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Posted Amdt. 106 to DPR-41 See Correction Letter of 7/22/85

Docket Nos. 50-250
and 50-251

April 22, 1985

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Mr. J. W. Williams, Jr., Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Williams:

The Commission has issued the enclosed Amendment No. 112 to Facility Operating License No. DPR-31 and Amendment No. 106 to Facility Operating License No. DPR-41 for the Turkey Point Plant Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letters dated February 8, 1985 and March 6, 1985.

These amendments revise the Technical Specifications to provide consistency in identification of the surveillance specimen capsules in the Technical Specifications and the actual surveillance specimen capsules. The surveillance specimen examination schedule is also modified to provide better information in accordance with the current regulations. The proposed changes combine the existing Reactor Materials Surveillance Program into a single integrated program which conforms to the requirements of 10 CFR 50, Appendices G and H. We have discussed concerns and actions necessary regarding future core designs and in-cavity dosimetry in Section III of our Safety Evaluation provided in support of the amendments.

Section II.C of 10 CFR 50 Appendix H, which was revised on July 26, 1983, permits an integrated surveillance program provided it meets the criteria specified and is approved by the Director, Office of Nuclear Reactor Regulation. We have indicated in our Safety Evaluation that the integrated surveillance program for the Turkey Point Plant permitted by the enclosed amendments meet the criteria specified in 10 CFR 50, Appendix H II.C. The Director, Office of Nuclear Reactor Regulation, has approved the enclosed amendments which authorize an integrated surveillance program at the Turkey Point Plant in accordance with the requirements of 10 CFR 50, Appendix H II.C.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly <u>Federal Register</u> notice.

Sincerely,

/s/BMcDonald

Daniel G. McDonald, Jr., Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No.112 to DPR-31
2. Amendment No.106 to DPR-41
3. Safety Evaluation

cc: w/enclosures
See next page

*SEE PREVIOUS WHITE FOR CONCURRENCE

| | | | | | | |
|---|---|---|---|---|---|---|
| ORB#1:DL* | ORB#1:DL* | BC-ORB#1:DL* | OELD* | AD:OR:DL | D:DL | D:NRR |
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Docket Nos. 50-250
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The Director, Office of Nuclear Reactor Regulation, has approved the enclosed amendments which authorize an integrated surveillance program at the Turkey Point Plant in accordance with the requirements of 10 CFR 50, Appendix H II.C.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Daniel G. McDonald, Jr., Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. to DPR-31
2. Amendment No. to DPR-41
3. Safety Evaluation

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| CParrish | DMcDonald;ps | SVarga | M Young | GLainas | HThompson | HDenton |
| 4/15/85 | 4/15/85 | 4/ /85 | 4/10/85 | 4/ /85 | 4/ /85 | 4/ /85 |

Has been revised

J. W. Williams, Jr.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
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 February 8, 1985 and March 6, 1985, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A and B, as revised through Amendment No. 112 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1985

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

# FLORIDA POWER AND LIGHT COMPANY

## DOCKET NO. 50-251

## TURKEY POINT PLANT UNIT NO. 4

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated February 8, 1985 and March 6, 1985, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A and B, as revised through Amendment No. 106 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 22, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 112 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 106 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-250 AND 50-251

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
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| iv | iv |
| *Table 4.2.1 (cont'd) | Table 4.2.1 (cont'd) |
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| B3.1-3 | B3.1-3 |
| B4.2-12 | B4.2-12 |
| B4.2-13 | B4.2-13 |
| ------------ | B4.20-1 |

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Correction Letter of 7/22/85

TABLE 4.2-1 (Cont'd)

| Item No. | Examination Category | Components and Parts To Be Examined | Method | Extent of Examination (Percent in 10 Year Interval) | Extent of Examination* (Percent in 5 Year Interval) | Remarks |
|---|---|---|---|---|---|---|
| 6.5 | C-2 | Pressure retaining bolt | Visual and Volumetric | 100% | 33% | Exception is taken for valves which are not accessible. |
| 6.6 | K-1 | Integrally-welded supports | | Not Applicable | Not Applicable | |
| 6.7 | K-2 | Supports and Hangers | Visual | 100% | 33% | Exception is taken for supports and hangers which are not accessible. |
| 7.1 | | Reactor Coolant Pump Flywheel | MT and UT | 100%(2) | In-place at bore and keyway (1) | Inservice inspection shall be performed on each reactor coolant pump flywheel during the refueling or maintenance shutdown coinciding with the In-Service Inspection schedule as required by Section XI of the ASME Boiler and Pressure Vessel Code:<br><br>(1) An in-place ultrasonic volumetric examination of the area of higher stress concentration at the bore and keyway at approximately 3 year intervals.<br><br>(2) A surface examination of all exposed surfaces and complete ultrasonic examination at or near the end of each 10 year interval. |

## REACTOR MATERIAL SURVEILLANCE PROGRAM

4.20.1 The following Irradiation Specimen Schedule shall be followed:

### CAPSULE REMOVAL SCHEDULE

| Capsule | Unit | Date |
|---|---|---|
| V | 3 | 12 years |
| V | 4 | 24 years |
| X | 3 | 33 years |
| X | 4 | Standby |

Capsules U, W, Y, and Z for Units 3 and 4 are held in standby.

4.20.2 The above surveillance shall be conducted using the Tensile and Charpy V Notch Test.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$. Adjusted reference temperatures, based upon the fluence and copper content of the material in question, are then determined. The heatup and cooldown limit curves include the shift in $RT_{NDT}$ at the end of the service period shown on the heatup and cooldown curves.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples has a definite relationship to the spectra at the vessel inside radius, the measured transition shift for a sample can be related with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in TS 4.20 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Item 6.5 (Category G-2) - Pressure-Retaining Bolting

The bolting subject to this examination will be the bonnet bolting in valves three (3) inches in size or greater. This bolting will be inspected in acordance with Section XI of the Code as shown in Table 4.2-1.

Item 6.6 (Category K-1) - Integrally-Welded Supports

There are no integrally-welded supports on the valves subject to this examination.

Item 6.7 (Category K-2) - Supports and Hangers

The supports and hangers of the valves subject to this examination will be visually examined in accordance with Section XI of the Code as shown in Table 4.2-1.

MISCELLANEOUS INSPECTIONS

Item 7.1 - Reactor Coolant Pump Flywheels

The flywheels shall be visually examined at the first refueling. At the fourth refueling, the outside surfaces shall be examined by ultrasonic methods. These examinations scheduled are shown in Table 4.2-1.

Item 7.2 - Deleted.



Amendment Nos. 112 and 106

Item 7.3 - Steam Generator Tube Inspection

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. In service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion racking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.2.5.4.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.2.a prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

## B4.20 BASES - REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 Charpy V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each Type I capsule contains four tensile specimens (two specimens from each f the two shell forgings) and six WOL specimens (three specimens from each of the two shelling forgings). Dosimeters of copper, nickel, aluminum-cobolt, and cadmium-shielded aluminum-cobalt wire are secured in holes drilled in spacers at the top, middle, and bottom of each Type I capsule.

Each Type II capsule contains 32 Charpy V-notch specimens: eight specimens machined from one of the shell forgings, eight specimens of weld metal and eight specimens of HAZ metal, the remaining eight specimens are correlation monitors. In addition, each Type II capsule contains four tensile specimens and four WOL specimens: two tensile specimens and two WOL specimens from one of the shell forgings and the weld metal. Each Type II capsule contains a dosimeter block at the center of the capsule. Two cadmium-oxide-shielded capsules, containing the two isotopes uranium-238 and neptunium-237, are contained in the dosimeter block. The double containment afforded by the dosimeter assembly prevents loss and contamination by the neptunium-237 and uranium-238 and their activation products. Each dosimeter block contains approximately 20 milligrams of neptunium-237 and 13 milligrams of uranium-238 contained in a 3/8-inch-OD sealed brass tube. Each tube is placed in a 1/2-inch-diameter hole in the dosimeter block (one neptunium-237 and one uranium-238 tube per block), and the space around the tube is filled with cadmium oxide. After placement of this material, each hole is blocked with two 1/16-inch aluminum spacer discs and an outer 1/8-inch-steel cover disc, which is welded in place. Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt are also secured in holes drilled in spacers located at the top, middle, and bottom of each Type II capsule.

| Capsule Type | Capsule Identification |
|---|---|
| I | S |
| II | V |
| II | T |
| I | U |
| II | X |
| I | W |
| I | Y |
| I | Z |

This program combines the Reactor Materials Surveillance Program into a single integrated program which conforms to the requirements of 10CFR50 Appendices G and H.

 Amendment Nos. 112 and 106

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. DPR-31 AND AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. DPR-41

## FLORIDA POWER AND LIGHT COMPANY

## TURKEY POINT UNIT NOS. 3 AND 4

## DOCKET NOS. 50-250 AND 50-251

## I. Introduction

In a letter from J. W. Williams, Jr. to D. G. Eisenhut, dated February 8, 1985, Florida Power & Light Company requested that the Turkey Point Units No. 3 and 4 Technical Specifications be amended to combine the reactor vessel material surveillance program for these units into a single integrated surveillance program. Additional information concerning the proposed change was provided by the licensee in a letter from J. W. Williams, Jr. to S. A. Varga dated March 6, 1985.

A revised Appendix H, 10 CFR 50 was published in the Federal Register on May 27, 1983 and became effective on July 26, 1983. Section II.C of the revised Appendix H permits an integrated surveillance program provided it is approved by the Director, Office of Nuclear Reactor Regulation. This section of Appendix H identifies the criteria to be used in evaluating the integrated surveillance program. The criteria are:

1. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

2. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

3. There must be an adequate dosimetry program for each reactor.

4. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

5. No reduction in the requirements for number of materials to be irradiated, specimen type, or number of specimens per reactor is **permitted, but the amount of testing may be reduced if the initial results agree with predictions.**

6. There must be adequate arrangement for data sharing between plants.

## II. Evaluation

Each unit at Turkey Point began commercial operation with 8 surveillance capsules in each reactor vessel. Ten capsules contained forging material and six capsules contained weld metal, forging, and heat affected zone (HAZ) materials. To date, two capsules containing forging material and two capsules containing weld metal, forging, and HAZ materials were irradiated, removed from the vessel, and tested. The test results from the surveillance material indicate that the weld metal will sustain the most irradiation damage. **Since, based on the initial test, the weld metal is more susceptible to irradiation damage than the forging material, the licensee has proposed to retain the capsules with forging material as standby specimens in the reactor vessel and test only those capsules with weld metal, forging, and HAZ materials. Since fewer capsules will be withdrawn than originally anticipated, the radiation exposure (ALARA) to plant personnel should be reduced.**

The licensee's FSAR Volume 2 indicates that the materials and designs for the core, thermal shield, core barrel and vessel are the same for each unit at Turkey Point. Since the neutron energy spectrum is a function of geometry, materials, and core loading, the relative neutron spectrum for both reactors should be equivalent for equivalent core loadings. The licensee indicates that fuel management and cycle lengths for both units have been similar. Thus neutron spectra profiles at the peak fluence locations should be equivalent.

The neutron fluence, which is used to predict radiation damage, is calculated using PDQT power distribution data, and computer codes SORREL and DOT 4.3. As built dimensions and individual material properties are incorporated into the DOT 4.3 models. Hence, using these codes, the licensee will be able to predict radiation damage as a function of power output for each unit.

Each vessel has both in-capsule and in-cavity dosimetry, which will be used to verify the neutron spectra and the codes that were used to predict neutron fluence as a function of power output. Since each plant has its own capsules and both plants are capable of independently predicting and monitoring radiation damage as a function of power output, the program will not be significantly jeopardized by operation at reduced power levels or by an extended outage of either plant.

Based on the intial test, the limiting material for each unit is weld material, which is identified as SA 1101. This material is in each capsule that will be irradiated and tested. Capsules that have been deleted from surveillance testing do not contain the limiting material and will be retained as standby specimens in the reactor vessel. Since the amount of limiting material in the surveillance program has not chnaged, the number of useful surveillance specimens available for testing has not changed.

Both units have common management and the surveillance program will be managed by their Nuclear Energy Department. Therefore, there should be adequate data sharing.

III. Findings

1. We have concluded based on the details in Section II of this Safety Evaluation, that the integrated surveillance program meets the evaluation criteria specified in 10 CFR 50, Appendix H II.C. If future core designs are significantly different than those documented by the licensee, the licensee must explain the effect that the changes have on neutron irradiation damage and the surveillance capsule withdrawal schedule.

2. In-cavity dosimetry testing should continue in order to reduce projected uncertainties in neutron fluence. If these test results provide an effective method of monitoring vessel neutron fluence, the in-cavity dosimetry should be incorporated into the integrated surveillance program.

IV. Environmental Consideration

These amendments involve changes in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20 and in surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

V. Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 22, 1985

Principal Contributors:

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