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Docket No. 50-250

Florida Power & Light Company  
ATTN: Dr. R. E. Uhrig  
Vice President  
P. O. Box 013100  
Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. 27 to Facility Operating License No. DPR-31 for the Turkey Point Nuclear Generating Unit No. 3. The amendment is in response to your application dated August 5, 1977, as supplemented by your letters of August 10, 1977 and August 12, 1977.

This operating license amendment permits continued operation of Turkey Point Unit No. 3 until the next refueling under new restrictions on allowable steam generator leakage and iodine concentrations. During the shutdown for refueling, or at an earlier time in the event the limitations 2 and 3 in the amended Facility Operating License, Paragraph 3.E are exceeded, the steam generators are to be inspected. Details of the proposed inspection program are to be submitted by September 15, 1977. The inspection procedures are to be developed by the licensee considering the items presented in the enclosed Safety Evaluation and your letters of June 16, 1977 and August 12, 1977.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

George Lear, Chief  
Operating Reactors Branch #3  
Division of Operating Reactors

Enclosures:

- 1. Amendment No. 27
- 2. Safety Evaluation
- 3. FEDERAL REGISTER Notice

cc: See next page

SEE PREVIOUS YELLOW FOR CONCURRENCE

*Constructive*  
*B*

OFFICE	ORB#3	ORB#3	OELD	ORB#3	
SURNAME	R. Diggs for C. Parrish	R. Clark		G. Lear	
DATE	8/12/77	8/15/77	8/17/77	8/10/77	

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 ATTN: Dr. R. E. Uhrig  
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OFFICE >	ORB#3	ORB#3	OELD	ORB#3		
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Docket No. 50-250

Florida Power & Light Company  
 ATTN: Dr. R. E. Uhrig  
 Vice President  
 P. O. Box 013100  
 Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-31 for the Turkey Point Nuclear Generating Unit No. 3. The amendment is in response to your application dated August 5, 1977, as supplemented by your letter of August 10, 1977.

This operating license amendment permits continued operation of Turkey Point Unit No. 3 until the next refueling under new restrictions on allowable steam generator leakage and iodine concentrations. During the shutdown for refueling, the steam generators are to be inspected. Details of the proposed inspection program are to be submitted by September 15, 1977. The inspection procedures are to be developed by the licensee considering the items presented in the enclosed Safety Evaluation and your letter of June 16, 1977.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

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 Operating Reactors Branch #3  
 Division of Operating Reactors

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SURNAME >	<i>CParrish</i>	RClark:acr		GLear		
DATE >	8/12/77	8/ /77	8/ /77	8/ /77		

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-250

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY  
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Facility Operating License No. DPR-31, issued to Florida Power and Light Company, for operation of the Turkey Point Nuclear Generating Unit No. 3 located in Dade County, Florida. The amendment is effective as of the date of issuance.

The amendment to the operating license will allow continued operation of Turkey Point Unit No. 3 until the next refueling (presently scheduled for January 1978) at which time the steam generators shall be inspected.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 5, 1977, as modified by letter dated August 10, 1977, (2) Amendment No. 27 to License No. DPR-31, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. and at the Environmental & Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16 day of August 1977.

FOR THE NUCLEAR REGULATORY COMMISSION



George Leav, Chief  
Operating Reactors Branch #3  
Division of Operating Reactors

cc:

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Bureau of Intergovernmental Relations  
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Honorable Dewey Knight  
County Manager of Metropolitan  
Dade County  
Miami, Florida 33130

Florida Power & Light Company  
ATTN: Mr. Henry Yaeger  
Plant Manager  
Turkey Point Plant  
P. O. Box 013100  
Miami, Florida 33101

Chief, Energy Systems Analysis Branch (AW-459)  
Office of Radiation Programs  
U. S. Environmental Protection Agency  
Room 645, East Tower  
401 M Street, S. W.  
Washington, D. C. 20460

U. S. Environmental Protection Agency  
Region VI Office  
ATTN: EIS COORDINATOR  
345 Courtland Street, N. E.  
Atlanta, Georgia 30308

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27  
License No. DPR-31

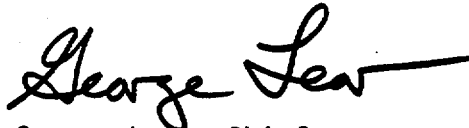
1. The Nuclear Regulatory Commission (the Commission) has found that:
    - A. The application for amendment by Florida Power and Light Company (the licensee) dated August 5, 1977, as modified on August 10, 1977 and August 12, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
    - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
    - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
    - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
    - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
-

2. Accordingly, paragraph 3.E of Facility Operating License No. DPR-31 is hereby amended to read as follows:

"E. Steam Generator Inspections

1. The concentration of radioiodine in the primary coolant shall be limited to 1.0 microcurie/gram during normal operation and to 30 microcuries/gram during power transients.
  2. Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm per steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours. The leaking tube(s) shall be evaluated and plugged prior to resuming power operation, if leaking is not attributable to the denting phenomena.
  3. Reactor operation shall be terminated and Nuclear Regulatory Commission approval shall be obtained prior to resuming operation if primary to secondary leakage attributable to the tube denting phenomena is detected from two or more tubes in the plant in any 20-day period.
  4. Unit No. 3 steam generators shall be inspected during the next refueling outage or sooner in the event the limitations of 2 and 3, above, are exceeded. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection.
3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



George Lear, Chief  
Operating Reactors Branch #3  
Division of Operating Reactors

Date of Issuance: August 16, 1977





UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 27 TO LICENSE NO. DPR-31,

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-250

Introduction

License Amendment No. 22 to Facility Operating License No. DPR-31 for Turkey Point Unit No. 3 required that Unit No. 3 be shutdown within six months of equivalent operation from January 14, 1977 (shutdown approximately July 16, 1977) for the purpose of steam generator inspection. After consideration of the performance record for Unit No. 3 and the then shutdown status of Unit No. 4, the Commission, on July 15, 1977, issued License Amendment No. 26 to Facility Operating License No. DPR-31 which allowed continued operation of Unit No. 3 until August 16, 1977.

Since the beginning of Cycle 4 operation for Unit No. 3, no steam generator tubes have had to be plugged as a result of primary to secondary leakage. Based on this operating performance of Unit No. 3, Florida Power and Light Company (FPL) proposed in a letter dated August 5, 1977 a License Amendment to allow continued operation of Unit No. 3 until the next refueling outage (scheduled for January 1978) at which time the steam generators

would be inspected. Following discussions with the staff, FPL proposed, by letter dated August 10, 1977, a restriction on maximum primary to secondary leakage through the steam generator tubes and on maximum radioiodine concentrations in the primary coolant during the remainder of Cycle 4 operations.

In the course of our review, we have found it necessary to modify the requested license change. These modifications were discussed with the licensee and he concurs.

#### Generic Background

##### Water Chemistry

For many years a sodium phosphate treatment for PWR secondary coolant was widely used for U-tube design steam generators that removed precipitated or suspended solids by blowdown. It was successful as a scale inhibitor, however, in the early use, many PWR U-tubed steam generators with Inconel-600 tubing experienced stress corrosion cracking. The cracking was attributed to free caustic which can be formed when the  $\text{Na}/\text{PO}_4$  ratio exceeds the recommended limit of 2.6. In addition, some of the insoluble metallic phosphates, formed by the reaction of sodium phosphates with the dissolved solids in the feed-water, were not adequately removed by blowdown. These precipitated

phosphates tended to accumulate as sludge on the tube sheet and tube supports at the central portion of the tube bundle where restricted water flow and high heat flux occurs. Phosphate concentration (hideout) at crevices in areas of the steam generator, noted above, caused localized wastage resulting in thinning of the tube wall. The problem of stress corrosion cracking was corrected by maintaining the  $\text{Na}/\text{PO}_4$  ratio between 2.6 and 2.3. Although the recommended  $\text{Na}/\text{PO}_4$  ratio was maintained, it did not correct the phosphate hideout problem that caused wastage of the Inconel-600. Largely to correct the wastage and caustic stress corrosion cracking encountered with the phosphate treatment, most PWRs with a U-tube design steam generator using a phosphate treatment for the secondary coolant have now converted to an all volatile chemistry (AVT).

In 1975, radial deformation, or the so-called "denting," of steam generator tubes occurred in several PWR facilities after 4 to 14 months operation, following the conversion from a sodium phosphate treatment to an AVT chemistry for the steam generator secondary coolant. Tube denting occurs predominantly in rigid regions or so-called "hard spots" in the tube support plates. These hard spots are located in the tube lanes between the six rectangular flow slots in the support plates near the center of the tube bundle and around the peripheral locations of the support plate where the plate is wedged to the wrapper and shell. The hard spot areas do not contain the array of water circulation holes found elsewhere in the support plates.

The phenomenon of denting has been attributed to the accelerated corrosion of the carbon steel support plates at the tube/tube support plate intersections (annuli). The corrosion product (magnetite) from the carbon steel plate occupies approximately twice the volume of the material corroded. Thus, the continuing corrosion exerts sufficient compressive forces to dent the tube and crack the tube support plate ligaments between the tube holes and water circulation holes. As a result of the tube support plate deformation, the rectangular flow slots began to "hourglass;" i.e., the central portion of the parallel

flow slot walls have moved closer so that some of flow slots are now narrower in the center than at the ends.

#### U-Bend Cracks

On September 15, 1976, during normal operation, one U-tube in the innermost row parallel to the rectangular flow slots in steam generator A at Surry Unit No. 2 rapidly developed a substantial primary to secondary leak (about 80 gpm). After removal of the damaged tube and subsequent laboratory analysis, it was established that the leak resulted from an axial crack, approximately 4-1/4 inches in length, in the U-bend apex due to intergranular stress corrosion cracking that initiated from the primary side. Since the initial parallel flow slot wall in the top support plate has moved closer, the support plate material around the tubes nearest this central portion of these flow slots has also moved inward, in turn forcing an inward displacement of the legs of the U-bends at these locations. This inward movement of the legs of the U-bends at these locations caused increase in the hoop strain and ovality of the tubes at the U-bend apex. It is this additional increase in strain at the apex of the U-bend which is believed to be required to initiate stress corrosion cracking of the Inconel 600 alloy tubing exposed to PWR primary coolant.

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Laboratory examination of 71 U-bends removed from flow slot locations in rows 1, 2, and 3 of the Surry Units No. 1 and 2 and Turkey Point Unit No. 4 steam generators has shown that intergranular cracking at the U-bend apex was found only in the row 1 tubes.

Of the 71 tubes removed from these operating reactors, which are the most severely affected, no cracks have been found in tubes with computed equivalent strains less than 13.5% after approximately 11,065 hours of effective full power operation since detection of the first tube dent. However, this same equivalent operating time led to the tube failure at Surry Unit No. 2, where the equivalent strain was estimated to be >14.3%. This indicates a strain level at which rapid development of stress corrosion cracking may occur in U-bends

of steam generators of this design.

Recent test work also indicates that long incubation periods are needed for the development of stress corrosion cracking at some strain rates. 2-5/ Tests indicated that at 12.5% outer fiber strain, 1/ Inconel 600 U-bend specimens tested in high purity water at 650°F took a long incubation time (>12,000 hours) for the nucleation of an intergranular crack, longer time 13,000 hours for >30% penetration and more than 18,000 hours to fail.

Although these test results are not directly applicable to the PWR steam generator tubing at Turkey Point, they do confirm the observed operating experience that (1) a long incubation time is required to initiate intergranular cracking in Inconel 600 material, and (2) a high strain is required for crack propagation.

In this regard, the staff requested that the licensees of affected plants address the following concern:

"Hourglassing" may continue and close the flow slots in the top support plate increasing the strain at the U-bend apex of the tubes in rows 2 and beyond.

In response to this concern and to supplement plugging of row 1 tubes, for example, VEPCO has installed stainless steel 304 alloy blocks in each of the six flow slots in the top support plate of all three Surry Unit No. 1 steam generators. These blocks prevent further closure of the flow slots and the inward displacement of the legs of the U-bends, thereby preventing further anticlastic straining at the U-bend apex of these tubes in rows 2 and beyond. As a result, intergranular stress corrosion cracking at the U-bends of tubes in row 2 and beyond is not anticipated during near term (next year) normal operation. However, the flow slot blocking devices or the complete closures of flow slots would cause: (1) an increase in strain in the support plate, (2) peripheral expansion of the support plate between wedge locations, (3) an increase of tube denting in the "hard spot" regions, and (4) additional bearing

stresses on the wedges, wrapper, channel, and steam generator shell due to the peripheral expansion of the support plate. The net overall effect of flow slot blocking devices would be similar to complete closure of the flow slots.

#### Support Plate Expansion

Continued growth of the magnetite in the tube-tube support plate annuli results in a non-uniform increase in strain in the support plates and corresponding in-plane expansion. In this regard, the staff requested that the licensees of affected plants address the following concerns:

- "1. Severe cracking of the support plate may result due to the continuing in-plane expansion of the support plate.
2. The rate of in-plane expansion in any support plate could increase the severity of tube denting in "hard spot" regions. Severe denting would restrain the tubes in the support plate and the plate may have a tendency to buckle or otherwise deform and thus exert additional bending loads on tubes.
3. With the closure of all the flow slots in any one support plate additional loads could be transmitted (due to the in-plane expansion of the plate) to the wedges, wrapper, channel spacer, tubes, and the steam generator vessel.
4. Thermal-hydraulic performance could be affected with the closure of all the flow slots in any support plate."

#### Anti-vibration Bar Fretting

On November 17, 1976 Southern California Edison Company (SCEC) reported to I&E, Region V, that, during the inspection of the San Onofre Unit No. 1 steam generators, excessive wear or mechanical fretting of anti-vibration bars was found in one of the steam generators. A failure of these bars could result in excessive flow induced vibration that might affect tube integrity, especially for those plants where the tube denting phenomenon was observed at the top support plate. Subsequent investigation revealed that the anti-vibration bar design of San Onofre Unit No. 1 and Connecticut Yankee is unique in comparison with other

Westinghouse plants. Differences in the design are summarized as follows:

- a. Materials - carbon steel for San Onofre Unit 1 and Connecticut Yankee; Inconel 600 for new models (44 and 51).
- b. Bar Cross-section - 3/8 inch round bars; changed to square bars in the new models.
- c. Clearances - (L-35 mils); was changed to (L-20 mils) for new models where L is the tube spacing.
- d. Changes in V-bar configuration and spacing.

#### DISCUSSION

Subsequent to the U-bend failure occurrence of September 15, 1976, at Surry Unit No. 2, Turkey Point Unit No. 3 was shutdown in December, 1976 to perform an inspection of the conditions of the top support plates and U-bends. The purpose of the inspection was to determine whether similar conditions existed at Turkey Point Unit No. 3 that had led to the U-bend failure at Surry Unit No. 2. By letters dated December 21, 22, and 30, 1976 and January 3, 1977, Florida Power and Light Company (the licensee) submitted results concerning the steam generator inspections, and additional information in response to the staff's concern over the cause and the consequences of continuing upper support plate deformation at flow slot locations. The NRC Staff's Safety Evaluation for License Amendment No. 22 to DPR-31, dated January 14, 1977 concluded that:

1. Tubes in rows 2 through 5 and outward in all the steam generators of Turkey Point Unit 3 would retain sufficient integrity to withstand normal operating and postulated accident conditions.
2. There was reasonable assurance of tube integrity to provide adequate protection to the public health and safety.
3. Turkey Point Unit 3 should be inspected at 5-6 month intervals during the next fuel cycle to assess the magnitude and consequences of tube support plate deformation.
4. Turkey Point Unit 3 operation beyond the 6 month interval should be dependent on the staff's ongoing evaluation of forthcoming

information from facilities with denting and deformation of the tube support plates.

Based upon above staff conclusions, approval was granted on January 14, 1977 for Turkey Point Unit No. 3 to return to power operation. During the past seven (7) months since the restart in mid-January, Unit No. 3 has not had any primary to secondary steam generator tube leakage and consequently no tubes have been plugged. In the same period, a large amount of field data has been obtained from Surry Units Nos. 1 & 2, and Turkey Point Unit No. 4, and analytical methods developed to correlate the field measurements of tube denting and flow slot deformation and the predicted behavior of tube support plates and tubes that have been subjected to the continuing growth of magnetite in the tube/tube support plate crevices. The following paragraphs summarize some of the important findings, both analytical and experimental, from other affected units that are pertinent to the conditions at Turkey Point Unit No. 3:

#### Potential for U-Bend Cracking

As a part of the submittal on June 9, 1977, the licensee provided an additional analysis to show that the residual hoop and axial stresses for tubes in rows 2 and beyond decrease as the U-bend radius increases. The magnitude of the residual stresses are dependent on the degree of cold working of the tube cross-section during the forming of the small radius U-bend. Bending of small radius U-bend tubes causes their cross-section to ovalize. However, the presence of an internal ball mandrel during bending of tubes for rows 1 and 2 forces the cross-section back into a configuration with considerable less ovality. Upon removal of the ball, the cross-section rebounds elastically and the residual hoop stresses on the inside surface of the tube are tensile at the top and bottom positions of the U-bend apex. Bending of tubes for rows greater than 2 does not employ the ball mandrel. Hence those tubes adopt the characteristic oval shape after bending, the tube cross-section rebounds elastically and the corresponding residual hoop stress for tubes in row 3 and beyond are compressive. The degree of residual tube ovality is a function of the bend radius, with the greatest



ovality associated with the tubes with the smallest bend radius that were bended without the internal ball mandrel, i.e., row 3.

The analyses indicate that the residual hoop stresses at the U-bend apex are equal to one half of the yield strength, or about 26,000 psi and the residual axial stress in small radius U-bend tubes is 15,000 psi. The magnitudes of these residual stresses are less than the ASME Code minimum yield strength. Therefore, with these low residual stresses at the U-bend apex, the potential for stress assisted intergranular attack at the U-bend for tubes in rows 2 and beyond is minimal. Furthermore, the fact that all row 1 tubes were already plugged should alleviate some of the concern over the potential for U-bend cracking. Since row 2 tubes in Unit 3 were not plugged during previous outage, concern over the potential U-bend failure for row 2 tubes is addressed separately in the Evaluation Section of this report.

#### "Islanding" Effect

As a consequence of the continuing growth of magnetite at the tube/tube support plate crevices, the support plate will continue to deform toward un-restrained regions such as at periphery and flow slots. Continuing plate deformation into flow slots will eventually crack at the edges of the slot, with the broken pieces moving into flow slots to form "islands". In the event this phenomenon were to occur, some inner row tube may lose lateral support(s) and, thus, may be subjected to flow induced vibration. The licensee has submitted results of a generic analysis of fluid structural vibrations considering loss of one, two and three lateral supports. The maximum vibration amplitude was calculated to slightly exceed one half of the gap between adjacent tubes for the case when three lateral supports were lost, with slight increase in bending stress in the tube.

### Typical Inspection Program

Of more significance in the long term is the need to carefully assess the condition of all three steam generators in Unit No. 3 and to determine, to the extent possible, the causes for possible occurrences of leakage in the facility. In order to do these, a certain amount of plant specific data has to be acquired through a coordinated inservice inspection program. The inspection program should be formulated such that sufficient amount of data can be obtained to adequately address various concerns pertaining to the integrity of the steam generators.

The NRC staff believes that, in general, the following items should be considered in an inspection program:

1. Standard eddy current techniques should be used to detect sludge accumulation on the tube sheet to determine possible areas where wastage could occur. If sludge lancing had been performed in the past, the possible areas will not be limited to the traditional "kidney" shape regions in the middle of the tube bundles.
2. A complete Regulatory Guide 1.83 type inspection should be performed.
3. An eddy current dent location program should be conducted using a standard 0.540 inch probe at reduced gain to measure the "average" dent size for sample tubes in all areas of the steam generators.
4. Tube gauging using at least three different probe sizes should be performed for all tubes in areas where severe denting was expected and confirmed by item 3 above. Specifically, tubes in areas that can be classified as hardspots such as those near the tube lane, wedges, patch plate plug weld, etc., should be gauged.
5. Flow slot deformations (hourglassing) should be measured for all three steam generators.
6. Wrapper to shell annulus measurements should be made, at least for one steam generator that has suffered worst hourglassing as confirmed by item 5.
7. Eddy current testing of U-bends for tubes in rows 2-5 should be made if flow slot deformations are measurable (item 5 above).

## EVALUATION

Prior to January 14, 1977, FPL inspected the Unit No. 3 steam generators and plugged all the row 1 tubes in each steam generator to prevent U-bend cracking of the tubes in row 1. FPL provided sufficient generic information to show that the likelihood of dented tubes developing leaks during six months operation was minimal during normal operation or postulated accidents. In addition there was no evidence that could lead to U-bend cracking in row 2 and beyond, and the fact that row 1 was plugged alleviated the generic concern about the small radius U-bend tubes. The NRC staff's Safety Evaluation for License Amendment No. 22 to DPR-31 concluded that (1) the condition of Turkey Point Unit No. 3 steam generators was not as severe as other PWR facilities with dented tubes (which have developed leaks), and (2) there was not an immediate potential during six months operation that would lead to similar tube failure which have occurred in other affected units to endanger the public health and safety. The six month operation was extended on July 15, 1977 until August 16, 1977 by Amendment No. 26

During the past seven months, there have not been any incidences of tube leakage in the Turkey Point Unit No. 3 steam generators. For this reason and in view of the NRC staff's ongoing evaluation of information from other facilities with denting and those that developed leaky dents, we conclude the following:

1. Further U-bend failures are not likely to occur for near term continued operation because of the following:
  - a. Laboratory examinations of 71 tubes removed from Surry Units No. 1 and 2 and Turkey Point Unit No. 4 steam generators indicate that cracking was confined only to row one tubes.
  - b. All the tubes in row one are plugged.
  - c. The susceptibility for U-bend cracking of tubes in row 2 would be substantially less because the residual hoop stress at the U-bend apex is 25-35% lower than the ASME Code minimum yield strength for Inconel 600 alloy tubing and the effective U-bend strain is 30-50% lower than in row 1.

- d. U-bend cracking in rows 3 and beyond will not occur because the residual hoop stresses on the inside surface of the tube are compressive at the top and bottom positions of the U-bend apex, and thus the potential for stress corrosion cracking is not possible.
2. The fact that all row 1 tubes are plugged lessens the concern over the possible loss of lateral support of tubes due to the so-called "islanding" effect.
3. Support plate expansion or continuing magnetite growth in the proposed period of operation will have insignificant effects on the wrapper and the steam generator vessel. Therefore, the wrapper and the vessel integrity during normal operating and accident conditions will not be affected by continued support plate expansion.

Due to the possible closure of the flow slots in the top support plates and the possible closure of flow slots in lower support plates, additional loads could be transmitted to the steam generator shell through the load path of the support plate, wedge, wrapper and channel spacer. Based on preliminary "crush" tests performed by Westinghouse the maximum load that can be developed along this load path is 60,000 pounds.

Analysis of the bearing stress along this path indicates that all stresses are less than the yield strength. Such stresses on the steam generator shell are highly localized and self limiting and will not adversely affect the integrity of the shell under accident conditions.

4. Since the total area of all six flow slots is only a small fraction of the total area for flow circulation, the effect of the possible flow slot closure or hourglassing on the thermal hydraulic performance of the steam generator will be negligible. There will be a slight decrease in the circulation ratio and the liquid flow velocities, with an increase in raw steam quality. But these are so small that they may be disregarded.

5. With regard to the anti-vibration bar degradation problem revealed during the inspection of San Onofre Unit No. 1 steam generators, there is no reason to believe that similar problems will occur at Turkey Point Unit No. 3 because there are basic differences in both the design and the material used. Anti-vibration bars in Turkey Point Unit No. 3 steam generator are made of Inconel 600 instead of carbon steel, of square cross-section instead of round, and have smaller clearances than those originally employed at San Onofre Unit No. 1.
6. Since the time that Turkey Point Unit No. 3 established AVT chemistry control, wastage and caustic stress corrosion cracking experience has been quite satisfactory. No substantial tube degradation from these corrosion mechanisms is expected to occur during normal operation.
7. At Turkey Point Unit No. 4, the licensee has recently investigated the possibility that 11 plugs in the 4B steam generator had worked loose during its last cycle of operation. As yet, no indications of the loose plugs or damage resulting therefrom, have been found by the licensee. We believe that similar experience will not occur during the proposed period of operation of Turkey Point Unit No. 3, prior to its shutdown for inspection of the steam generators, because there has been a sustained seven months period of operation during which there has been no indication of leakage from the currently plugged tubes in the Unit No. 3 steam generators. This performance provides the assurance of the integrity of the tube plugs.

#### Consideration of Continued Denting

With respect to the effect of continued magnetite growth which causes the support plate expansion and thus denting, the licensee has been unable to quantify the effects of tubes at intersections due to the continuing growth of magnetite. Therefore, concern over a possible increase in tube failures at the tube/support intersections cannot be completely alleviated. Also, because of the absence of an explicit

plugging criteria directed toward tubes subjected to increase plate strain, there is some concern that the complete integrity of some un-plugged dented tubes may not be maintained during postulated accidents. We do believe, however, that the probability of having tubes dented to such a severity is extremely low.

In addition, there are several factors which support an operating period with more stringent operating conditions while additional information is being generated; i.e., qualitative and preliminary quantitative integrity data, the low consequences of the relatively limited tube leakage that would be expected under postulated accident conditions and the very low probability of an initiating accident coincident with large numbers of tube failures.

The qualitative and preliminary quantitative integrity data is summarized as follows:

- a. Preliminary analyses of the support plate expansion (with a flow slot closure) indicated small hard spot strain increases and plate perimeter deformations.
- b. All leaks associated with dented tubes experienced to date have been small; e.g., well below the 0.3 GPM limit specified for Turkey Point Unit No. 4.
- c. Observed through-wall cracks in the dented regions in other affected plants, i.e., tube/support plate intersections, are constrained by the support plates; therefore, cracks with leakage rate equal to or less than 0.3 GPM, should not burst during postulated accidents.
- d. Through wall cracks at dented locations, with the amount of leakages experienced to date (less than 0.3 GPM), have been stable during normal operation (no rapid failures), and are not anticipated to become unstable during postulated accidents.

- e. Even though some non-through-wall cracks may exist and may crack through during postulated accidents, the associated leakage rate with such an event would be similar to that resulting from through wall crack found during normal operation and the crack would not be unstable. This consideration is consistent with the rationale upon which the preventive plugging limits were set for wastage or fretting type of degradation.
- f. Even if a LOCA or a MSLB were to occur during the proposed period of operation and that some tubes were in a state of incipient failure, the consequence of such an event will not be severe for Turkey Point Unit No. 3 (as discussed in the safety evaluation report, pages 13 through 15, attached to 2/8/77 NRC Order for Turkey Point Unit No. 4).

Because of the need to assure that any stress corrosion cracking which occurs during operation remains small and stable, and that an extensive number of tubes do not incur penetrating cracks or substantial part thru-wall cracks, certain additional operating limitations for the operation of Turkey Point Unit No. 3 have been developed and proposed by the licensee. These limitations are designed to assure the detection of the onset of tube degradation before it becomes a significant concern. The limits on allowable primary to secondary leakage should be required during the next period of operation.

In addition, we also considered the less probable event in which the steamline and steam generator tube failures occur at a time when the plant is operating with high coolant activity resulting from previous power level changes. As a result, we recommended and the licensee agreed that the license for Turkey Point Unit No. 3 be modified to limit the allowable primary coolant iodine activity, as part of the operational limitations. This will provide assurance that, were the steamline break with resulting tube leaks postulated to occur at such a time, the calculated doses would not exceed the 10 CFR Part 100 guidelines.

### Operational Limitations

1. A limit for primary to secondary leakage of 0.3 GPM will assure that no individual cracks will reach such proportions that it may become unstable during normal or accident loading conditions.
2. A substantial increase in the frequency at which leaking tubes are encountered could signal the development of more extensive general degradation. The potential for such a development during operation has been substantially alleviated by the limitations described below, requiring operation to be terminated in the event that the frequency of the detection of leaking tubes per plant should increase substantially to more than 1 in twenty days. Specifically, the restriction is that operation is to be terminated if two (2) or more tube leaks per plant occur during any twenty (20) day period.
3. The concentration of iodine in the primary coolant shall be limited to 1.0  $\mu$ Ci/gram during normal operation and to 30  $\mu$ Ci/gram Dose Equivalent I-131 during power transients.
4. At the end of the proposed operating period, or sooner in the event that the limitations in 1 and 2, above, are exceeded, the unit shall be brought to cold shutdown condition for a re-inspection of the conditions of the steam generators and to re-assess the subsequent duration and mode of operation. The inspection program should be as comprehensive as the one discussed previously. Detailed requirements will be determined by the NRC staff on the basis of the continuing operating experience.

### CONCLUSIONS

We have concluded, based on the considerations discussed above that the Turkey Point Unit No. 3 steam generator tubes have sufficient integrity with an adequate margin of safety so that the proposed



additional period of operation would not significantly affect the ability of the tubes to withstand normal operating and postulated accident conditions. In addition, if any reactor coolant system leakage indication in excess of 1.0 GPM occurs during this limited period of operation, the Technical Specifications (Section 3.1, page 3.1-4) require an investigation and evaluation within 4 hours of the indication.

We conclude that Unit No. 3 can continue to operate beyond August 16, 1977 without adverse effect on the public health and safety. Therefore, we find Florida Power and Light Company's proposed amendment for Section 3.E of License DPR-31 acceptable, provided the operational limitations described in the evaluation section are implemented.

We have further concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accident previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

#### Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §1.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Dated: August 16, 1977

REFERENCES

1. H.A. Domian, et al. Effect of Microstructure on Stress Corrosion Cracking of Alloy 600 in High Purity Water. Corrosion, Vol. 33, P. 26, (January 1977).
2. R.L. Cowan and G.M. Gordon. Intergranular Stress Corrosion Cracking and Grain Boundary Composition of Fe-Ni-Cr Alloys, Preprint G-14 of paper presented at Stress Corrosion Cracking and Hydrogen Embrittlement of Iron Base Alloys Conference, Firminy, France (June 1973).
3. J. Blanchet, H. Coriou and et al. Influence of Various Parameters on Intergranular Cracking of Inconel 600 and X-750 in Pure Water at Elevated Temperature, Preprint G-13 of paper presented at Stress Corrosion Cracking and Hydrogen Embrittlement of Iron Base Alloys Conference, Firminy, France, (June 1973).
4. F.W. Pement and N.A. Graham. Stress Corrosion Cracking in High Purity Water, Scientific Paper 74-1B6-TUCOR-P1, Westinghouse Research Laboratories, (June 23, 1974).
5. Ph. Berge, H.D. Bui, J.R. Donati and D. Villard, Corrosion, Vol. 32, p. 357, (September 1976).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-250

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY  
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Facility Operating License No. DPR-31, issued to Florida Power and Light Company, for operation of the Turkey Point Nuclear Generating Unit No. 3 located in Dade County, Florida. The amendment is effective as of the date of issuance.

The amendment to the operating license will allow continued operation of Turkey Point Unit No. 3 until the next refueling (presently scheduled for January 1978) at which time the steam generators shall be inspected.

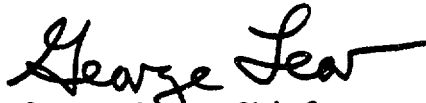
The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §1.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 5, 1977, as modified by letter dated August 10, 1977, (2) Amendment No. 27 to License No. DPR-31, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. and at the Environmental & Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16 day of August 1977.

FOR THE NUCLEAR REGULATORY COMMISSION



George Lear, Chief  
Operating Reactors Branch #3  
Division of Operating Reactors