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and 50-251

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 10 to Facility Operating License No. DPR-31 and Amendment No. 9 to Facility Operating License No. DPR-41 for Turkey Point Nuclear Generating Units 3 and 4. These amendments include Change No. 22 to the Technical Specifications and are in response to your request dated September 19, 1974, as supplemented January 9, 1975.

These amendments incorporate in the Technical Specifications a revised Administrative Controls Section except for that portion relating to the reporting requirements which is still under review.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 10
2. Amendment No. 9
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Florida Power & Light Company

cc: w/enclosure

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
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 September 19, 1974, as supplemented January 9, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-41 is hereby amended to read as follows:

"(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 24."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Change No. 22 to the
Technical Specifications

Date of Issuance: SEP 3 1975

ATTACHMENT TO LICENSE AMENDMENTS NOS. 10 AND 9

CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NOS. DPR-31 AND DPR-41

DOCKETS NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii through v	ii through v
6.1-1 through 6.6-11	6-1 through 6-37
Tables 6.1-1	
6.4-1	
6.6-1	
Figures 6.1-1	
6.1-2	
6.1-3	

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22

TABLE OF CONTENTS (Continued)

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LIST OF FIGURES

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Superintendent - Nuclear shall be responsible for overall licensed facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

6.2.1 OFFSITE

The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

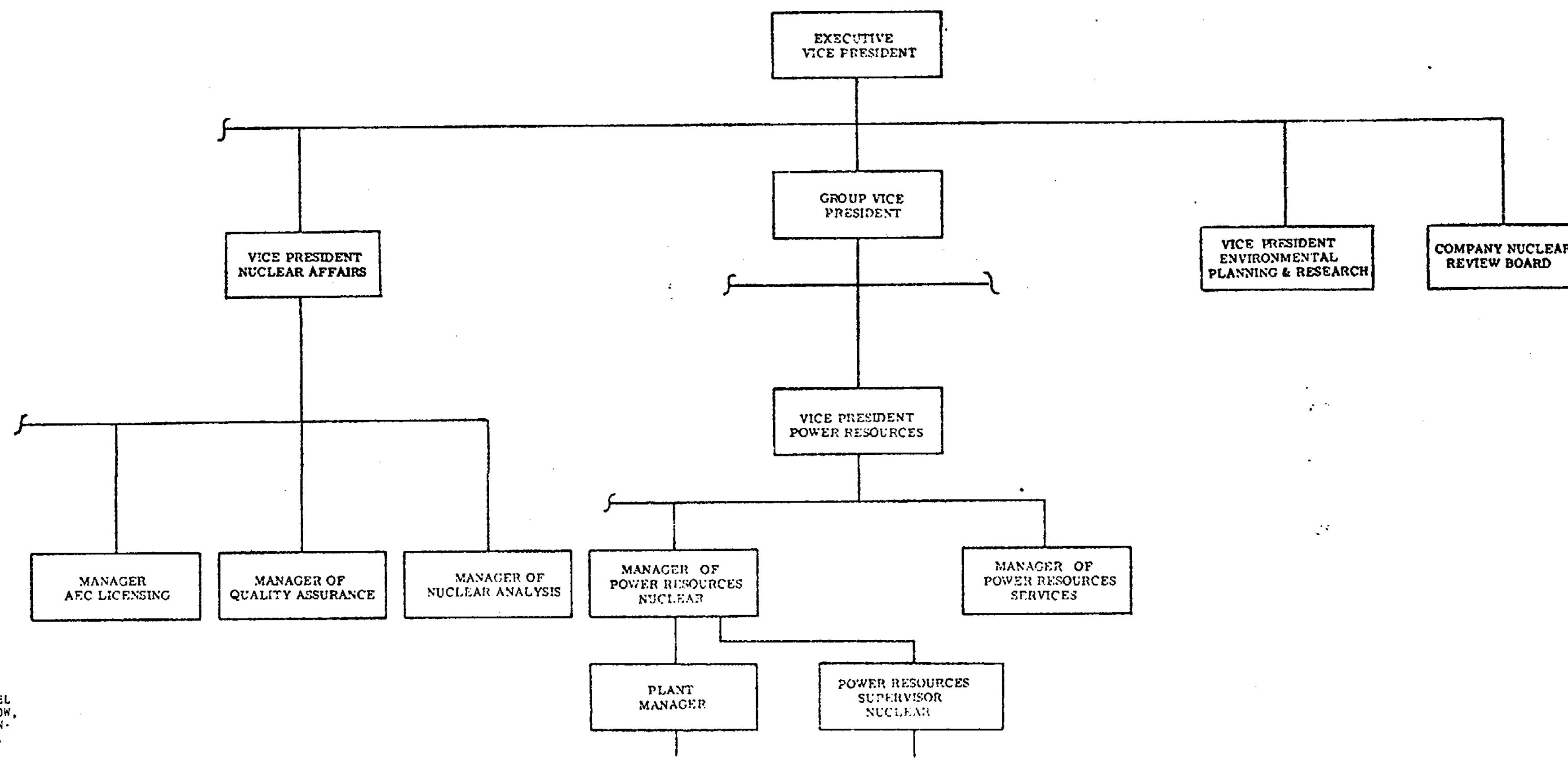
6.2.2 FACILITY STAFF

The Facility organization shall be as shown on Figure 6.2-2 and: 22

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.



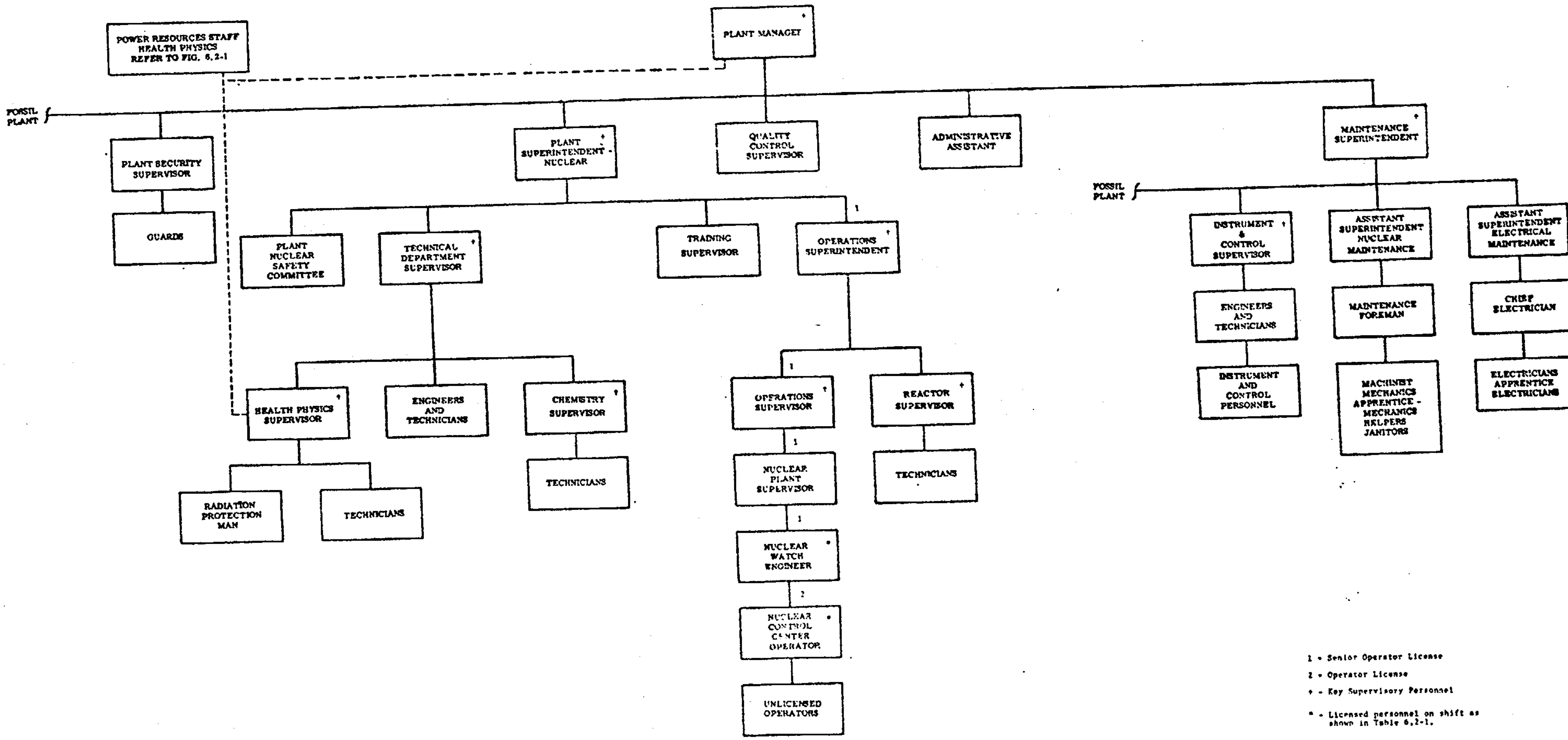
*INCLUDES HEALTH PHYSICS, FUEL MANAGEMENT, OPERATIONS FOLLOW, DESIGN REVIEW, REACTOR ENGINEERING, RADIOCHEMISTRY, ETC.

SEE FIGURE 6.2-2

CONTINUED ON FIGURE 6.2-2

OFFSITE ORGANIZATION CHART

FIGURE 6.2-1



1 = Senior Operator License

2 = Operator License

+ - Key Supervisory Personnel

* - Licensed personnel on shift as shown in Table 6.2-1.

PLANT ORGANIZATION CHART
FIGURE 6.2-2

TABLE 6.2-1

OPERATING PERSONNEL

Personnel on Shift	CONDITION OF ONE UNIT (No Fuel in Second Unit)		
	Initial Fuel Loading or During Refueling	Cold Shutdown or Refueling Shutdown	Above Cold Shutdown Through Power Operation
Senior Reactor Operator	1	1	1
Reactor Operator	2	1	2
Unlicensed Operators	(as req'd)	1	3

Personnel on Shift	CONDITION OF SECOND UNIT (One Unit at Hot Shutdown or at Power)		
	Initial Fuel Loading or During Refueling	Cold Shutdown or Refueling Shutdown	Above Cold Shutdown Through Power Operation
Senior Reactor Operator	2	2	2
Reactor Operator	3	2	3
Unlicensed Operators	3+ (as req'd)	3	3

Personnel on Shift	CONDITION OF SECOND UNIT (One Unit at Cold Shutdown or Refueling Shutdown)		
	Initial Fuel Loading or During Refueling	Cold Shutdown or Refueling Shutdown	Above Cold Shutdown Through Power Operation
Senior Reactor Operator	2	1	2
Reactor Operator	3	2	2
Unlicensed Operators	3+ (as req'd)	3	3

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

6.4 TRAINING 22

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5, ANSI N18.1-1971 and Appendix A to 10 CFR Part 55.

6.5 REVIEW AND AUDIT

6.5.1 Plant Nuclear Safety Committee PNSC.

6.5.1.1 FUNCTION

The PNSC shall function to advise the Plant Superintendent - Nuclear on all matters related to nuclear safety.

6.5.1.2 COMPOSITION

The Plant Nuclear Safety Committee shall be composed of the:

1. Chairman: Plant Superintendent - Nuclear

2. Vice Chairman: Operations Superintendent

3. Technical Department Supervisor

4. Assistant Superintendent - Nuclear Maintenance

5. Instrument and Control Supervisor 22

6. Health Physics Supervisor

7. Reactor Supervisor

6.5.1.3 ALTERNATES

Alternate members shall be appointed in writing by the PNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PNSC activities at any one time.

6.5.1.4 MEETING FREQUENCY

The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman.

6.5.1.5 QUORUM

A quorum of the PNSC shall consist of the Chairman or Vice Chairman and four (4) members including alternates.

6.5.1.6 RESPONSIBILITIES

The Plant Nuclear Safety Committee shall be responsible for:

a. Review of 1) all procedures and changes thereto required by Section 6.8 and 2) any other proposed procedures or changes thereto as determined by the Plant Superintendent - Nuclear to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications in Appendix A of the license.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety. 22

e. Investigation of all violations of the Technical Specifications and preparation and forwarding a report covering evaluation and recommendations to prevent recurrence to the Manager of Power Resources - Nuclear, to the Vice President of Power Resources and to the chairman of the Company Nuclear Review Board.

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Company Nuclear Review Board.

h. Review of the Plant Security Plan and implementing procedures and submitting recommended changes to the Chairman of the Company Nuclear Review Board.

i. Review of the Emergency Plan and implementing procedures and submitting recommended changes to the Chairman of the Company Nuclear Review Board. 22

6.5.1.7 <u>AUTHORITY</u>

The Plant Nuclear Safety Committee shall:

a. Recommend to the Plant Superintendent - Nuclear written approval or disapproval (in minutes of PNSC meeting) of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing (in minutes of PNSC meetings) with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide immediate written notification to the Director of Power Resources and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Superintendent - Nuclear; however, the Nuclear Plant Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

6.5.1.8 RECORDS

The Plant Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to the Director of Power Resources and Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

6.5.2.1 FUNCTION

22

The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations.
b. Nuclear engineering.
c. Chemistry and radiochemistry.
d. Metallurgy.
e. Instrumentation and control.
f. Radiological safety.
g. Mechanical and electrical engineering.
h. Quality assurance practices.

6.5.2.2 COMPOSITION

The CNRB shall be composed of the:

1. Chairman: Vice President, Nuclear Affairs.
2. Member: Chief Engineer - Power Plants.
3. Member: Vice President of Power Resources
4. Member: Power Plant Engineering supervisor.
5. Member: Manager of Power Resources - Nuclear.
6. Member: Manager of Quality Assurance.
7. Member: Power Plant Engineering supervisor.

6.5.2.3 ALTERNATES

Alternate members shall be appointed in writing by the (CNRB) Chairman to serve on a temporary basis; however, no more than two alternates shall participate in (CNRB) activities at any one time. 22

6.5.2.4 CONSULTANTS

Consultants shall be utilized as determined by the CNRB to provide expert advice to the CNRB.

6.5.2.5 MEETING FREQUENCY

The CNRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

6.5.2.6 QUORUM

A quorum of the CNRB shall consist of the Chairman or designated acting Chairman and four (4) members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

6.5.2.7 REVIEW

The CNRB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and, 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 22
10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or Licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. ABNORMAL OCCURRENCES, as defined in Section 1.14 of these Technical Specifications.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Plant Nuclear Safety Committee.

6.5.2.8 AUDITS

Audits of facility activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year. 22

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The Emergency Plans and implementing procedures at least once per two years.

f. The Security Plan and implementing procedures at least once per two years.

g. Any other area of facility operation considered appropriate by the CNRB or the Executive Vice President.

6.5.2.9 AUTHORITY

The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

6.5.2.10 RECORDS

Records of CNRB activities shall be prepared, approved and distributed as indicated below: 22

a. Minutes of each CNRB meeting shall be prepared, approved and forwarded to the Executive Vice President within fourteen days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the Executive Vice President within fourteen days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within thirty (30) days after completion of the audit.

6.6 ABNORMAL OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of an ABNORMAL OCCURRENCE:

a. The ABNORMAL OCCURRENCE shall be reported to the Commission pursuant to the requirements of Section 6.9.

b. An Abnormal Occurrence Report shall be prepared. The report shall be reviewed by the Plant Nuclear Safety Committee.

c. The Abnormal Occurrence Report shall be submitted to the CNRB, the Vice President of Power Resources, and the Commission within the time allotted in Section 6.9.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied within immediately. 22

b. The Safety Limit violation shall be reported immediately to the Commission, the Vice President of Power Resources and to the CNRB.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PNSC. This report shall describe 1) applicable circumstances preceding the violation, 2) effects of the violation upon facility components, systems or structures, and 3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the CNRB, the Vice President of Power Resources and the Commission within ten (10) days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USNRC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PNSC and approved by the Nuclear Plant Superintendent prior to implementation and periodically as provided by procedure.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided: 22

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator License on the unit affected.

c. The change is documented, reviewed by the PNSC and approved by the Plant Superintendent - Nuclear within seven days of implementation.

6.9 PLANT REPORTING REQUIREMENTS 22

In addition to reports required by Title 10 Code of Federal Regulations, Florida Power & Light Company shall provide the following information: 22

6.9.1 ROUTINE REPORTS 22

a. <u>Operations Reports</u>

Operations Reports shall be submitted in writing to the Director of the appropriate Regional Office of Inspection and Enforcement. 22

(1) Startup Report

A summary report of unit startup and power escalation testing and an evaluation of the results from these test programs shall be submitted following receipt of operating licenses, following amendments to the licenses involving the planned increase in power level, following the installation of a new core, or following modifications to an extent that the nuclear, thermal, or hydraulic performance of the unit may be significantly altered. The test results shall be compared with design predictions and specifications. Startup reports shall be submitted within 60 days following commencement of rated power operation.

(2) First Year Operation Report

A report shall be submitted within 14 months following commencement of rated power operation. This report may be incorporated into the semiannual operating report and shall cover the following:

an evaluation of unit performance to date in comparison with design predictions and specifications;

(b) a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analyses;

(c) an assessment of the performance of structures, systems and components important to safety;

(d) a progress and status report on any items identified as requiring additional information during the operating license review or during the startup of the nuclear units, including items discussed in the AEC's safety evaluation, items on which additional information was required as conditions of the license and items identified in the licensee's startup report.

(3) Semiannual Operating Reports

Routine operating reports shall be submitted within 60 days after January 1 and July 1 of each year. The first such period should begin with the date of initial criticality. These reports should include the following:

(a) Operations Summary

A summary of operating experience occurring during the reporting period that relates to the safe operation of the nuclear units, including a summary of:

22

(i) changes in nuclear unit design,

(ii) performance characteristics (e.g., equipment and fuel performance),

(iii) changes in procedures which were necessitated by (i) and (ii) or which otherwise were required to improve the safety of operations.

(iv) results of surveillance tests and inspections required by these technical specifications,

(v) the results of any periodic containment leak rate tests performed during the reporting period,

(vi) a brief summary of those changes, tests and experiments requiring authorization from the Commission pursuant to 10 CFR Part 50.59(a), and

(vii) any changes in the nuclear unit operating organization which involve positions which are designated as key supervisory personnel on
Figure 6.1-2. 22

(viii) results of required leak tests performed on sources if the tests reveal the presence of 0.005 μCi or more of removable contamination.

(b) Power Generation

A summary of power generated during the reporting period including:

(i) gross thermal power generated (in MWH)

(ii) gross electrical power generated (in MWH)

(iii) net electrical power generated (in MWH)

(iv) number of hours the reactor was critical

(v) number of hours the generator was on-line

(vi) histogram of thermal power vs. time

(c) Shutdowns

Descriptive material covering all outages occurring during the reporting period. For each outage, information shall be provided on:

(i) the cause of the outage,

(ii) the method of shutting down the reactor; e.g., trip automatic rundown, or manually controlled deliberate shutdown,

(iii) duration of the outage,

(iv) unit status during the outage; e.g., cold shutdown or hot shutdown,

(v) corrective action taken to prevent repetition, if appropriate,

(d) Maintenance

A discussion of safety-related maintenance (excluding preventative maintenance) performed during the reporting period on systems and components that are designated to prevent or mitigate the consequences of postulated accidents or to prevent the release of significant amounts of radioactive material. Included in this category are systems and components which are part of the reactor coolant pressure boundary defined in 10CFR§50.2(v), part of the engineered safety features, or associated service and control systems that are required for the normal operation of engineered safety features, part of any reactor protection or shutdown system, or part of any radioactive waste treatment handling and disposal system or other system which may contain significant amounts of radioactive material. For any malfunctions for which corrective maintenance was required, information shall be provided on:

(i) the system or component involved,

(ii) the cause of the malfunction,

(iii) the results and effect on safe operation,

(iv) corrective action taken to prevent repetition,

(v) precautions taken to provide for reactor safety during repair.

(c) Changes, Tests and Experiments

A summary of all changes in the plant design and procedures that relate to the safe operation of the plant shall be included in the Operations Summary section of these semiannual reports. Changes, tests, and experiments performed during the reporting period that require authorization from the Commission pursuant to 10CFR50.59(a) are covered in paragraph 6.9.1.a(3)(a)(vi) of these technical specifications; those changes, tests, and experiments that do not require Commission authorization pursuant to § 50.59(a) shall also be addressed. The report shall include a brief description and the summary of the safety evaluation for those changes, tests, and experiments, carried out without prior Commission approval, pursuant to the requirements of § 50.59(b) of the Commission's regulations, that "The licensee shall furnish to the Commission, annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each".

) Radioactive Effluent Relea

A statement of the quantities of radioactive effluents released from the plant, with data summarized on a monthly basis following the format of Appendix A of USAEC Safety Guide 21 of January 1972:

(i) Gaseous Releases

(a) Total radioactivity (in curies) releases of noble and activation gases.

(b) Maximum noble gas release rate during any one-hour period.

(c) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(d) Percent of technical specification limit.

(ii) Iodine Releases

(a) Total (I-131, I-133, I-135) radioactivity (in curies) released.

(b) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(c) Percent of technical specification limit.

(iii) Particulate Releases

(a) Gross radioactivity (β,γ) released (in curies) excluding background radioactivity.

(b) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(c) Total radioactivity released (in curies) of nuclides with half-lives greater than eight days.

(d) Percent of technical specification limit.

(iv) Liquid Releases

(a) Gross radioactivity (β,γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

(b) Total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(c) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(d) Total volume (in liters) of liquid waste released.

(e) Total volume (in liters) of dilution water used prior to release from the restricted area.

(f) The maximum concentration of gross radioactivity (β,γ) released to the unrestricted area (averaged over the period of release).

(g) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(h) Percent of technical specification limit for total activity released.

(v) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet)

(b) The total estimated radioactivity (in curies) involved.

(c) The dates of shipment and disposition (if shipped off-site).

(g) Environmental Radiological Monitoring

(i) For each medium sampled e.g., air, baybottom, surface water, soil, fish including:

(a) Number of sampling locations

(b) Total number of samples

(c) Number of locations at which levels are found to be significantly above local backgrounds

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(ii) If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 3% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(iii) If statistically significant variations of off-site environmental concentrations with time are observed, correlation of these results with effluent releases shall be provided.

(h) Occupational Personnel Radiation Exposure

A tabulation in 0.15 rem increments of personnel exposures between 0 - 1.25 rem by work groups.

6.9.2 NON-ROUTINE REPORTS 22

a. Reporting of Abnormal Events

Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone
or telegraph to the Director of the Region II Office of 22
Inspection and Enforcement followed by a written report
within 10 days to the Director of the Region II Office 22
of Inspection and Enforcement in the event of the
abnormal occurrences as defined in Section 1.13.

The written report on these abnormal occurrences, and to the extent possible, the preliminary telephone or telegraph notification, shall: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, and (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems.

In addition, the written report shall relate any failures or degraded performance of systems and components for the incident to similar equipment failures that may have previously occurred at the plant. The evaluation of the safety implications of the incident should consider the cumulative experience obtained from the record of previous failures and malfunctions of the affected systems and components or of similar equipment.

 22

A written report shall be forwarded within 30 days to the Director of the Region II Office of Inspection and Enforcement, in the event of: 22

(1) Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the technical specifications.

(2) Any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.

(3) Any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

6.9.3 SPECIAL REPORTS 22

Special reports shall be submitted in writing within 90 days to the Director, Division of Reactor Licensing, U. S. NRC, Washington, D. C. 20555: 22

Special reports shall be submitted covering inspections, tests and maintenance that are appropriate to assure safe operation of the plant. The frequency and content of these special reports are determined on an individual case basis and designated in these technical specifications. Examples of subjects for such reports include:

(a) In-service inspection, reference 4.2

(b) Tendon surveillance, reference 4.4

(c) Containment structural tests, reference 4.4

(d) Special maintenance reports

(e) Authorization of changes, tests, and experiments in accordance with 10CFR50.59

(f) Containment leak rate tests, reference 4.4

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five (5) years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. ABNORMAL OCCURRENCE Reports. 22

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Record of annual physical inventory verifying accountability of sources on record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys. 22

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of tansients or cycles.

g. Records of training and qualification for members of the plant staff for the duration of their employment.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities as required by Corporate Quality Assurance Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PNSC and the CNRB.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 RESPIRATORY PROTECTION PROGRAM

ALLOWANCE

6.12.1 Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix "B", Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over seven consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix "B", Table I, Column 1, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix "B", Table I, Column 1, of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the
limitation on individual dose in §20.101. These materials shall 22
be subject to applicable process and other engineering controls.

6.12.2 PROTECTION PROGRAM

In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in 6.12.1 above, are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix "B", Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any 22
other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

   1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

   2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee shall use equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in 22
Table 6.12-1. Equipment not approved under U. S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

6.12.3 REVOCATION

The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mRem/hr but less than 1000 mRem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter 22
such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mRem/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under administrative control.

TABLE 6.12-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION		MODES	PROTECTION FACTORS[2] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I.	AIR-PURIFYING RESPIRATORS			
	Facepiece, half-mask[4, 7]	NP	5	21B 30 CFR § 14.4(b)(4)
	Facepiece, full[7]	NP	100	21B 30 CFR § 14.4(b)(5); 14F 30 CFR 13
II.	ATMOSPHERE-SUPPLYING RESPIRATOR			
	1. Airline Respirator			
	Facepiece, half-mask	CF	100	19B 30 CFR § 12.2(c)(2) Type C(i)
	Facepiece, full	CF	1,000	19B 30 CFR § 12.2(c)(2) Type C(i)
	Facepiece, full[7]	D	100	19B 30 CFR § 12.2(c)(2) Type C(ii)
	Facepiece, full	PD	1,000	19B 30 CFR § 12.2(c)(2) Type C(iii)
	Hood	CF	[5]	[6]
	Suit	CF	[5]	[6]
	2. Self-contained breathing apparatus (SCBA)			
	Facepiece, full[7]	D	100	13E 30 CFR § 11.4(b)(2)(i)
	Facepiece, full	PD	1,000	13E 30 CFR § 11.4(b)(2)(ii)
	Facepiece, full	R	100	13E 30 CFR § 11.4(b)(1)
III.	COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR § 12.2(e) or applicable schedules as listed above

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[1] See the following symbols:

- CF: continuous flow
- D: demand
- NP: negative pressure (i.e., negative phase during inhalation)
- PD: pressure demand (i.e., always positive pressure)
- R: recirculating (closed circuit)

[2] (a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

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(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency (above 99.9% removal efficiency by U. S. Bureau of Mines type dioctyl phthalate (DOP) test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

TABLE 6.12-1 (CONTINUED)

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

[3] Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately two is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5, below, concerning supplied-air suits and hoods.

[4] Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix "B", Table I, Column 1 of 10 CFR Part 20.

[5] Appropriate protection factors must be determined taking account of
the design of the suit or hood and its permeability to the contaminant 22
under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

[6] No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

[7] Only for shaven faces.

<u>NOTE 1:</u> Protection factors for respirators, as may be approved by the

TABLE 6.12-1 (CONTINUED)

U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines 22
in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix "B", Table I of this part are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO LICENSE NO. DPR-31, AND

AMENDMENT NO. 9 TO LICENSE NO. DPR-41

(CHANGE NO. 22 TO TECHNICAL SPECIFICATIONS)

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNITS 3 AND 4

DOCKET NOS. 50-250 AND 50-251

Introduction

By letters dated September 19, 1974, and January 9, 1975, Florida Power and Light Company (FPL) requested changes to the Technical Specifications appended to Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Nuclear Generating Units 3 and 4. The proposed amendments include changes to the Administrative Controls section of the Technical Specifications. Although changes were also proposed by FPL to the reporting requirements as specified in the Technical Specifications, these license amendments do not include changes to presently specified reporting requirements.

By letter dated October 22, 1974, we requested all nuclear power facility licensees, with the exception of FPL, to submit an application for a license amendment to revise the Administrative Controls section of their Technical Specifications. Our request was made to all licensees in order that the Administrative Controls section of the Technical Specifications for all operating facilities would be consistent in format and content with the Standard Technical Specifications. Our October 22, 1974, letter was not sent to FPL as FPL had previously submitted proposed license amendments which fulfilled the requirements of our October 22, 1974 letter.

Discussion

The proposed changes to the administrative controls section are intended to provide uniform license requirements. The major areas covered by the proposed changes are licensee staffing qualifications, management procedures involved with operating the reactor, and the respiratory protection program.

Each member of the facility staff is required to meet the qualification requirements stated in ANSI Standard N18.1-1971. The retraining and replacement training program for the facility staff must be maintained to meet as a minimum the requirements and recommendations of Section 5.5 of ANSI Standard N18.1-1971 and Appendix A of 10 CFR Part 55. The personnel composition of management review groups and procedures for performing reviews are specified in ANSI Standard N18.7-1972. The members of the Turkey Point facility staff, the Plant Nuclear Safety Committee and the Company Nuclear Review Board presently meet the requirements of ANSI, N18.1-1971 and ANSI N18.7-1972. The radiation protection program delineates use of respiratory equipment in the event personnel are to be exposed to concentrations in excess of Part 20 concentrations.

Evaluation

We have compared the proposed Technical Specifications to: (1) the requirements specified in the appropriate ANSI standards and (2) the Standard Technical Specifications being applied to recently licensed facilities (e.g. D. C. Cook, Docket 50-315). Based on this comparison, we have determined that: (1) the minimum acceptable qualifications for key personnel are identified, and (2) the requirements for training and replacement training of the facility staff are specified. Therefore, facility operation using the proposed Technical Specifications will assure that the facility staff will continue to consist of personnel who are competent in facility operation.

Other administrative requirements such as abnormal occurrence action, safety limit violation action, administrative procedures and record retention, imposed by the proposed Technical Specifications assure uniformity throughout the Technical Specifications and conformance to NRC requirements in the areas of review, staffing and procedures.

The respiratory protection program contained in the present Technical Specifications has been modified to correspond to the NRC approved standard respiratory protection program. Incorporating the standard respiratory protection program into the Technical Specifications assures that a method of using respiratory equipment, which is acceptable to the NRC staff, is available whenever needed.

Technical Specification changes similar to those proposed by this action are being approved for all power reactor licensees. Therefore, all licensees will have the same administrative requirements presented in a uniform manner.

As a result of the review and evaluation, we have concluded that the proposed administrative controls will improve and/or assure a high level of proficiency of plant operation and of plant performance evaluation. Moreover, we conclude that these changes do not involve any relaxation of controls presently in the license.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: SEP 3 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 10 and No. 9 to Facility Operating Licenses Nos. DPR-31 and DPR-41, respectively, issued to Florida Power and Light Company which revised Technical Specifications for operation of the Turkey Point Nuclear Generating Units 3 and 4, located in Dade County, Florida. The amendments are effective as of the date of issuance.

These amendments incorporate into the Turkey Point Nuclear Generating Units 3 and 4, Technical Specifications re-formatted Administrative Controls consistent with presently issued licenses. They relate to licensee staffing qualifications and management procedures involved with operating the facilities. The revisions to the reporting requirements proposed by the licensee as part of the Administrative Controls have not been incorporated into the Technical Specifications since they are still under review.

The application for the amendments 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 amendments. Prior public notice of the amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendments dated September 19, 1974, as supplemented January 9, 1975, (2) Amendment No. 10 to License No. DPR-31 and Amendment No. [illegible] to License No. DPR-41 with Change No. 22, 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 and Urban Affairs Library, Florida International University, Miami, Florida.

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 Reactor Licensing.

Dated at Bethesda, Maryland, this 3rd day of September, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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