first duly sworn, deposes and says:

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

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

Robert E. Uhrig

Robert E. Uhrig

Subscribed and sworn to before me

this 14 day of May, 1976.

Theresa Garceau

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 JAN 26, 1979
BONDED THRU [illegible]

APPENDIX A

St. Lucie I Reduced Flow ECCS Performance Results

I. Introduction and Summary

The results of a St. Lucie I break spectrum analysis, using the approved Combustion Engineering large break evaluation model(2), are reported in Reference 1. This analysis, which employed a system flow rate of 139.44 x 10^6 lbs/hr, demonstrated that the LOCA Acceptance Criteria(3) were met at a peak linear heat generation rate (PLHGR) of 15.8 kw/ft.

Subsequent to the analysis reported in Reference 1, a conservative reanalysis has been performed to determine the allowable PLHGR when the system flow rate is reduced to 134.06 x 10^6 lbs/hr, which is 4% less than the flow rate used in Reference 1. The allowable PLHGR at the reduced flow was found to be 15.6 kw/ft. In order to determine sensitivity to linear heat rate, a second reduced flow case was examined, at a PLHGR of 14.2 kw/ft. The results for this case, as well as those discussed above, are summarized below:

Worst Break* LOCA Results

System Flow (X10^6 lb/hr)	139.44	134.06	134.06
PLHGR (kw/ft)	15.8	15.6	14.2
Peak Clad Temperature (°F)	2192	2189	1956
Peak Local Clad Oxidation (%)	10.42	9.71	5.32
Peak Core-Wide Clad Oxidation (%)	<.787	<.787	<.787

These results show that all LOCA acceptance criteria(3) are met at a PLHGR of 15.6 kw/ft, and there is substantial margin at 14.2 kw/ft.

An analysis is presented in Reference 1 which demonstrates that one Safety Injection Tank can be isolated without reducing core power or PLHGR. The flow rate reduction will not alter this conclusion.

II. Discussion and Results

In order to determine the sensitivity of peak clad temperature to flow, the worst break (0.8 DEG/PD) identified in Reference 1 was repeated at the lower flow rate. As a conservative approximation, the CEFLASH-4A(4) flow

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* The worst break is the 0.8 Double-Ended Guillotine at the Pump Discharge (0.8 DEG/PD)

rate determined for the worst break in Reference 1 was simply reduced 4% and the STRIKIN-II$^{(5)}$ calculation was repeated; the CEFLASH-4A calculation was not rerun at the lower flow rate (or at a reduced core power).

If the CEFLASH-4A calculation had been rerun with a lower initial flow rate, the transient flow during blowdown would not have been as conservative as the flow obtained by uniformly reducing the reference flow by 4%.

As noted in Section I, calculations were performed at two PLHGR values: 15.6 kw/ft and 14.2 kw/ft. The core and system parameters used for these calculations are the same as those reported in Reference 1; the only differences are the flow and parameters dependent on the PLHGR, which are tabulated in Table II-1.

Pertinent information for the two cases are reported in Table II-2. Since the peak clad temperatures and peak local oxidation percentages for the reduced flow cases are less than the values reported in Reference 1, it follows that the core-wide oxidation percentages would also be lower. Therefore, it was not necessary to recalculate the core-wide oxidation percentage; the value in Table II-2 is from Reference 1.

The results of the low flow reanalysis are depicted graphically in Figures II-1 and II-2, which show peak clad temperature and local clad oxidation percentage as a function of time for the two PLHGR values considered.

III. REFERENCES

1. Safety Injection System Analysis of St. Lucie Unit I for FSAR, Section 6.3.3.6 as amended by Revision #55, 2-9-76.

2. CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model", August, 1974 (Proprietary).

   CENPD-132, Supplement 1, "Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model", December, 1974 (Proprietary).

   CENPD-132, Supplement 2, "Calculational Methods for the C-E Large Break LOCA Evaluation Model", July, 1975.

3. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3 - Friday, January 4, 1974.

4. CENPD-133, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis", April 1974 (Proprietary).

   CENPD-133, Supplement 2, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modification)", December, 1974 (Proprietary).

5. CENPD-135 - "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program", April, 1974 (Proprietary).

   CENPD-135, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)", December, 1974 (Proprietary).

Table II-1

Selected System Parameters
Reduced Flow Study

Quantity			
System Flow Rate (total)	134.06×10^6		lbs/hr
Core Flow Rate	129.10×10^6		lbs/hr
Peak Linear Heat Generation Rate	15.6	14.2	kw/ft
Gap Conductance at PLHGR	806.3	725.7	Btu/hr-ft^2-^{o}F
Fuel Centerline Temp. at PLHGR	4021.6	3765.9	^{o}F
Fuel Average Temp. at PLHGR	2640.6	2513.3	^{o}F

Table II-2

Times of Interest, Peak Clad Temperature, and Oxidation Percentages
0.8 DEG/PD with Reduced Flow

PLHGR	15.6 kw/ft	14.2 kw/ft
Hot Rod Rupture Time	9.97 sec.	10.2 sec.
Peak Clad Temperature	2188.5 °F	1956.1 °F
Peak Local Clad Oxidation	9.71%	5.32%
Peak Core-Wide Clad Oxidation	<0.787%	<0.787%

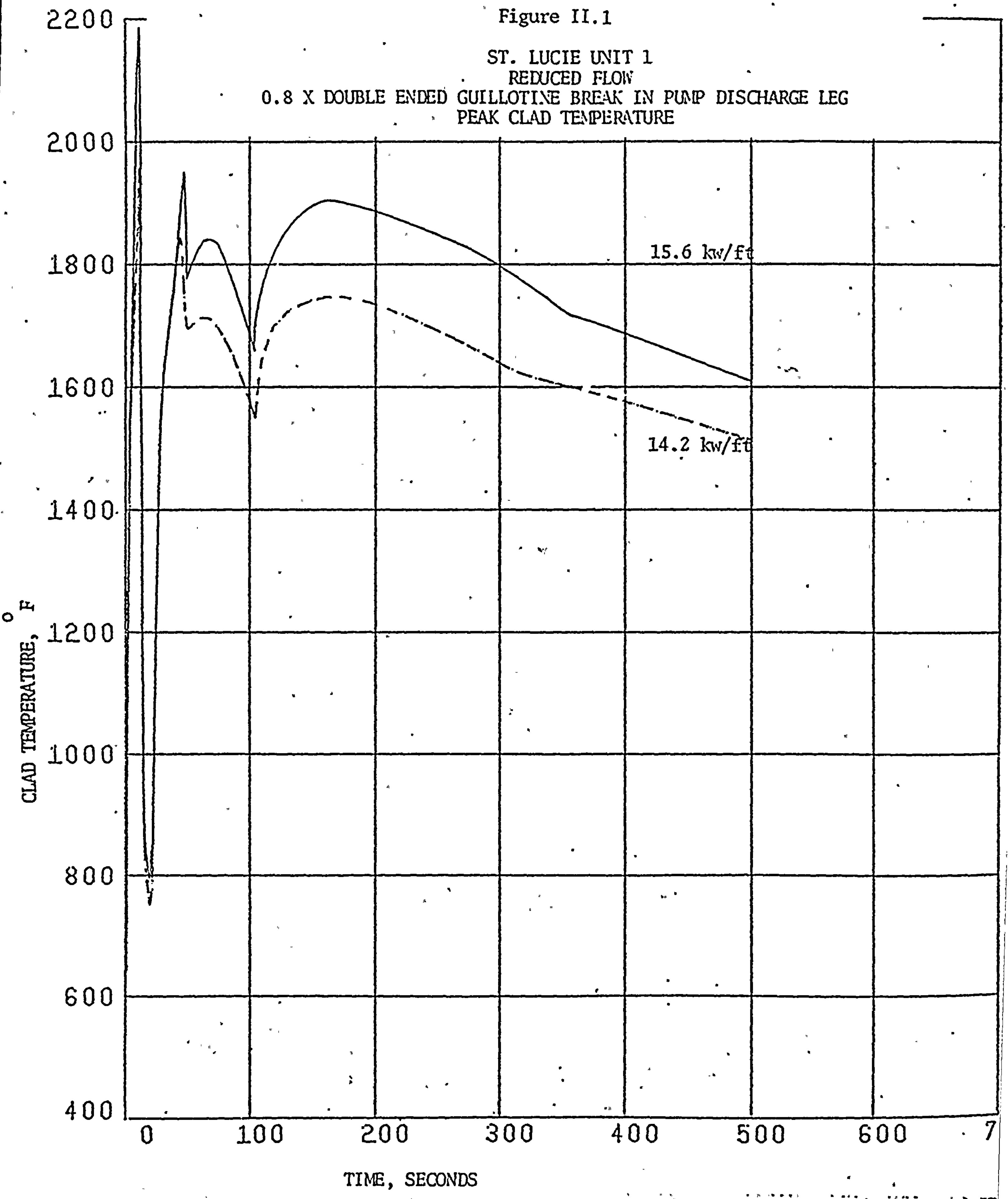

Figure II.1

ST. LUCIE UNIT 1
REDUCED FLOW
0.8 X DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

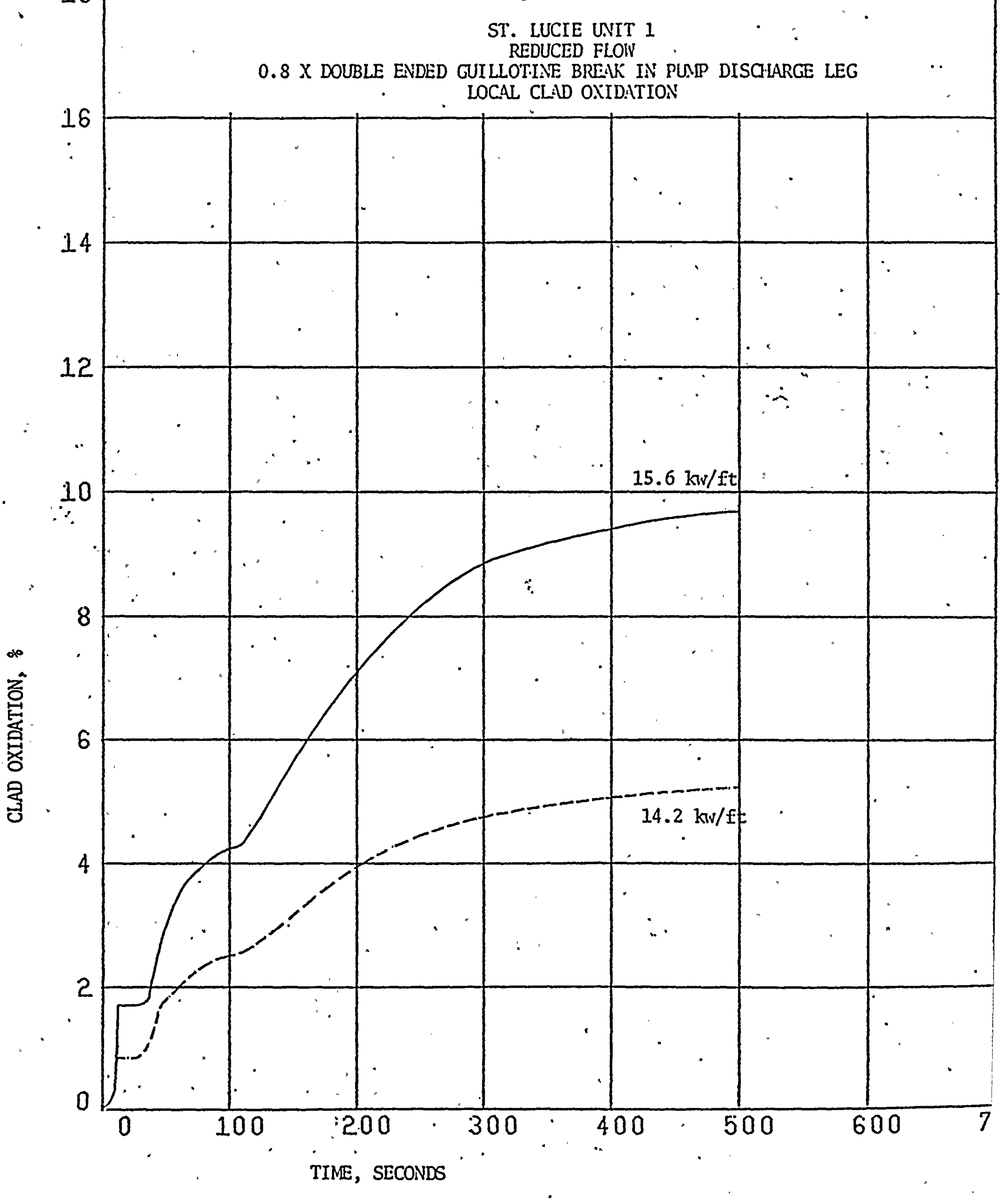

Figure II.2

ST. LUCIE UNIT 1
REDUCED FLOW
0.8 X DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
LOCAL CLAD OXIDATION

APPENDIX B

Calorimetric techniques are methods of measurement employing basic NSSS heat balance equations. Easily measured parameters, such as temperature and pressure, are used in a system heat balance equation to estimate other parameters which may be difficult to measure directly. Calorimetric techniques are used in power ascension testing to determine a variety of system parameters. The context in which calorimetrics are germane to the interim tech. specs. is limited to the extent that the subject tech. specs. allow ascension to the higher power levels which are prerequisite to accurate calorimetric determinations of flow.

The general form of the NSSS heat balance equation is as follows:

$$Q_{SG} = Q_{Core} + Q_{other\ sources} - Q_{losses}$$

i.e., the total heat output from the steam generators equals the sum of the sources (core, pumps, pressurizer heaters, etc.) minus the heat losses (to containment, ambient component cooling water, etc.), where heat output of the steam generator (Q_{SG}) is determined by measurement of the saturated steam pressure and the feedwater flow rate and temperature. The feedwater flow is accurately measurable due to the calibrated nozzles in the feedwater lines. The heat output of the core (Q_{Core}) equals $W\Delta h$, where W is the flow rate being derived and the enthalpy rise (Δh) is derived from measured inlet and outlet coolant temperatures. It should be noted that the enthalpy rise measurement error decreases with increasing power level. $Q_{other\ sources}$ is obtained by applying known efficiencies and measurements of electrical power consumption of the R. C. pump motors and pressurizer heaters. Q_{losses} are measured during the post-core hot functional testing in conjunction with measurement of the $Q_{other\ sources}$.

Solving this equation for flow yields an estimate which has an accuracy equivalent to that of the pump Δp flow measurement techniques. Calorimetrics provide a useful confirmatory technique due to the use of totally independent and accurately measurable system parameters.