① Correction ltr of 3/20/87
② Correction ltr of 4/3/87
③ Correction ltr of 6/18/87

March 6, 1987

Docket Nos. 50-250
and 50-251

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Mr. C. O. Woody, Group Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Woody:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-31 and Amendment No. 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 letter dated August 20, 1985, as supplemented on May 13, 1986.

These amendments revise Turkey Point Plant Units 3 and 4 Technical Specifications in two areas. The first change revises the immediate notification requirements and the Licensee Event Reporting System per guidance provided in the Nuclear Regulatory Commission (NRC) staff's Generic Letter 83-43 to assure compliance with the revised Section 50.72 and the new Section 50.73 of Title 10 of the Code of Federal Regulations. The second change revises the Off-Site Organization for Facility Management and Technical Support and the Plant Organization Chart to reflect the current structure and position titles. The bases section is updated for the steam generator inspection results to support the existing Technical Specifications and reflect the changes in reporting requirements.

In addition to the two areas discussed, areas dealing with communications between the licensee and NRC which contain conflicting submittal directions have been corrected to be in accordance with the Final Rule on "Communications Procedures Amendments, 10 CFR Part 50" that was published in the Federal Register on November 6, 1986, and which became effective 60 days after the publication.

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

Sincerely,

/s/

Daniel G. McDonald, Jr., Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.123 to DPR-31
2. Amendment No.116 to DPR-41
3. Safety Evaluation

cc: w/enclosures
See next page

LA:PAD#2	PM:PAD#2	OGC	PD:PAD#2
DMiller	for DMcDonald:bg		LRubenstein
2/17/87	2/18/87	2/20/87	3/5/87

check w/ STAR before issuance

see note 3/2/87

Okay See note 3/3/87

Mr. C. O. Woody
Florida Power and Light Company

Turkey Point Plant

cc:
Harold F. Reis, Esquire
Newman and Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

Mr. Jack Shreve
Office of the Public Counsel
Room 4, Holland Building
Tallahassee, Florida 32304

Norman A. Coll, Esquire
Steel, Hector and Davis
4000 Southeast Financial
Center
Miami, Florida 33131-2398

Mr. C. M. Wethy, Vice President
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

Mr. M. R. Stierheim
County Manager of Metropolitan
Dade County
Miami, Florida 33130

Resident Inspector
U.S. Nuclear Regulatory Commission
Turkey Point Nuclear Generating Station
Post Office Box 57-1185
Miami, Florida 33257-1185

Mr. Allan Schubert, Manager
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32301

Intergovernmental Coordination
and Review
Office of Planning & Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

Administrator
Department of Environmental
Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

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

Martin H. Hodder, Esquire
1131 NE, 86th Street
Miami, Florida 33138

Joette Lorion
7269 SW, 54 Avenue
Miami, Florida 33143

Mr. Chris J. Baker, Plant Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

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. 123
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 20, 1985, as supplemented on May 13, 1986, 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 Appendices A and B, as revised through Amendment No. 123 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

L. S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 6, 1987

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. 116
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 August 20, 1985, as supplemented on May 13, 1986, 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 Appendices A and B, as revised through Amendment No. 116 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

L. S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 6, 1987

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 123 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 116 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
ii	ii
iii	iii
vi	vi
Pg 1-3	Pg 1-3
Pg 3.1-5	Pg 3.1-5
Pg 3.2-7	Pg 3.2-7
Pg 3.14-1 thru 3.14-3	Pg 3.14-1 thru 3.14-3
Table 4.2-3	Table 4.2-3
Pg 4.2-6	Pg 4.2-6
Pg 4.11-1	Pg 4.11-1
*Pg 6.1	Pg 6.1
Fig. 6.2-1	Fig. 6.2-1
Fig. 6.2-2	Fig. 6.2-2
Table 6.2-1	Table 6.2-1
Pg 6-5 thru 6-27	Pg 6-5 thru 6-31
B3.14-1	B3.14-1
B4.2-1	B4.2-1
B4.2-13	B4.2-13

*The entire Section 6, "Administrative Controls", is being replaced per these Amendments due to changes and reformatting. Pages 6-1,-5,-6,-12,-13,-20,-22,-25, 26,-27 and Table 6.2-1 have no changes but have been retyped to be consistent with the format of Section 6.

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TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

1.7 INSTRUMENTATION SURVEILLANCE

1) Channel Check

Channel check is a qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable or radioactive source check of the Area and Process Radiation Monitoring Systems for channels.

2) Channel Functional Test

A channel functional test consists of injecting a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

3) Channel Calibration

Channel calibration consists of the adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.8 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3 within 30 days indicating the number of hours above this limit.

b. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1-1, be in a SHUTDOWN condition with average reactor coolant temperature less than 500 F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\overline{E}$ microcuries per gram, be in a SHUTDOWN condition with average reactor coolant temperature less than 500 F within 6 hours.

2. For all modes of operation

a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ microcuries per gram, perform the sampling and analysis requirements of item 1.h.1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits.

1) The hot channel factors shall be determined within 2 hours and the power level and trips adjusted to meet the requirements of Item 6a.

2) If the hot channel factors are not determined within two hours, the power shall be reduced from rated power 2% for each percent of quadrant tilt.

3) If the quadrant to average power tilt exceeds ±10%, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

i. If after a further period of 24 hours, the power tilt in h. above is not corrected to less than +2%, and

1) If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and submitted as a Report to the Commission pursuant to Specification 6.9.3.1.

2) If the hot channel factors are not determined, the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

7. <u>IN-CORE INSTRUMENTATION</u>

a. A minimum of 16 thimbles, at least 2 per quadrant, and the necessary associated detectors shall be operable during the check and calibration of nuclear instrumentation ion chambers.

b. Power shall be limited to 90% of rated power for 3 loop or 50% of rated power for 2 loop operation if the requirements in Section 7.a are not met.

3.14 FIRE PROTECTION SYSTEMS

Applicability: Applies to the availability of fire protection systems in nuclear safety related areas.

Objective: To define those conditions of fire protection availability.

Specification:

1. FIRE DETECTION INSTRUMENTATION

   a. As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.14-1 shall be OPERABLE, whenever equipment in that fire detection zone is required to be OPERABLE.

   b. If Specification 3.14.1.a cannot be met because one or more of the fire detection instrument(s) shown in Table 3.14-1 is inoperable:

      (1) Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour.

      (2) Restore the inoperable zone(s) to OPERABLE status within 14 days, or prepare and submit a Report to the Commission pursuant to Specification 6.9.3.c.

   c. A fire watch patrol shall be established to inspect the 18 foot level of the turbine area once each hour.

2. FIRE SUPPRESSION WATER SYSTEM

   a. The fire suppression water system shall be OPERABLE, at all times with:

      (1) Two fire suppression pumps, each with a capacity of 2,000 gpm, with their discharge aligned to the fire suppression header,

      (2) Separate water supplies, with a minimum contained volume of 30,000 gallons in the elevated storage tank (EST) and 150,000 gallons in the raw water storage tank, and

(3) An OPERABLE flow path capable of taking suction from the raw water storage tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrants and the last root valve ahead of each sprinkler or hose standpipe.

b. (1) If Specification 3.14.2.a cannot be met because one pump and/or one water supply is inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, prepare and submit a Report to the Commission pursuant to Specification 6.9.3.c.

(2) With one water supply below the minimum specified limit for one day, connect the spool piece to make the screen wash pump available for fire water supply.

c. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

3. SPRAY AND/OR SPRINKLER SYSTEMS

a. The following spray and/or sprinkler systems shall be OPERABLE, whenever equipment in the spray/sprinkler protected area is required to be OPERABLE.

1. Unit 3 - 4160 V Switchgear Room louver spray.
2. Unit 4 - 4160 V Switchgear Room louver spray.
3. Emergency Diesel Generator Building water curtain
4. Control Point Guard House sprinkler system.

b. If Specification 3.14.3.a cannot be met because one or more of the above required spray and/or sprinkler systems is inoperable, establish a fire watch patrol with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, prepare and submit a Report to the Commission pursuant to Specification 6.9.3.c.

4. FIRE HOSE STATIONS

a. Fire hose stations shown in Table 3.14-2 in the vicinity of safety related equipment shall be operable at all times when the safety related equipment in their area of protection is required to be operable, or within one hour of finding an inoperable hose station, an equivalent capacity fire hose shall be run from an equivalent water source to the inoperable location.

5. FIRE BARRIER PENETRATIONS

a. All fire barrier penetrations (including cable penetration barriers, fire doors and fire dampers), in fire zone boundaries, protecting safety related areas shall be functional at all times.

b. If Specification 3.14.5.a cannot be met because one or more of the above required fire barrier penetrations is non-functional, within one hour either;

   1. Establish a continuous fire watch on at least one side of the affected penetration, or

   2. Verify the OPERABILITY of fire detectors on at least one side of non-functional fire barrier and establish a hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, prepare and submit a Report to the Commission pursuant to Specification 6.9.3.c.

TABLE 4.2-3

STEAM GENERATOR TUBE INSPECTION

SAMPLE SIZE	1st SAMPLE INSPECTION		2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 2S tubes tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Paragraph 50.72(b)(2) of 10 CFR 50	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to Paragraph 50.72(b)(2) of 10 CFR 50	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

4.2.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Changes, Tests and Experiment Reports for the period in which this inspection was completed. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 of Table 4.2-3 shall be reported to the Commission pursuant to Specification 6.9.3.k.

4.11 REACTIVITY ANOMALIES

Applicability: Applies to potential reactivity anomalies.

Objective: To require evaluation of reactivity anomalies within the reactor.

Specification: Following a normalization of the computed boron concentration as a function of burnup, the actual concentration in the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady state values reaches the equivalent of one percent in reactivity, make an evaluation as to the cause of the discrepancy and prepare and submit a Report to the Commission pursuant to Specification 6.9.3.m.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager - 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 shown on Figure 6.2-1.

6.2.2 FACILITY STAFF

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

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 trip.

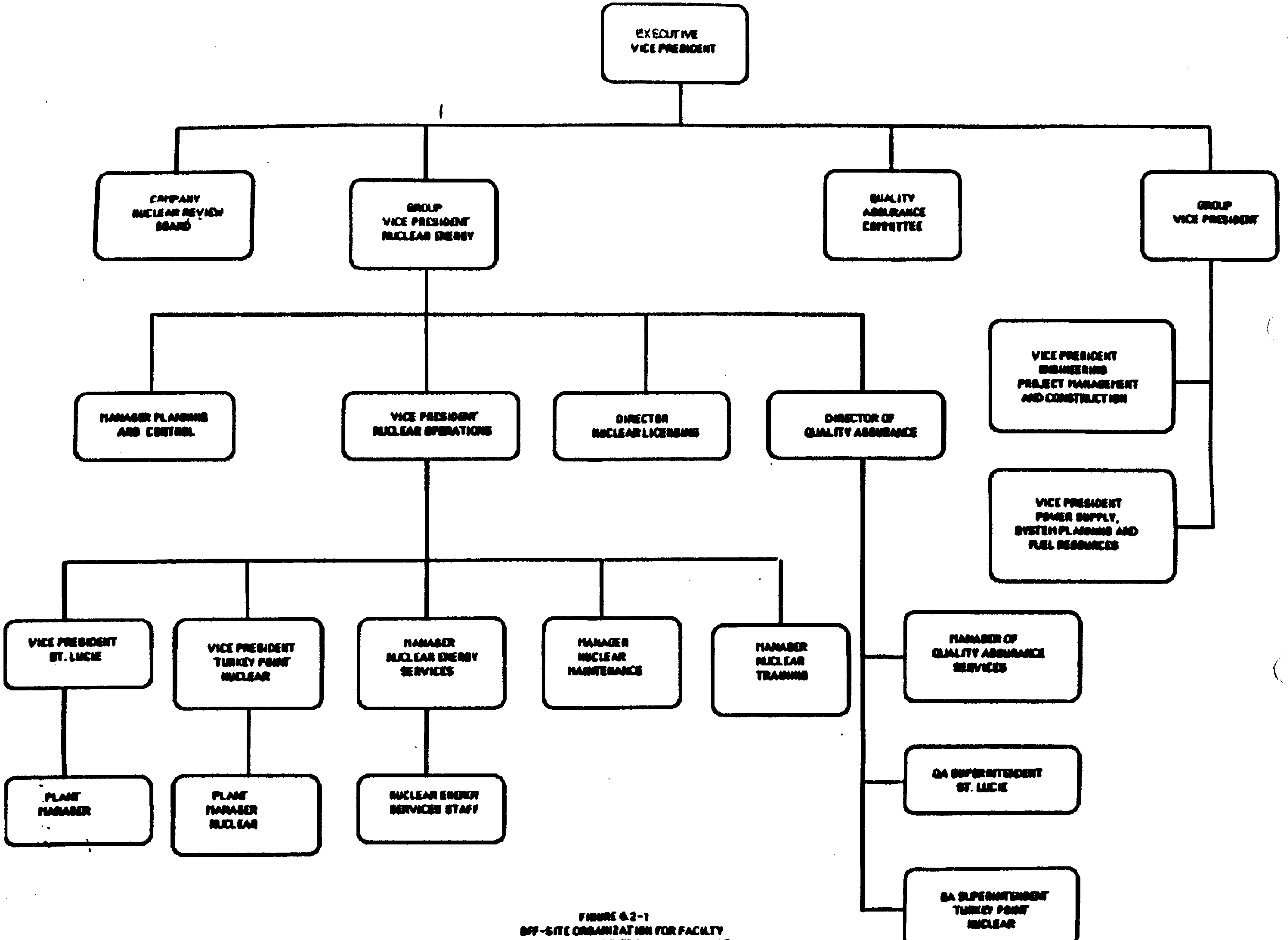
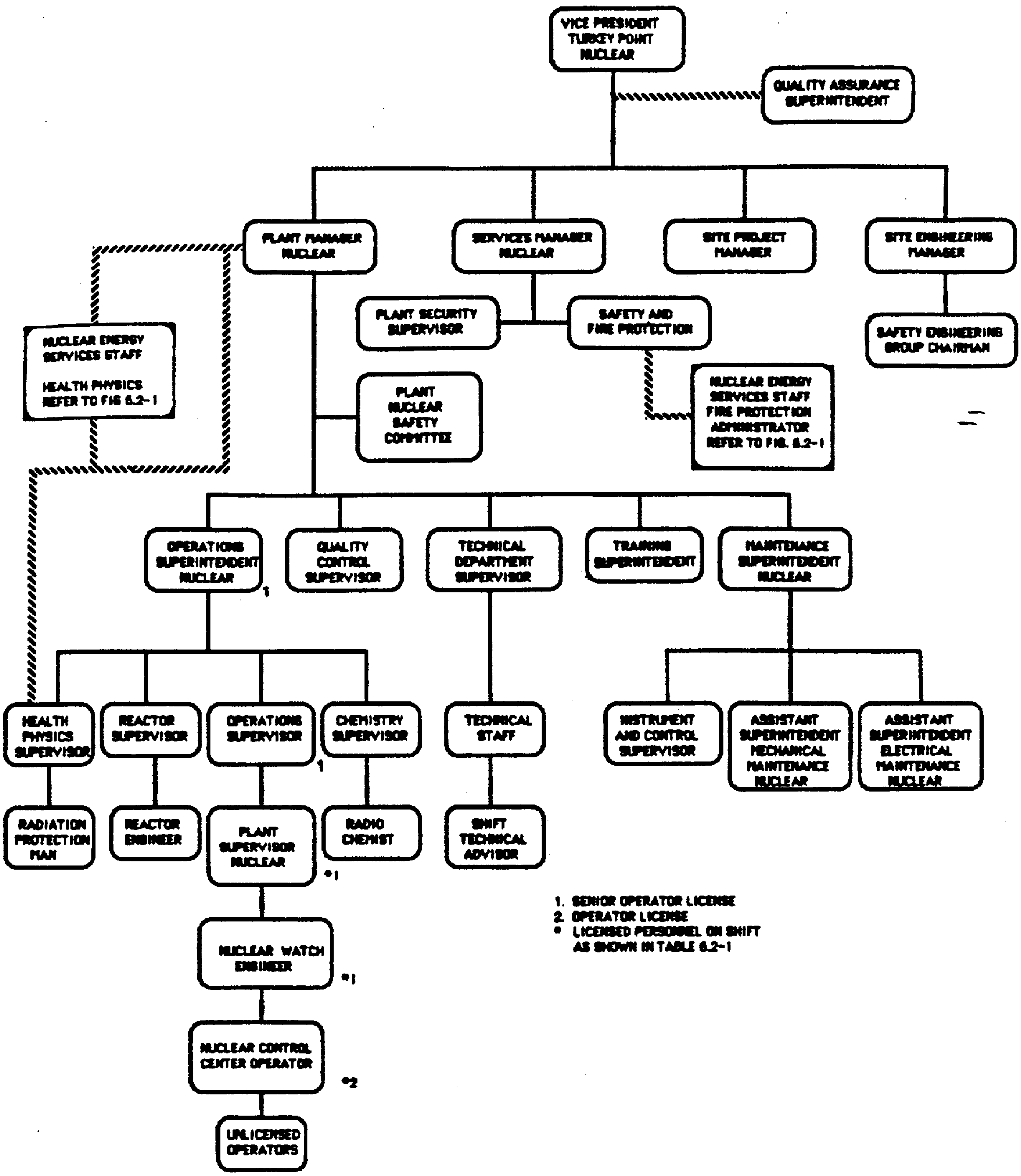


FIGURE 6.2-1
OFF-SITE ORGANIZATION FOR FACILITY
MANAGEMENT AND TECHNICAL SUPPORT



1. SENIOR OPERATOR LICENSE
2. OPERATOR LICENSE
* LICENSED PERSONNEL ON SHIFT AS SHOWN IN TABLE 6.2-1

PLANT ORGANIZATION CHART
FIGURE 6.2-2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

LICENSE CATEGORY QUALIFICATIONS	ONE OR TWO UNITS OPERATING[A]	ALL UNITS SHUTDOWN
SRO*	2	1 **
RO	3	2
Non-Licensed Auxiliary Operators	3	3
Shift Technical Advisor	1+	None Required

\+ This position may be filled by one of the SROs above, provided the individual meets the qualification requirements of 6.3.1.

* Includes the licensed Senior Reactor Operator serving as Shift Supervisor.

** Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising the movement of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel.

A Operating is defined as $K_{eff} \geq 0.99$, % thermal power excluding decay heat greater than or equal to zero, and an average coolant temperature $T_{avg} \geq 200$ F.

\# Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

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.

f. A site Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include 2 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

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 except for the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.

6.3.2 Health Physics Supervisor Qualifications

6.3.2.1 The Health Physics Supervisor at the time of appointment to the position, shall, except as indicated below, meet the following:

1. He shall have a bachelor's degree or equivalent in a science or engineering subject, including some formal training in radiation protection.
2. He shall have five years of professional experience in applied radiation protection; where a master's degree in a related field is equivalent to one year experience and a doctor's degree in a related field is equivalent to two years of experience.
3. Of his five years of experience, three years shall be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered at Turkey Point Plant.

6.3.2.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC Office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.2.1.

* Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accomodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

6.4 TRAINING

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.

6.4.2 A training program for the fire brigade shall be maintained under the direction of the Fire Protection Adminstrator and shall meet or exceed the requirements of 10 CFR 50.48 and 10 CFR 50 Appendix R.

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 Manager - 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 Manager - Nuclear
2. Vice Chairman: Operations Superintendent - Nuclear
3. Technical Department Supervisor
4. Maintenance Superintendent - Nuclear
5. Instrument and Control Supervisor
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 Manager - 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.

e. Investigation of all violations of the Technical Specifications and preparation and forwarding a report covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Operations, to the Group Vice President - Nuclear Energy 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.

j. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

k. Review of all REPORTABLE EVENTS.

6.5.1.7 AUTHORITY

The Plant Nuclear Safety Committee shall:

a. Recommend to the Plant Manager - 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 Vice President of Nuclear Operations and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Manager - Nuclear; however, the Plant Manager - Nuclear 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 Vice President - Nuclear Operations and Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

6.5.2.1 FUNCTION

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 following members:

1.	Chairman:	Group Vice President - Nuclear Energy
2.	Member:	Vice President - Nuclear Operations
3.	Member:	Vice President - Engineering, Project Management, and Construction
4.	Member:	Chief Engineer - Power Plant Engineering
5.	Member:	Director - Nuclear Licensing
6.	Member:	Director - Quality Assurance
7.	Member:	Manager - Nuclear Energy Services
8.	Member:	Manager - Nuclear Fuels
9.	Member:	Senior Project Manager - Power Plant Engineering
10.	Member:	Group Vice President

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 as voting members in CNRB activities at any one time.

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, 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. All REPORTABLE EVENTS.

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

o. 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.

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 year.

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

g. The Facility Fire Protection Program and implementing procedures at least once per two years.

h. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified licensee personnel or an outside fire protection firm.

i. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than three (3) years.

j. The radiological environmental monitoring program and the results thereof, at least once per year.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once every two years.

l. The PROCESS CONTROL PROGRAM and implementing procedures for dewatering of radioactive bead resin at least once every two years.

m. The performance of activities required by the Quality Control Program to meet the criteria of Regulatory Guide 1.21, Revision 1 June 1974 and Regulatory Guide 4.1, Revision 1, April 1975, at least once per year.

n. 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:

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 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PNSC, and the results of this review shall be submitted to the CNRB, the Vice President - Nuclear Operations, and the Group Vice President - Nuclear Energy.

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 with immediately.

b. The Safety Limit violation shall be reported immediately to the Commission, the Vice President of Nuclear Operations 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 Nuclear Operations 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, Appendix "A" of USNRC Regulatory Guide 1.33, PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, Quality Control Program for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974, Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975, and the Facility Fire Protection Program 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, except the Quality Control Program for environmental monitoring, shall be reviewed by the PNSC and approved by the Plant Manager - Nuclear 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:

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 Operators License on the unit affected.

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

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington DC. pursuant to 10 CFR 50.4.

6.9.1 ROUTINE REPORTS

a. Startup Report - A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. A tabulation of occupational exposure data shall be submitted annually.

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing and refueling. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report - Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the appropriate regional offices, to arrive no later than the fifteenth of each month following the calendar month covered by the report.

6.9.2 (Deleted)

1. This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

6.9.3 SPECIAL REPORTS

Special reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate.

a. In-service inspection, reference 4.2

b. Tendon surveillance, reference 4.4

c. Fire protection systems, reference 3.14.

d. Peaking Factor Limit Report - The W(Z) function(s) for Base-Load Operation corresponding to a $\pm 2\%$ band about the target flux difference and/or a $\pm 3\%$ band about the target flux difference, the Load-Follow function $F_Z(Z)$ and the augmented surveillance turnon power fraction, P_T, shall be provided to the U.S. Nuclear Regulatory Commission, at least 60 days prior to cycle initial criticality, whenever P_T is , 1.0. In the event, the option of Baseload Operation (as defined in Section 3.2.6.a "3&) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_Z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the values would be submitted 60 days prior to the date the values would become effective unless otherwise approved by the Commission.

e. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding the limits of Technical Specifications 3.9.1.b, 3.9.2.b, or 3.9.2.c, submit a report which identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the limits of 3.9.1.b, 3.9.2.b, or 3.9.2.c.

f. With untreated radioactive liquid effluents exceeding the limits of 3.9.1.d pursuant to Specification 3.9.1.d.3, submit a report which includes the following information:

   (1) Identification of the inoperable equipment or subsystems and the reason for inoperability,

   (2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   (3) Summary description of action(s) taken to prevent a recurrence.

g. With untreated gaseous effluents exceeding the limits of 3.9.2.e pursuant to Specification 3.9.2.e.2, submit a report which includes the following information:

(1) Identification of the inoperable equipment or subsystems and the reason for inoperability,

(2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and

(3) Summary description of action(s) taken to prevent a recurrence.

h. With the annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC from all uranium fuel cycle sources exceeding the limits of Technical Specification 3.9.2.h, submit a report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.9.2.h and includes the schedule for achieving conformance with those limits. This report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the limits of Specification 3.9.2.h and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

i. With the measured levels of radioactivity in environmental samples as a result of plant effluents pursuant to Specification 4.12.1.b, submit a report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential dose to a MEMBER OF THE PUBLIC is less than the limits of Specifications 3.9.1.b, 3.9.2.b and 3.9.2.c.

j. If the limits of Technical Specification 3.20 are exceeded, submit a report describing the cause of the unavailability, action taken and a schedule for restoration within 30 days.

k. Whenever the results of steam generator tube inspections fall into Category C-3 of Table 4.2-3, a Special Report shall be submitted within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. Reference T.S. 4.2.5.5.c.

l. If the power tilt in Technical Specification 3.2.6.h is not corrected to less than 2% within 24 hours and its design hot channel factors for rated power are not exceeded, a Special Report with an evaluation as to the cause of the discrepancy shall be submitted within 30 days. Reference T.S. 3.2.6.i(1)

m. Following a normalization of the computed boron concentration as a function of burnup, if the difference between the observed and predicted boron concentration reached the equivalent of one percent in reactivity, a Special Report shall be submitted within 30 days. Reference T.S. 4.11.

6.9.4 UNIQUE REPORTING REQUIREMENTS

6.9.4.a SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effleunts released from the unit as outlined in Regulatory guide 1.21, "Measureing, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year.

* A single submittal may be made for a multiple unit station. The submittal should combine those secitons that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

** In lieu of submission with the Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

The Radioactive Effluent Release Report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, and determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods may be used in lieu of actual meteorological measurements. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases for the previous calendar year. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, March 1976.

The Radioactive Effluent Release Report shall also include the results of specific atcitivy analysis in which the primary coolant exceeded the limits of Specification 3.1.4.2.a. The following information shall be included: 1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded, 2) Fuel burnup by core region, 3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded, 4) History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and 5) The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Volume,

b. Total curie quantity (specific whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the OFFSITE DOSE CALCULATION MANUAL (ODCM) and the PROCESS CONTROL PROGRAM (PCP) as well as a listing of new locations for dose calculations and/or environmental monitoring:

a. Necessitated by the unavailability of environmental samples, pursuant to Specification 4.12.1.c; report shall also include the causes for unavailability of samples

b. Identified by the land use census pursuant to Specification 4.12.2.

6.9.4.b <u>ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*</u>

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and information based on trend analysis of the results of the radiological environmental surveillance activities for the report period, including a comparison, as appropriate, with preoperational studies, with operational controls and with previous environmental surveillance reports and an assessment of the observed impact of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 4.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

* A single submittal may be made for a multiple unit station.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions form the centerline of the plant vent stack; the results of the Interlaboratory Comparison Program required by Specification 4.12.3; discussion of all deviations from the sampling schedule of Table 4.12-1, and discussion of all analyses in which the LLD required by Table 4.12-3 was not achievable.

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. All REPORTABLE EVENTS.

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. Records of annual physical inventory verifying accountability of sources on record.

** One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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

a. Records 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.

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 transients 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 except as listed in Specification 6.10.1.

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.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lives of all snubbers required by specification 3.13 including the date of which the service life commences and associated installation and maintenance records.

n. Annual Radiological Environmental Monitoring Reports and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Monitoring Report.

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 HIGH RADIATION AREA

6.12.1 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 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 6.12.1(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.

6.13 POST ACCIDENT SAMPLING

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis,
3. Provisions for maintenance of sampling and analysis equipment.

6.14 SYSTEMS INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.15 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

6.16 BACKUP METHODS FOR DETERMINING SUBCOOLING MARGIN

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and
2. Procedures for monitoring.

6.17 PROCESS CONTROL PROGRAM (PCP)

6.17.1 The PCP shall be reviewed by PNSC prior to implementation.

6.17.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

2. Shall become effective upon reivew and acceptance by the PNSC.

6.18 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.18.1 The ODCM shall be reviewed by the PNSC prior to submittal to the Commission.

6.18.2 The ODCM shall be approved by the Commission prior to implementation.

6.18.3 Licensee initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   3. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

B3.14 BASES FOR FIRE PROTECTION SYSTEMS

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, fire hose stations and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility.

This design feature minimized the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, the fire detectors on at least one side of the affected barrier must be verified OPERABLE and a hourly fire watch patrol established, until the barrier is restored to functional status.

B4.2 BASES FOR REACTOR COOLANT SYSTEM IN-SERVICE INSPECTION

This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. This is accomplished by referencing the inspections required by Specification 4.0.3.

MISCELLANEOUS INSPECTIONS

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, of Table 4.2-3 these results shall be reported to the Commission pursuant to Specification 6.9.3.k. 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.

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, of Table 4.2-3, these results shall be reported to the Commission pursuant to Specification 6.9.3.k.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO.116 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 BACKGROUND

On December 19, 1983, the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 83-43. This letter notified licensees of changes in reporting requirements of 10 CFR Part 50, Sections 50.72 and 50.73. These changes are designed to provide the Commission with more useful reports regarding the safety of operating nuclear power plants. These changes will require early reporting only on those matters of value to the exercise of the Commission's responsibilities.

II DISCUSSION

By letter dated August 20, 1985 as supplemented on May 13, 1986, Florida Power and Light Company (the licensee) provided their response to GL 83-43. The licensee's submittal also requested that the Technical Specifications be modified to reflect the current off-site Organization, reporting of reactor coolant specific activity as per the guidance provided in GL 85-19 "Reporting Requirements on Primary Coolant Iodine Spikes," and to correct appropriate references consistent with the amendment request.

The initial request also had changes relating to fire protection requirements as defined in 10 CFR 50, Appendix R. We indicated in our renotice of No Significant Hazards Consideration, (50 FR 43679) published in the Federal Register on December 3, 1986 that the licensee was in the process of proposing an amendment to the Technical Specifications which would delete the existing Appendix R Technical Specifications, incorporate the Appendix R requirements in the Final Safety Analysis Report (FSAR) and maintain their approved fire protection program as described in the FSAR. This would be accomplished in accordance with the guidance provided in GL 86-10. As stated in the notice the staff has not acted on changes relating to fire protection requirements. However, as noted in the evaluation portion of this Safety Evaluation, we have corrected a typographical error relating to reporting requirements in previously approved Amendment Nos. 45 and 37.

Subsequent to the licensee's submittal, the Final Rule "Communications Procedures Amendments, 10 CFR Part 50" became effective. This rule supersedes all existing requirements and guidance with respect to the number of copies submitted to the NRC by the licensee and mailing procedures. The Commission's guidance documents dealing with communications procedures will be revised to conform with the rule. Licensees whose Technical Specifications contain conflicting submittal directions are authorized by this rule to delete the conflicting directions by pen-and-ink changes to their Technical Specifications. The Commission does not expect formal applications for amendment of license to result from this rulemaking. However, in this case the licensee's proposed amendment, modifying certain technical specifications to conform to 10 CFR 50.72 and 50.73, contained submittal directions conflicting with the new communications rule. Rather than issue the technical specifications with incorrect submittal directions, the submittal directions have been revised to conform to the new Communications rule. These changes do not affect the substance of the amendments as noticed.

If my changes accurately reflect what you intended to do, please make appropriate changes. With these changes, the package is okay and there is no need to return to OGC.

III EVALUATION

The evaluation of the proposed changes are grouped into the following areas:

A. GL 83-43 changes
B. Off-site Organization changes
C. 10 CFR Part 50.4 "Written Communications" changes per the Final Rule
D. Table of Contents, Bases and Format Changes

A. GL 83-43 Changes

Page 1-3. The definition of Reportable Event was added to be consistent with 10 CFR 50.73.

Page 3.1-5. The requirement to submit a reportable occurrence report for primary coolant specific activity is deleted. This is in accordance with GL 85-19 which requires that primary coolant specific activity need only be reported annually. Should the high specific activity last longer than the time allowed in Technical Specification (TS) 3.1.4.1, a shutdown would be required and a report per 10 CFR 50.72 would be submitted with full details of the event. The records of high activity for shorter periods would be available on site for review. The information required for TS 3.1.4.2 will now be included in the Radioactive Effluent Release Report.

Page 3.2-7. The requirement to submit an abnormal occurrence report to the NRC for power tilt is revised to require a Special Report. This is consistent with GL 83-43.

Pages 3.14-1, 3.14-2 and 3.14-3. These pages have been corrected to reference Section 6.9.3.c for reporting requirements. The previous reference was

Section 6.9.3.c for reporting requirements. The previous reference was Section 6.9.2.b which identified occurrences which required thirty day written notice. This was inconsistent with Section 6.9.3.c which indicated that fire protection systems be special reports. This is no change from the existing requirement and corrects a previous typographical error which existed in the Technical Specifications when Amendment Nos. 45 and 37 were approved.

Page 4.2-6 and Table 4.2-3. The prompt reportable occurrence report required for steam generator inspections which fall into Category C-3 is changed to a Special Report. This is consistent with our current requirements.

Page 4.11-1. The 24-hour report with written followup for reactivity anomalies is changed to a Special Report in accordance with GL 83-43.

Page 6-8. Specification 6.5.1.6 is revised in accordance with GL 83-43 to add the requirement for the plant Nuclear Safety Committee to review Reportable Events.

Page 6-11. Specification 6.5.2.7.g is revised to require CNRB review of Reportable Events in accordance with GL 83-43.

Page 6-14. Specification 6.6 is revised in accordance with GL 83-43 to address Reportable Event actions.

Page 6-16. Specification 6.9.2, Reportable Occurrences, is deleted in accordance with GL 83-43. The redundant reference to Section 20.407 of 10 CFR Part 20 is also deleted.

Pages 6-18 and 6-19. Special Reports are added to Specifcation 6.9.3 to be consistent with the proposed specifications in accordance with GL 83-43.

Page 6-21. Specification 6.9.4.a is changed to be consistent with the changes on Page 3.1-5 discussed above and is consistent with the guidance provided in GL 85-19.

Pages 6-23 and 6-24. The record retention requirements for Reportable Events is revised in accordance with GL 83-43.

The proposed changes discussed above are in response to GL 83-43 and are consistent with the guidance provided in the GL. The changes are also consistent with the reporting requirements of 10 CFR 50.72 and 50.73 and are therefore acceptable.

B. Off-site Organization Changes

Pages 6-6, 6-7, 6-8, 6-9, 6-10 and Figures 6.2-1 and 6.2-2. The additions include a Group Vice President for the engineering, project management, construction, power supply, system planning and fuel resources areas, and a Site Engineering Manager reporting to the Vice President Turkey Point Nuclear. The Group Vice President replaced the Power Plant Engineering Principal Engineer as a member of the Company Nuclear Review Board (CNRB). The composition of the CNRB (6.5.2.2, Page 6-9) is changed to reflect this. Specification 6.5.2.3 is revised to reflect the NRC staff's current position on the participation of alternate members on the CNRB. Other pages in Section 6 of the Technical Specifications have been corrected to reflect corrected titles. Figures 6.2-1 and 6.2-2 reflect the current licensee's organization.

We conclude that the organization structure changes as discussed above do not diminish FPL's ability to safely operate the Turkey Point Units 3 & 4 and the proposed modifications to Figures 6.2-1 and 6.2.-2 to reflect these changes are acceptable. The corporate position titles of the designated CNRB members are such that the incumbents are expected to be qualified to perform the independent review functions and these changes are therefore acceptable.

C. 10 CFR 50.4, Written Communications Changes per the Final Rule.

Pages 6-15, 6-16 and 6-17. These pages have been modified in respect to the number of copies and the mailing procedures as specified in the Final Rule (10 CFR 50.4). These changes were provided with this amendment to delete conflicting submittal directions. The submittal directions are consistent with the Final Rule and are therefore acceptable.

D. Table of Contents, List of Figures, Bases and Format-only changes.

The Table of Contents and List of Figures (Pages i, ii, iii and iv) are revised to be consistent with the changes requested and the existing Turkey Point Technical Specifications. The entire Section 6, "Administrative Controls," has been reformatted and consists of less pages. Pages 6-1, 6-5, 6-6, 6-12, 6-13, 6-20, 6-22, 6-25, 6-26, 6-27 and Table 6.2-1 have no changes to the content of the Technical Specifications. These pages have been retyped to be consistent with the format of Section 6. The Bases Section, Page B3.14-1 has been updated to indicate that immediate corrective actions must be taken if a fire suppression water system becomes inoperable. This change is consistent with the staff's fire protection evaluations. Pages B4.2-1 and B3.14-1 have been modified to reference the correct reporting requirements of Section 6 of the Technical Specification.

The staff has determined that these changes support the proposed revisions to the Technical Specifications previously discussed and are therefore acceptable.

Based on the above evaluation, we conclude that the proposed changes are in accordance with the guidance provided in GL 83-43 and GL 85-19; do not diminish FPL's ability to safely operate the Turkey Point Facility; deletes conflicting submittal directions; and are therefore acceptable.

IV. <u>ENVIRONMENTAL CONSIDERATION</u>

These amendments involve changes in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. 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 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: March 6, 1987

Principal Contributors:
P. Moore
D. McDonald