Mr. Thomas F. Plunkett
President, Nuclear Division
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408-0420

July 16, 1996

SUBJECT: REQUEST FOR INFORMATION CONCERNING STEAM GENERATOR TUBE INTEGRITY ANALYSIS - ST. LUCIE PLANT, UNIT 1 (TAC NO. M95230)

Dear Mr. Plunkett:

As a result of the NRC staff's review of steam generator inspection and repair criteria for St. Lucie Unit 1, you provided additional information concerning these criteria in a letter dated June 25, 1996. Based on a review of the material provided, the NRC staff has several comments and questions which are specified in the enclosure to this letter. Your response to these comments and questions should be included in the probabilistic tube integrity analysis, scheduled to be completed in September 1996. This information, in part, is necessary for the staff to adequately perform an independent evaluation of this analysis.

Sincerely,

Original signed by

Leonard A. Wiens, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosure: Staff Review Comments

cc w/enclosure: See next page

DF01 1/1

DISTRIBUTION

Docket File	J. Zwolinski
PUBLIC	OGC
St. Lucie Rdg.	ACRS
S. Varga	J. Johnson

DOCUMENT NAME: G:\STLUCIE\SLLTRSG.RAI

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PDII-3/LA		PDII-3/PM		PDII-3/D			
NAME	BClayton		LWiens		FHebdon			
DATE	07/15/96		07/16/96		07/16/96			

OFFICIAL RECORD COPY

Mr. T. F. Plunkett
Florida Power and Light Company

St. Lucie Plant

cc:
Jack Shreve, Public Counsel
Office of the Public Counsel
c/o The Florida Legislature
111 West Madison Avenue, Room 812
Tallahassee, Florida 32399-1400

Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
7585 S. Hwy A1A
Jensen Beach, Florida 34957

Joe Myers, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

J. R. Newman
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, DC 20036

John T. Butler, Esquire
Steel, Hector and Davis
4000 Southeast Financial Center
Miami, Florida 33131-2398

Mr. Thomas R.L. Kindred
County Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

Mr. Charles Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering, Nuclear Power
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. Bill Passetti
Office of Radiation Control
Department of Health and Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Regional Administrator,
Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W. Suite 2900
Atlanta, Georgia 30323

H. N. Paduano, Manager
Licensing & Special Programs
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

J. A. Stall, Site Vice President
St. Lucie Nuclear Plant
P. O. Box 128
Ft. Pierce, Florida 34954-0128

J. Scarola
Plant General Manager
St. Lucie Nuclear Plant
P.O. Box 128
Ft. Pierce, Florida 34954-0128

Mr. Kerry Landis
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W. Suite 2900
Atlanta, Georgia 30323-0199

E. J. Weinkam
Licensing Manager
St. Lucie Nuclear Plant
P.O. Box 128
Fort Pierce, Florida 34954-0128

Mr. Thomas J. Saporito
National Litigation Consultants
6230 W. Indiantown Road #7-355
Jupiter, Florida 33458

STAFF COMMENTS AND QUESTIONS RELATING TO
TUBE INTEGRITY ANALYSIS

1. Clarify how free span indications which are not crack-like and which can not be traced back to the preservice baseline inspection were dispositioned.

2. Discuss the extent to which the data used in the sizing qualification program have similar morphologies and eddy current responses (e.g., extent of intergranular attack, cracking, voltage responses, noise levels, etc.).

   Discuss the basis for including data from different frequencies in the qualification program (400 kHz and 560 kHz) given that the ability to reliably size defects will, in part, depend on the frequency.

3. In the June 25, 1996, response to NRC Request 3, it was indicated that the burst pressure correlation that will be used relies on average crack depth. The qualification data for the sizing program relies on maximum crack depth. Discuss how the uncertainties associated with converting from maximum to average crack depth will be addressed (e.g., using a curve similar to that in Figure 3 of Attachment A to the June 25, 1996 letter). Provide the supporting metallographic and eddy current data to support this approach.

   Discuss how the uncertainties in the predicted burst pressure will be accounted for in the probabilistic methodology (i.e., the predicted burst pressure does not exactly match the observed burst pressure as is illustrated in Figure 1 of Attachment A).

   In Figures 4, 5, and 6 of Attachment A to your June 25, 1996 letter, information regarding the distribution of material properties expected in the St. Lucie Unit 1 steam generator tubes is provided. Discuss if this data is from steam generators tubes that have been in service. If not, provide the distribution of material properties based on destructive examination of pulled tubes from similarly fabricated Combustion Engineering steam generators. Provide the mean and standard deviation of this data along with the 95/95 confidence value (i.e., the lower tolerance limit).

   Discuss how the growth rate distribution was developed. Discuss how the average growth rate was determined.

   Clarify whether a lower 95% prediction interval curve for the burst pressure versus crack size correlation was used in the deterministic run-time evaluation. Use of the lower 95% prediction interval curve is consistent with the approach in Generic Letter 95-05. Please provide for comparative purposes a deterministic run-time evaluation assuming a lower 95% prediction interval curve for the burst pressure correlation adjusted for lower bound (95/95) material properties based on the destructive examination of inservice steam generator tubes. This analysis should also assume a 95% cumulative probability value for non-destructive examination uncertainty.

4. In the June 25, 1996, response to NRC request 4, EPRI report TR-104788 was cited. EPRI report TR-104788 contains guidelines for primary-to-secondary leakage monitoring programs. In this report, deviations from the guidelines are considered acceptable following plant-specific evaluations. Please clarify any significant deviations taken from these guidelines. In addition, specifically address the leakage limits/action levels to be implemented at St. Lucie Unit 1.

5. In the June 25, 1996, response to NRC Request 5, it was indicated that undetected flaws will be projected to give the end-of-cycle (EOC) distribution of through-wall cracks. Discuss whether the detected flaws will also be included in this projection. Discuss the need to include in the leakage analysis indications which are near through-wall and will pop-through the wall under postulated accident conditions. Discuss how nondestructive examination (NDE) uncertainty will be accounted for in the projection of the EOC distribution.

   The staff has accepted a value for the probability of detection which is independent of flaw depth of 0.6. Clarify the value of the probability of detection that will be used in your analyses for predicting the EOC distribution of indications. If different than the 0.6 value discussed above, provide a sensitivity study using this value.

   Clarify whether the analyses to be performed to determine the EOC distribution will start from a beginning of cycle distribution which has been adjusted for the probability of detection and the number of indications repaired similar to the methodology described in Generic Letter 95-05.

6. Discuss and provide the qualification data for the leakage model that will be implemented. Discuss how the results from the in-situ pressure tests will be factored into the leakage model.

7. Provide tabularized and graphical data for the distribution of indications detected (length and depth) for each steam generator, the distribution of indications repaired for each steam generator, the growth rate of indications (length and depth) for each steam generator, the material properties distribution, the NDE uncertainty models, and the burst pressure correlation. For the distribution of indications detected and repaired, the growth rate distribution, and the material properties distribution, provide the number of indications with depths/lengths/material properties within a given interval (e.g., 10 indications with depths between 35% and 40% through-wall). For the remaining distributions, provide the data in a format suitable for the staff to independently verify the results of the tube integrity analysis.

8. In the June 25, 1996, response to NRC Request 7, it is stated that the "burst pressure will be treated deterministically." Clarify what is meant by this statement.