UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323

Report Nos.: 50-335/86-22 and 50-389/86-21

Licensee: Florida Power and Light Company
9250 West Flagler Street
Miami, FL 33102

Docket Nos.: 50-335 and 50-389 License Nos.: DPR-67 and NPF-16

Facility Name: St. Lucie 1 and 2

Inspection Conducted: November 3 - 7, 1986

Inspector: P. T. Burnett 11-21-86
Date Signed

Approved by: Frank Jape 11/21/86
F. Jape, Section Chief
Engineering Branch
Division of Reactor Safety
Date Signed

SUMMARY

Scope: This routine, unannounced inspection addressed the areas of reactor coolant system leakrate testing, thermal power monitoring, and followup of previous outstanding items.

Results: No violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. H. Barrow, Operations Superintendent
   J. A. Dyer, Quality Control
   *R. J. Frechette, Chemistry
   K. N. Harris, Vice President
   C. F. Leppla, (I&C) Supervisor
   E. Ordway, I & C Engineer
   *L. W. Pearce, Operations Supervisor
   *N. G. Roos, Quality Control Supervisor
   D. A. Sager, Plant Manager
   *D. M. Stewart, Technical Staff
   *D. H. West, Technical Staff
   *C. L. Wilson, Mechanical Maintenance
   *B. Winnard, Independent Safety Engineering Group
   *E. Wunderlich, Reactor Engineering

   Other licensee employees contacted included nuclear plant supervisors, assistant nuclear plants supervisors, operators, and office personnel.

   NRC Resident Inspectors

   R. V. Crlenjak, Senior Resident Inspector
   H. T. Bibb, Resident Inspector

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on November 7, 1986, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings. No dissenting comments were received from the licensee. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspector during this inspection.

3. Licensee Action on Previous Enforcement Matters

   This subject was not addressed in the inspection.

4. Unresolved Items

   No unresolved item was identified during this inspection.

5. Reactor Coolant System Leakrate Measurement - Unit 1 (61728)

Data were collected at 15 minute intervals over a 3.25 hour period for analysis of the Unit 1 reactor coolant system gross and identified leak rates. This period encompassed the time the licensee was performing their routine surveillance to satisfy the requirements of Technical Specification 4.7.2.1.d. The licensee's procedure calls for a minimum two-hour test. The time was extended to facilitate comparison with the inspector's calculations, which were performed using the microcomputer program RCSLK9. That program is described in NUREG-1107, RCSLK9: Reactor Coolant System Leakage Determination for PWRs. The plant-specific data used in the analysis are shown in attachment 1 and the result of the 3.25 hour calculation in attachment 2.

The 13 sets of data collected allowed calculation of ten one-hour duration tests and six two-hour duration tests. The licensee's procedure allows addition of makeup water during the test, but for a 2.25 hour period no water was added. That period was separately analyzed as well as the total period. Finally, the program was used to obtain the temporal increase in mass deficit and collected mass at the end of each set of observations. These two data sets were each reduced to best fit straight lines using least-squares analysis. The slopes of the lines were taken as gross and identified mass leakage. The process was repeated for the ten observations of gross leakage without water addition. The results are summarized in the table below:

MEAN GROSS LEAKAGE (GPM)

10	1-HOUR TESTS	0.99 +/- 0.434	
6	2-HOUR TESTS	1.10 +/- 0.126	
1	2.25-HOUR TEST	1.04	(no water addition)
1	3.25-HOUR TEST	0.96	
FIT TO 13 OBSERVATIONS		1.06 +/- 0.074	
FIT TO 10 OBSERVATIONS		1.08 +/- 0.147	(no water addition)

The slope of the identified leakage line for 13 observations was 0.71 +/- 0.015 gpm. The two-thirteen observation fits are shown in attachment 3 and the one-ten observation fit is shown in attachment 4. Both the regression analysis and the plotting of the results were performed with the SUPERCALC3 spreadsheet program.

One conclusion derived from these analyses was that water addition had no adverse effect on the quality of the results. This was surprising in light of results at other facilities. However, the makeup flow sensor, which inputs to the flow integrator, is based upon a calibrated orifice rather than a rotating vane or other less precise device, and calibration records showed the flow integrator to have good long term stability. (The calibration records of all instruments used in the measurement were inspected and found satisfactory.)

Another conclusion, based upon the standard deviations of the sets of one- and two-hour tests, was that the routine surveillance test duration should not be reduced below two hours. The data collected here as well as those used by the licensee were obtained by visual observation of control board instruments, and the tests are nothing more than comparison of endpoints of the observables. Errors in reading the endpoints become less significant as test duration increases.

Finally, the licensee's method of determining reactor coolant system leak-rate is acceptable. Their result was bracketed by those obtained by the inspector.

6. Thermal Power Measurements (61706)

For each unit the hourly surveillance of thermal power is performed by the unit's digital data processing system (DDPS). The hourly log in addition to showing the result also echoes the variable data input to the calculation. That set of data when augmented by observation of steam generator and pressurizer levels is sufficient to perform an independent analysis of thermal power using the microcomputer program TPDCER2 (Thermal Power Determination in Combustion Engineering Reactors). That program has yet to be formally documented, but it is similiar to the program documented in NUREG-1167, TPDWR2: Thermal Power Determination in Westinghouse Reactors, Version 2.

To configure TPDCER2 for use with each unit, unit-specific parameters are required. To obtain the required data, the following references were reviewed:

a. Updated Final Safety Analysis Report, St. Lucie Unit 1, January 22, 1986

b. Updated Final Safety Analysis Report, St. Lucie Unit 2, April, 1986,

c. C.E. Book No. 74267 (12/72), Steam Generators, St. Lucie Plant Unit No. 1, January 22, 1986, and

d. C.E. Book No. 71272 (10/77), Steam Generators, St. Lucie Plant Unit No. 2.

Subsequently, it was determined that the FSAR descriptions of reactor coolant pump power and efficiency were not adequate, and those parameters were replaced by values determined during preoperational testing and used in the DDPS. Values for insulation surface area and conductivity were manipulated to force agreement with the heat losses determined during preoperational testing and used in the DDPS, hence the warning messages on attachments 5 and 6. The plant parameters used in the final analyses are shown in attachments 5 and 6 for Units 1 and 2, respectively.

With the cooperation of the licensee, the inspector was able to obtain four sets of data at 15 minute intervals from each DDPS. At the same time, the steam generator and pressurizer levels were recorded by hand from main control board instrumentation. Each analysis performed using TPDCER2 used two sets of data taken 15 minutes apart. Thus, two analyses were performed on each reactor resulting in four power determinations for each. The data from the DDPS were, in most cases, in different units from those required by TPDCER2, and it was necessary to set up a SUPERCALC 3 spreadsheet to make the required conversions reliably. Clearly, the versatility and flexibility of TPDCER2 would be much enhanced if it could accept a variety of units for the input.

The results of the eight individual power calculations from TPDCER2 were consistently higher than those reported by the DDPS, by from 5.8 to 9.1 megawatts thermal, but, in perspective, the worst disagreement was only 0.34% of the calculation. A review of the outputs did not reveal any obvious source of the disagreement. The results of a typical set of calculations by TPDCER2 are given in attachment 7.

It was concluded that the licensee's method of calculating thermal power to assure conformance to the license limit is acceptable.

7. Followup of Outstanding Items (92701)

(Closed) Inspector followup item 335/85-28-01: Discuss the monotonic change in the Unit 1 reactivity deviation. A review of the licensee's correspondence files confirmed that there had been continuing discussion with the fuel vendor on the issue of reactivity deviation, and that the vendor had submitted new prediction curves during the cycle. The licensee ascribed part of the vendor's problem in prediction with lack of experience in analysing the performance of boron carbide burnable poison rods. This item is closed.

(Closed) Unresolved item 335/389/85-28-03: The adequacy of the at-power moderator temperature coefficeient procedure as written and approved is in question. The procedure originally reviewed was OP 320051 (Revision 0). Since that time the licensee has revised the procedure and issued Revision 1 on March 13, 1986. The revised procedure contains adequate guidance on the collection and analysis of the required data. Based upon this observation and discussions with the licensee, this item is closed.

Attachments:
1. RCSLK9 Parameter List - Unit 1
2. RCSLK9 Results - Unit 1
3. Fit of 13 Observations
4. Fit of 10 Observations
5. TPDCER2 Plant Parameters - Unit 1
6. TPDCER2 Plant Parameters - Unit 2
7. TPDCER2 Heat Balance Data - Unit 2 (4 pages)

ATTACHMENT 1

PARAMETER LIST

Unit Identification:	
Plant Name	ST. LUCIE
Unit Number	1
Docket Number	50-335
Nuclear Steam System Supplier	Combustion Engineering
Vessel and Piping:	
Volume	9218 cubic feet
Pressurizer:	
Level Units	%
Temperature Compensated	No
Calibration Curve	
Slope	559.05 pounds per %
Upper Level Limit	96.5 %
Lower level Limit	10.1 %
Relief	Quench Tank
Volume Control Tank:	
Level Units	%
Calibration Curve	
Slope	281.5 pounds per %
Upper Level Limit	100 %
Lower level limit	0 %
Geometric Method Available	No
Drain Tank:	
Level Units	%
Calibration Curve	
Slope	145 pounds per %
Upper Level Limit	60 %
Lower level limit	20 %
Geometric Method Available	No
Quench Tank:	
Level Units	%
Calibration Curve	
Slope	137.44 pounds per %
Upper Level Limit	60 %
Lower level limit	50 %
Geometric Method Available	No

ATTACHMENT 2

NRC

INDEPENDENT MEASUREMENTS PROGRAM

REACTOR COOLING SYSTEM LEAK RATES

STATION: ST. LUCIE	TEST DATE : 4 November 1986
UNIT : 1	START TIME: 0800
DOCKET : 50-335	DURATION 3.25 hours

TEST DATA

	Initial	Final
System Parameters		
Pressure, psia	2250	2250
T Ave, degrees F	571.75	571.5
Water Levels		
Pressurizer, %	67	66.8
Quench Tank, %	53.5	60.5
Volume Control Tank, %	50.5	51
Drain Tank, %	27	28.5

Water Charged = 210 gal   Water Drained = 0 gal

TEST RESULTS

Change in Water Inventory in pounds:

Vessel & Piping	166	Quench Tank (1)	962
Pressurizer	-112	Drain Tank (1)	218
Volume Control Tank (1)	141		-------
Less: Water Charged	1748	Collected Leakage	1180
Plus: Water Drained	0		

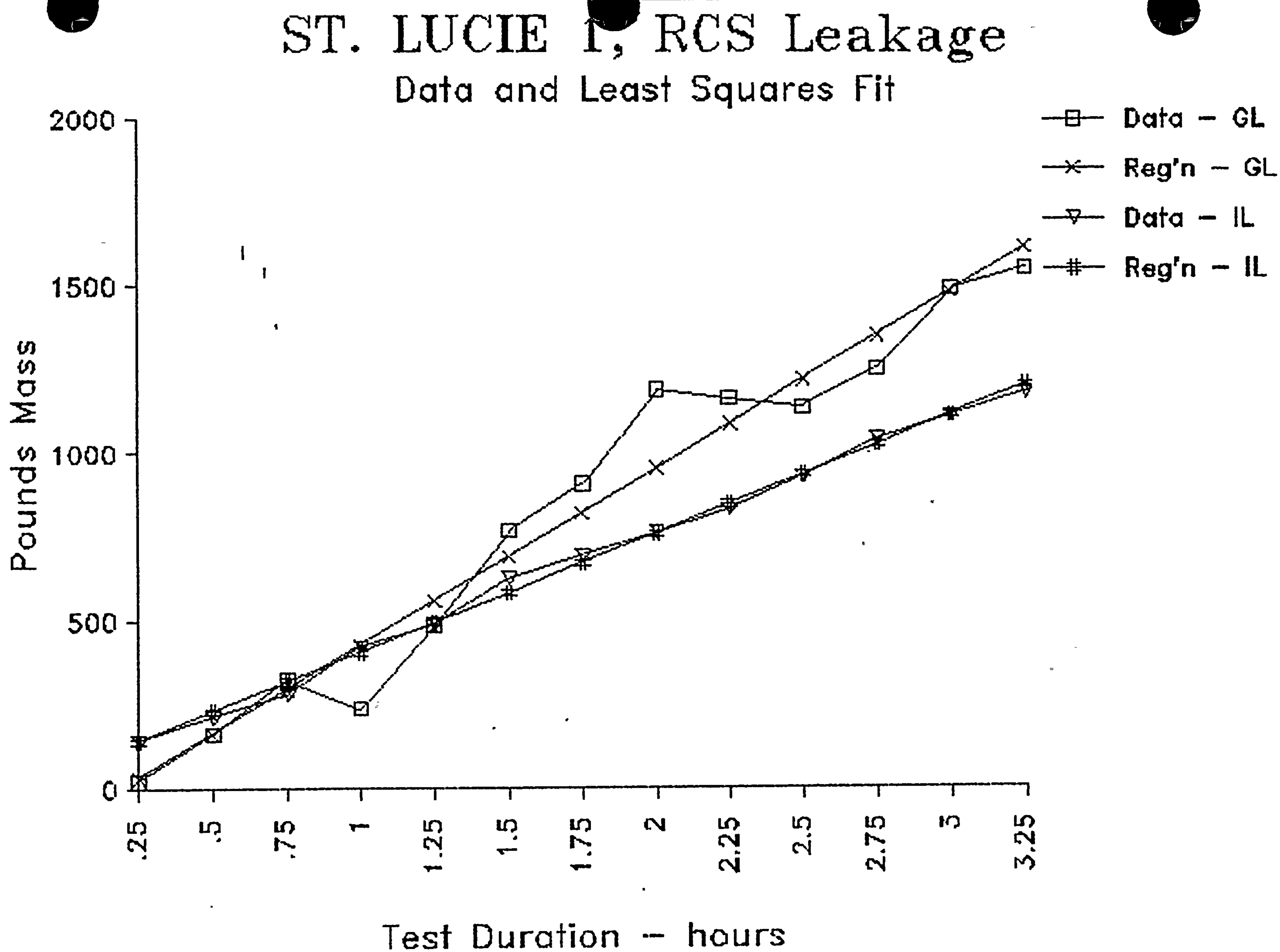
Cooling System	-1553		

Leak Rates in gpm (3):

Gross	0.96
Identified	0.73
Unidentified	0.23

(1) Determined from tank calibration curve.
(2) Determined from tank dimensions.
(3) The density used for converting inventory change to leak rate was 62.31 pounds/cubic foot based on standard conditions.

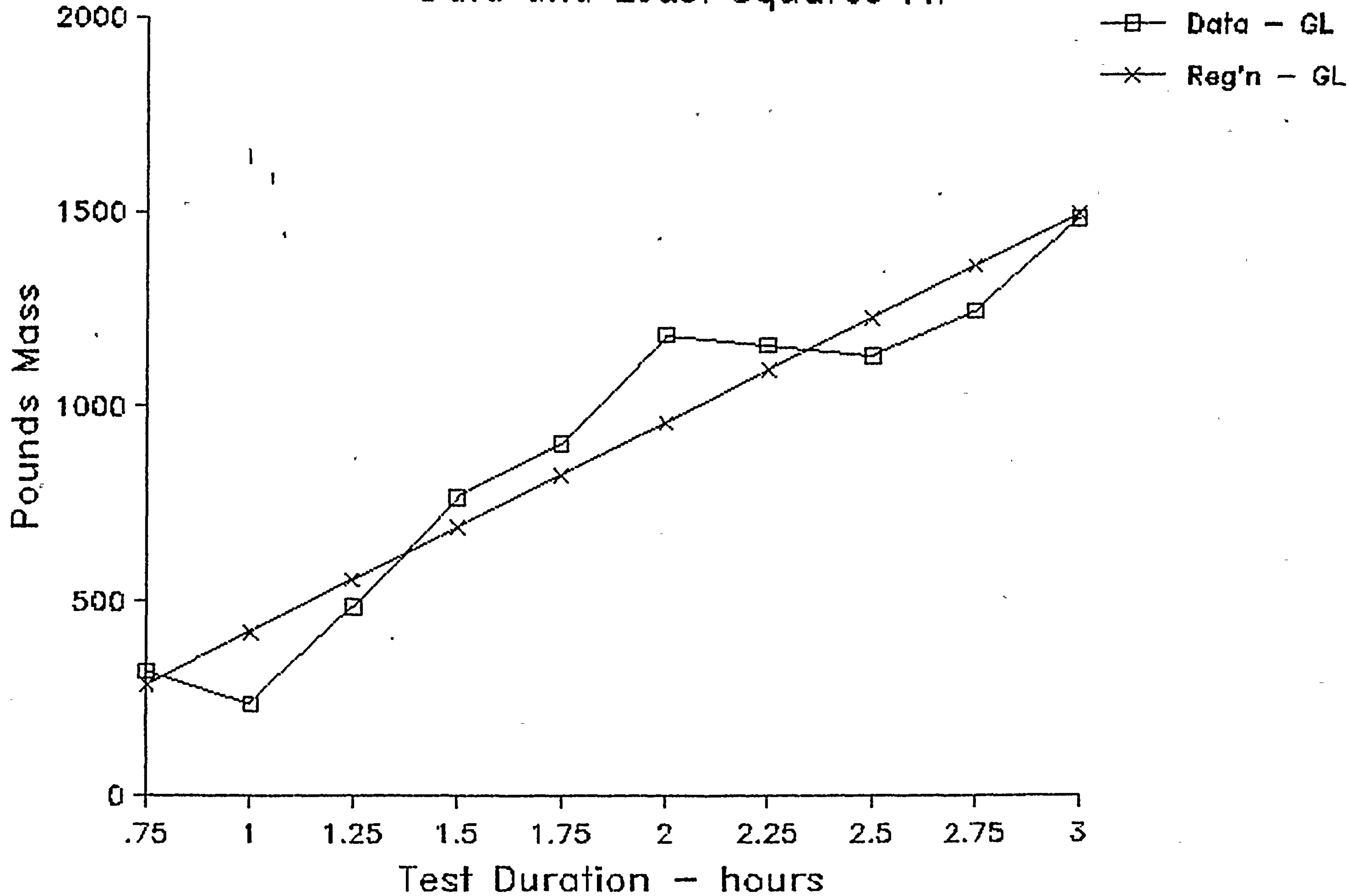
ATTACHMENT 3



ATTACHMENT 4

ST. LUCIE 1, RCS Leakage

Data and Least Squares Fit



ATTACHMENT 5

Plant Parameters

Identification

Plant Name	ST. LUCIE 1	
Unit Number	1	

Reactor Coolant System

Piping and Components

Number of coolant loops	2	(2/3)
Pump power	4.53	MW each
Pump efficiency	93.3	percent
Pressurizer inner diameter	95.3125	inches

Reflective Thermal Insulation

Surface area	13350	sq ft
Heat loss coefficient	W 240	BTUs/hr-sq ft

Nonreflective Thermal Insulation

Surface area	8800	sq ft
Thickness	3	inches
Thermal conductivity	.39	BTUs/hr-ft-F

Steam Generators

Moisture carry-over	.2	percent
Dome inside diameter	232.5	inches
Low level downcomer O.D.	156.25	inches
High level downcomer O.D.	230.125	inches
Low water level	79.656	inches
High water level	155.656	inches

W - Warning, Data or Parameters preceded by W are suspicious or in error.

ATTACHMENT 6

Plant Parameters

Identification

Plant Name	ST. LUCIE
Unit Number	2

Reactor Coolant System

Piping and Components

Number of coolant loops	2	(2/3)
Pump power	4.25	MW each
Pump efficiency	99	percent
Pressurizer inner diameter	95.5625	inches

Reflective Thermal Insulation

Surface area	13350	sq ft
Heat loss coefficient	W 240	BTUs/hr-sq ft

Nonreflective Thermal Insulation

Surface area	8800	sq ft
Thickness	3	inches
Thermal conductivity	.39	BTUs/hr-ft-F

Steam Generators

Moisture carry-over	.2	percent
Dome inside diameter	232.5	inches
Low level downcomer O.D.	156.375	inches
High level downcomer O.D.	231.18	inches
Low water level	79.656	inches
High water level	155.656	inches

W - Warning, Data of Parameters preceded by W are suspicious or in error.

ATTACHMENT 7

Heat Balance Data, Data Set 1

Time	951	0000-2400 hours
Letdown Line		
Letdown Flow	34.27	gpm
Letdown Temperature	195.6	deg. F
Charging Line		
Charging Flow	43.07	gpm
Charging Temperature	110.1	deg. F
Pressurizer		
Pressure	2250	psia
Water Level	201	inches
Reactor		
T ave	573.88	deg. F
T cold	549	deg. F
Steam Generator A		
Steam pressure	859.8	psia
Feedwater Flow	5.869	E6 lb/hr
Feedwater Temperature	437.1	deg. F
Surface Blowdown	0	gpm
Bottom Blowdown	24.48	gpm
Water level	123	inches
Moisture carry-over	.2	percent
Steam Generator B		
Steam pressure	868.8	psia
Feedwater Flow	6.04	E6 lb/hr
Feedwater Temperature	435.9	deg. F
Surface Blowdown	0	gpm
Bottom Blowdown	20.21	gpm
Water level	117	inches
Moisture carry-over	.2	percent

Heat Balance Data, Data Set 2

Time	1005	0000-2400 hours

Letdown Line

Letdown Flow	35.09	gpm
Letdown Temperature	195.3	deg. F

Charging Line

Charging Flow	43.47	gpm
Charging Temperature	110.1	deg. F

Pressurizer

Pressure	2250	psia
Water Level	201	inches

Reactor

T ave	573.88	deg. F
T cold	548.95	deg. F

Steam Generator A

Steam pressure	859.8	psia
Feedwater Flow	5.868	E6 lb/hr
Feedwater Temperature	437	deg. F
Surface Blowdown	0	gpm
Bottom Blowdown	25.17	gpm
Water level	123	inches
Moisture carry-over	.2	percent

Steam Generator B

Steam pressure	868.4	psia
Feedwater Flow	6.06	E6 lb/hr
Feedwater Temperature	435.8	deg. F
Surface Blowdown	0	gpm
Bottom Blowdown	20.43	gpm
Water level	117	inches
Moisture carry-over	.2	percent

ATTACHMENT 7

TPDCER2 HEAT BALANCE

Plant Name : ST. LUCIE   Unit No.: 2

Docket No. : 50-389   Date : 11/06/86

DATA SET 1 0951 hours	ENTHALPY (BTUs/lb)	FLOW (E6 lb/hr)	POWER (E9 BTUH)	POWER (MWt)
STEAM GENERATOR A				
Steam	1196.3	5.859	7.009	
Feedwater	416.1	-5.869	-2.442	
Surface Blowdown	520.1	0.00000	0.00000	
Bottom Blowdown	466.6	0.00983	0.00458	
Power Dissipated			4.5715	1339.8
STEAM GENERATOR B				
Steam	1196.0	6.032	7.214	
Feedwater	414.8	-6.040	-2.506	
Surface Blowdown	521.6	0.00000	0.00000	
Bottom Blowdown	466.6	0.00811	0.00379	
Power Dissipated			4.7125	1381.1
OTHER COMPONENTS				
Letdown Line	168.8	0.01667	0.00281	
Charging Line	83.9	-0.02152	-0.00181	
Pressurizer	611.5	-0.00002	-0.00001	
Pumps			-0.05742	
Insulation Losses			0.00971	
Power Dissipated			-0.04672	-13.7
REACTOR POWER				2707.3

ATTACHMENT 7

DATA SET 2 1005 hours	ENTHALPY (BTUs/lb)	FLOW (E6 lb/hr)	POWER (E9 BTUH)	POWER (MWt)
STEAM GENERATOR A				
Steam	1196.3	5.858	7.008	
Feedwater	416.0	-5.868	-2.441	
Surface Blowdown	520.1	0.00000	0.00000	
Bottom Blowdown	466.5	0.01010	0.00471	

Power Dissipated			4.5712	1339.7
STEAM GENERATOR B				
Steam	1196.0	6.052	7.238	
Feedwater	414.7	-6.060	-2.513	
Surface Blowdown	521.5	0.00000	0.00000	
Bottom Blowdown	466.5	0.00820	0.00383	

Power Dissipated			4.7288	1385.9
OTHER COMPONENTS				
Letdown Line	168.5	0.01707	0.00288	
Charging Line	83.9	-0.02172	-0.00182	
Pressurizer	611.5	-0.00002	-0.00001	
Pumps			-0.05742	
Insulation Losses			0.00971	

Power Dissipated			-0.04667	-13.7
REACTOR POWER				2712.0