DISTRIBUITON
Docket File
NRC PDR
Local PDR
NSIC
TERA
ORB 1 File
D. Eisenhut
C. Parrish
M. Grotenhuis
OELD
OI&E (4)
G. Deegan (8)
B. Scharf (10)
J. Wetmore
ACRS (10)
R. Diggs

NOV 4 1981

RECEIVED
NOV 19 1981
U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-250
and 50-251

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company
Post Office Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

We have completed our review of your request for an exemption from the requirements of Appendix J, for the Turkey Point Plant Unit Nos. 3 and 4. The request for exemption was with respect to testing frequency and method of testing as outlined in your letters dated September 12, 1975, and July 27, 1977. We have also evaluated your most recent submittal dated November 26, 1980 and find the alternative you proposed acceptable. Furthermore, no exemption from the requirements of Appendix J is necessary because your proposed alternative is within the revised version of Appendix J to 10 CFR Part 50 Section III.D.2 (effective October 22, 1980).

The Commission has issued the enclosed Amendment No. 73 to Facility Operating License No. DPR-31 and Amendment No. 67 to Facility Operating License No. DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated September 20, 1974, as supplemented on July 27, 1977.

These amendments revise the Technical Specifications to: (1) include the air lock testing according to Appendix J to 10 CFR Part 50; (2) make certain corrections in terminology to be consistent with Appendix J; and (3) make certain administrative corrections.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By:

Marshall Grotenhuis, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 73 to DPR-31
2. Amendment No. 67 to DPR-41
3. Safety Evaluation
4. Notice of Issuance

SEE PREVIOUS 318 FOR CONCURRENCES*

OFFICE	cc: See next page	ORB 1*	ORB 1	ORB 1*	AD:OR*	OELD*
SURNAME	8112070128 811104 PDR ADOCK 05000250 P PDR	CParrish	MGrotenhuis/rs	SVarga	TNovak	
DATE		10/20/81	11/4/81	10/29/81	10/29/81	11/2/81

DISTRIBUTION
Docket File
NRC PDR
Local PDR
NSIC
TERA
ORB 1 File
D. Eisenhut
C. Parrish
M. Grotenhuis
OELD
OI&E (4)
G. Deegan (8)
B. Scharf (10)
J. Wetmore
ACRS (10)
R. Diggs

Docket Nos. 50-250
and 50-251

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company
Post Office Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

We have completed our review of your request for an exemption from the requirements of Appendix J, for the Turkey Point Plant Unit Nos. 3 and 4. The request for exemption was with respect to testing frequency and method of testing as outlined in your letters dated September 12, 1975, and July 27, 1977. We have also evaluated your most recent submittal dated November 26, 1980 and find the alternative you proposed acceptable. Furthermore, no exemption from the requirements of Appendix J is necessary because your proposed alternative is within the revised version of Appendix J to 10 CFR Part 50 Section III.D.2 (effective October 22, 1980).

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 Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated September 20, 1974, as supplemented on July 27, 1977.

These amendments revise the Technical Specifications to (1) include the air lock testing according to Appendix J; (2) to make certain corrections in terminology to be consistent with Appendix J; and (3) to make certain administrative corrections.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Marshall Grotenhuis, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. to DPR-31
2. Amendment No. to DPR-41
3. Safety Evaluation
4. Notice of Issuance

OFFICE ▶	cc: See next page		ORB 1	ORB 1	ORB 1	AD:OR	OELD
SURNAME ▶			CParrish	MGrotenhuis/rs	SVarga	TNovak	Young FRN and admin only
DATE ▶			10/27/81	10/28/81	10/29/81	10/29/81	10/2/81

make noted corrections

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1980—335-960

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. 73
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 September 20, 1974, as supplemented on July 27, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, 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. 73, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 4, 1981

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. 67
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 20, 1974, as supplemented on July 27, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and 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. 67, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 4, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-3	1-3
1-6	1-6
2.1-1	2.1-1
T3.5-2	T3.5-2
3.6-1	3.6-1
3.6-2	3.6-2
3.10-2	3.10-2
4.4-1	4.4-1
4.4-2	4.4-2
4.4-3	4.4-3
4.4-4	4.4-4
4.4-5	4.4-5
4.4-6	4.4-6
4.4-7	4.4-7
4.7-1	4.7-1
4.8-1	4.8-1
4.8-2	4.8-2
4.10-1	4.10-1
B.2.3-2	B.2.3-2

number of operable channels and the number of channels which when tripped will cause reactor trip.

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 SHUTDOWN

1) Cold Shutdown

The reactor is in the cold shutdown condition when the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is less than 200F.

2) Hot Shutdown

The reactor is in the hot shutdown condition when it

1.16 FUEL RESIDENCE TIME LIMIT

The reactor shall not be operated with less than three reactor coolant pumps in operation.

1.17 LOW POWER PHYSICS TESTS

Low power physics tests are tests below a nominal 5% of rated power which measure fundamental characteristics of the reactor core and related instrumentation.

1.18 ENGINEERED SAFETY FEATURES

Features such as containment, emergency core cooling, and containment atmospheric cleanup systems for mitigating the consequences of postulated accidents.

1.19 REACTOR PROTECTION SYSTEM

Systems provided to act, if needed, to avoid exceeding a safety limit in anticipated transients and to activate appropriate engineered safety features as necessary.

1.20 SAFETY RELATED SYSTEMS AND COMPONENTS

Those plant features necessary to assure the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents which could result in off-site exposures comparable to the guideline exposures of 10 CFR 100.

1.21 PER ANNUM

During each calendar year.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability: Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, average coolant temperature and flow during power operation.

Objective: To maintain fuel cladding integrity.

Specification:

1. THREE LOOP OPERATION

   The combination of thermal power level, coolant pressure and average coolant temperature shall not exceed the limits shown in Figure 2.1-1 for full flow from three reactor coolant pumps.

2. TWO LOOP OPERATION

   The combination of thermal power level, coolant pressure and average coolant temperature shall not exceed the limits shown in Figure 2.1-2 for full flow from two reactor coolant pumps.

3. ONE LOOP OPERATION

   The thermal power level shall not exceed 20%, coolant pressure shall be maintained in the 1820 - 2400 psig range, and the average coolant temperature shall not exceed 590 F for full flow from one reactor coolant pump.

4. NATURAL CIRCULATION

   The thermal power level shall not exceed 12%, coolant pressure shall be maintained in the 2135 - 2400 psig range and the average coolant temperature shall not exceed 602 F, when no reactor coolant pumps are in operation.

TABLE 3.5-2

ENGINEERED SAFETY FEATURES ACTUATION

NO.	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 MIN. DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1.	SAFETY INJECTION			
1.1	Manual	1	0	Cold Shutdown
1.2	High Containment Pressure	2	1	Cold Shutdown
1.3	High Differential Pressure between any Steam Line and the Steam Line Header	2	1	Cold Shutdown
1.4	Pressurizer Low Pressure*	2	1	Cold Shutdown
1.5	High Steam Flow in 2/3 Steam Lines with Low T_{avg} or Low Steam Line Pressure	1/line in each of 2 lines	1	Cold Shutdown
2.	CONTAINMENT SPRAY			
2.1	High Containment Pressure and High-High Containment Pressure (Coincident)	2 per set	1/set	Cold Shutdown
3.	AUXILIARY FEEDWATER			
3.1	Low-Low Steam Generator Level	2	1	Hot Shutdown
3.2	Loss of Voltage (both 4KV busses)	2	0	Cold Shutdown
3.3	Safety Injection		(---See 1 above---)	
3.4	Trip of both Main Feedwater Pump Breakers	2	0	Cold Shutdown

* This signal may be manually bypassed, when the reactor is shut down and pressure is below 1800 psig.

3.6 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability: Applies to the operational status of the Chemical and Volume Control System.

Objective: To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification:

a. When fuel is in the reactor there shall be at least one flow path to the core for boron injection.

b. A reactor shall not be made critical unless the following Chemical and Volume Control System conditions are met:

1. TWO associated charging pumps shall be operable.

2. TWO boric acid transfer pumps shall be operable.

3. The boric acid tanks in service shall contain a total of at least 3,080 gallons of a 20,000 to 22,500 ppm boron solution at a temperature of at least 145 F.

4. System piping, interlocks and valves shall be operable to the extent of establishing one flow path from the boric acid tanks, and one flow path from the refueling water storage tank, to the Reactor Coolant System.

5. TWO channels of heat tracing shall be operable for the flow path from the boric acid tanks.

6. The primary water storage tank contains not less than 30,000 gallons of water.

c. The second reactor shall not be made critical unless the following conditions are met:

1. TWO associated charging pumps shall be operable.

2. THREE boric acid transfer pumps shall be operable.

3. The boric acid tanks in service shall contain a total of at least 6160 gallons of a 20,000 to 22,500 ppm boron solution at a temperature of at least 145 F.

4. System piping, interlocks and valves shall be operable to the extent of establishing one flow path from the boric acid tanks, and one flow path from the refueling water storage tank, to each Reactor Coolant System.

5. TWO channels of heat tracing shall be operable for the flow path from the boric acid tanks.

6. The primary water storage tank contains not less than 30,000 gallons of water.

d. During power operation, the requirements of 3.6.b and c may be modified to allow one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.6b and c within the time period specified, the reactor(s) shall be placed in the hot shutdown condition. If the requirements of 3.6.b and c are not satisfied within an additional 48 hours, the reactor(s) shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.6.d.

1. One of the two operable charging pumps may be removed from service provided that it is restored to operable status within 24 hours.

2. One boric acid transfer pump may be out of service provided that it is restored to operable status within 24 hours.

3. One channel of heat tracing may be out of service for 24 hours.

5. At least ONE residual heat removal pump shall be in operation, unless reactor coolant temperature is less than 160F.

6. When the reactor vessel head is removed and fuel is in the vessel, the minimum boron concentration of 1950 ppm shall be maintained in the reactor coolant system and verified daily.

7. Direct communication between the control room and the refueling cavity manipulator crane shall be available during refueling operation.

8. The spent fuel cask shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

9. Fuel which has been discharged from a reactor will not be moved outside the containment in fewer than 100 hours after shutdown.

If any one of the specified limiting conditions for refueling is not met, refueling shall cease until specified limits are met, and there shall be no operations which may increase reactivity.

4.4 CONTAINMENT TESTS

Applicability: Applies to containment integrity testing, tendon surveillance, end anchorage concrete surveillance, and liner surveillance.

Objective: To verify that potential leakage from the containment and the tendon loading are maintained within specified limits.

4.4.1 INTEGRATED LEAKAGE RATE TEST - POST OPERATIONAL

Post Operational Containment Integrated Leak Rate Tests shall be performed and reported in accordance with 10 CFR 50, Appendix J, (type A tests).

Pa, the peak calculated containment internal pressure related to the design basis accident is 49.9 psig.

Pt, the containment vessel reduced test pressure is 25 psig.

La, the maximum allowable leakage rate at pressure Pa is 0.25 weight percent of containment atmosphere per day.

4.4.2 LOCAL PENETRATION TESTS

Test Procedure and Frequency

Local leak detection tests of the following components shall be performed at a pressure not less than 50 psig using pressure decay, soap bubble, halogen detection or equivalent methods at the frequency listed, unless otherwise noted:

1. Containment purge valves (pressure applied in connecting duct) - each refueling.

2. Personnel and Emergency Airlocks

   a. *Within 3 days of every first of a series of openings when containment integrity is required, verify that door seals have not been damaged or seated improperly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

   b. At least once per 6 months, conduct an overall airlock leakage test to verify that the overall airlock leakage rate is within its limit.

3. Equipment access opening (pressure applied between gaskets) - annually and after use.

4. Fuel transfer tube flange (pressure applied between gaskets) - each refueling.

5. Electrical penetrations (pressure applied to canister) - each refueling

Acceptance Criteria

Repairs and tests shall be made whenever the sum of the local leak rate tests, including the isolation valves discussed in 4.4.4, exceeds sixty percent of the total containment allowable leak rate.

4.4.3 ISOLATION VALVES

Containment isolation valves shall be tested in accordance with 10 CFR 50, Appendix J, (type C tests).

4.4.4 RESIDUAL HEAT REMOVAL SYSTEM

a. The portion of the Residual Heat Removal System that is downstream of the first isolation valve outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified below.

b. Visual inspection shall be made for excessive leakage from components of the system. Any visual leakage that cannot be stopped at test conditions shall be measured by collection and weighing or by another equivalent method.

c. The acceptance criterion is that maximum allowable leakage from the Residual Heat Removal System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

d. Repairs shall be made as required to maintain leakage with the acceptance criterion in (c) above.

e. If repairs are not completed within 7 days, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion in (c) above is satisfied.

f. Tests of the Residual Heat Removal System shall be conducted each refueling.

4.4.5 TENDON SURVEILLANCE

Lift-off

Lift-off readings will be taken for the following nine (9) tendons available for inspection:

	Unit 3	Unit 4
Horizontal	62H18,42H70,64H51	62H38,42H80,64H70
Vertical	23V1,45V7,61V1	12V29,34V29,56V29
Dome	1D53,2D28,3D28	1D28,2D 3,3D28

Wire Inspection

One horizontal, one vertical and one dome tendon will be relaxed and one wire will be removed from each as a sample. (At subsequent inspections different tendons will be used for the sample). Wires will be visually inspected for corrosion and pitting. Tensile tests will be performed on three (3) samples cut from each wire (one from each end and one from the middle) of a length equal to the maximum length acceptable for the test apparatus to be used.

After samples are taken, tendons will be re-tensioned and final lift-off readings will be taken.

Test Frequency

Lift-off readings and wire inspection will take place at the end of the first, third and every fifth year thereafter from the date of the structural integrity test (July 4, 1971, for Unit 3 and February 19, 1972, for Unit 4). Tendon surveillance may be conducted during reactor operation.

Additional Surveillance on Unit 3 Dome

On Unit 3 dome 12 tendons (including the three listed in the first paragraph under 4.4.6) will be subjected to surveillance testing at 6, 12, 24 and 36 months after the structural integrity test (July 4, 1971, for Unit 3). The additional tendons are: 1D15, 1D18, 1D36, 2D24, 2D11, 2D21, 3D4, 3D21 and 3D24.

Lift-off readings will be taken on each of these tendons. The decrease in prestress force measured from $0.73f's\,A$ will be recorded and compared with the predicted loss, for the period the tendons were stressed.

The surveillance tendons will be stressed to $0.8f's$, and the elongation recorded, the tendons will then be relaxed and observation will be made at the stressing washer for any indications of a broken wire. The tendons will be retensioned and lift-off readings taken.

Wire Inspection

One wire each will be removed from three tendons, not listed in the first paragraph of 4.4.6 (one from each directional group); wires will be visually inspected for corrosion and pitting. Tensile tests will be performed on three (3) samples cut from each wire (one from each end and one from the middle) of a length equal to the maximum length acceptable for the test apparatus to be used.

After the samples are taken, the tendons will be retensioned and final lift-off readings taken.

4.4.6 END ANCHORAGE CONCRETE SURVEILLANCE

The following end anchorages will be subject to surveillance at the 346° buttress on Unit 3 and the 194° buttress on Unit 4:

Elevations 14'-0", 35'-0", 60'-0", 85'-0", 110'-0", 152'-0" and in the tendon inspection gallery of each unit at tendon numbers:

12V11, 12V23, 23V9, 23V23, 34V12, 34V28, 45V14, 45V26, 46V24, 56V16, 61V9, 61V26.

The inspection intervals will be approximately one-half year and one year after the structural integrity test (July 4, 1971, for Unit 3 and February 19, 1972, for Unit 4) and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the test. The inspections made shall include:

1. Visual inspection of the end anchorage concrete exterior surfaces.

2. The mapping of the predominant visible concrete crack patterns.

3. The measurement of the crack widths, by use of optical comparators or feeler gauges.

The measurements and observations shall be compared with those to which prestressed structures have been subjected in normal and abnormal load conditions and with those of preceding measurements and observations at the same location on the structures.

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the second inspection and a report will be prepared. If the inspections detect symptoms of greater than normal cracking or movements, an investigation will be made to determine the cause.

4.4.7 LINER SURVEILLANCE

Three representative areas of the liner plate shall be examined and measured for inward deformation (1) prior to the structural integrity test, (2) after the structural integrity test (3) approximately one year after the structural integrity test. Measurements shall be taken between vertical anchors using a straight edge to determine liner profile to within a $\pm$ 0.01 inch accuracy. If changes are less than 0.25 inches no further tests or action is required other than preparation of records. Otherwise an investigation and corrective action will be taken.

Measurements locations shall be:

Elevation	Unit 3	Unit 4
34'0"	70°	70°
62'0"	190°	190°
118'0"	226°	318°

When measurements are made, liner plate and exterior concrete surface temperatures in the area of measurement, and inside and outside ambient temperatures, will be determined and recorded.

The requirements of this Technical Specification have been met.

4.7 EMERGENCY CONTAINMENT FILTERING AND POST ACCIDENT CONTAINMENT VENT SYSTEMS

Applicability: Applies to the Emergency Containment Filtering and the Post Accident Containment Vent System components.

Objectives: To verify that these systems and components will be able to perform their design functions.

Specification: 4.7.1 EMERGENCY CONTAINMENT FILTERING SYSTEM

1. OPERATING TESTS

System tests shall be performed at approximately quarterly intervals. These tests shall consist of visual inspection and pressure drop measurements across each filter bank. Visual inspection shall include inspection of general condition for evidence of: water, oil, or other foreign material; gasket deterioration; adhesive deterioration in the HEPA units; excessive dust cake on the demisters; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across any filter bank shall not exceed two times the pressure drop when new and shall not be less than the pressure drop when new.

2. PERFORMANCE TESTS

During each refueling operation, "in-place" DOP and freon tests shall be conducted at design flow on each unit (all flow paths). 99.9% DOP removal and 99.5% freon removal shall constitute acceptable performance.

Amendment Nos. 73 & 67

4.8 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability: Applies to periodic testing and surveillance requirements of the emergency power system.

Objective: To verify that the emergency power system will respond promptly and properly.

Specification: The following tests and surveillance shall be performed as stated:

1. DIESEL GENERATORS

   a. Each diesel generator shall be manually started and synchronized with normal power sources and loaded to 2750 KW monthly.

   b. Each diesel generator shall be started automatically by a simulated loss of all normal A-C power supplies together with a simulated safety injection signal and loaded sequentially with vital loads during each refueling shutdown. Each diesel shall start and assume loads in the time sequence stated in FSAR Table 8.2-3. The safety injection pumps will be operated using the test lines.

   c. Each diesel generator shall be given a thorough inspection at least annually following the manufacturer's recommendations for this class of stand-by service.

   d. The above tests will be considered satisfactory if all applicable equipment operates as designed.

   e. Diesel generator electric loads shall not be increased beyond 2850 KW during a test. The connected loads shall not be increased above those listed in FSAR Table 8.2-2 during the test in 1.b. above.

f. The diesel fuel oil transfer pumps shall be tested monthly.

2. <u>STATION BATTERIES</u>

a. Pilot cell specific gravities shall be read and recorded daily. The pilot cell shall be rotated on a monthly basis.

b. Monthly each battery shall be given an equalizing charge, and afterwards specific gravity and voltage readings shall be taken and recorded for each cell. Water shall be added to restore normal level and total water use shall be recorded. Complete visual inspection of batteries shall be made monthly.

c. Quarterly detailed visual inspection shall be made of chargers.

d. Annually connections shall be checked for tightness and anti-corrosion coating shall be applied to interconnections.

e. Perform load test annually.

The $f(\Delta q)$ function in the Overpower ΔT and Overtemperature ΔT protection system setpoints includes effects of fuel densification on core safety limits. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and DNBR of 1.30 will not be violated.[10]

Pressurizer

The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[6]

The high pressurizer pressure reactor trip is set below the set pressure of the pressurizer safety valves and limits the reactor operating pressure range. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows margin for instrument error [3] and transient level overshoot before the reactor trips.

Reactor Coolant Flow

The low flow reactor trip protects the core against DNB in the event of loss of one or more reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis.[7] The low frequency and under voltage reactor trips protect against a decrease in flow. The specified set points assure a reactor trip signal before the low flow trip point is reached. The underfrequency trip set point preserves the coastdown energy of the reactor coolant pumps, in case of a system frequency decrease, so DNB does not occur. The undervoltage trip set point will cause a trip before the peak motor torque falls below 100% of rated torque.

Steam Generators

The low-low steam generator water level reactor trip assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting of the auxiliary feedwater system.[8]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-31 AND AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT PLANT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By letter dated August 7, 1975(1), the NRC requested Florida Power & Light Company (FPL) to review its containment leakage testing program for Turkey Point, Units 3 and 4, and the associated Technical Specifications, for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1973. Since there already were many operating nuclear power plants and a number of others in advanced stages of design or construction, the NRC decided to have these plants re-evaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the Turkey Point, Units 3 and 4 licensee. The results of our evaluation are provided below.

The amendments would revise the Technical Specifications to: (1) include the the air lock testing according to Appendix J to 10 CFR Part 50; (2) make certain corrections in terminology to be consistent with Appendix J; and (3) make certain administrative corrections.

2.0 EVALUATION

Our consultant, the Franklin Research Center, has reviewed the licensee's submittals [2, 3, 4, 5] and prepared the attached Technical Evaluation Report (TER) of containment leak rate tests for Turkey Point, Units 3 and 4. We have reviewed this evaluation and concur in its bases and findings.

In the TER, the staff's consultant agreed with the licensee's proposed change to Technical Specification (T.S.) 4.4.2.2 as stated in Reference 4, which requires that airlocks be tested as follows:

"4.4.2 LOCAL PENETRATION TESTS

Test Procedure and Frequency

"Local leak detection tests of the following components shall be performed at a pressure not less than 50 psig using pressure decay, soap bubble, halogen detection or equivalent methods at the frequency listed, unless otherwise noted:

2. Personnel and Emergency Airlocks

   a. Within 3 days of every first of a series of openings when containment integrity is required, verify that door seals have not been damaged or seated improperly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

   b. At least once per 6 months, conduct an overall airlock leakage test to verify that the overall air lock leakage rate is within its limit."

The proposed exemption from the requirements of Appendix J involves testing the airlock seals with a vacuum test instead of a pressure test within 3 days of every first of a series of openings when containment integrity is required.

In a majority of plants, the airlock door seals are tested for proper seating by pressurizing the volume between the inner and outer seals of the inner and outer doors. The inner door seal is on the containment side of the door. With the pressurization of the volume between the seals, the inner seal would tend to be lifted off its seat and the outer seal would tend to be better seated. The pressurization test is conservative because during an accident both the inner and outer seals would tend to be better seated by the containment high pressure during an accident. The vacuum test proposed by the licensee is also a conservative test because it will tend to lift the outer seal and seat the inner seal.

3.0 CONCLUSION

Based on our review of the enclosed technical evaluation report as prepared by our consultant (FRC), the following conclusions are made regarding the Appendix J review for Turkey Point Plant, Units 3 and 4.

1) FPL's proposal to verify that airlock door seals have not been damaged or seated improperly by vacuum testing the volume between the seals within 3 days of every first of a series of openings when containment integrity is required in the interim between full-pressure 6-month tests is an acceptable alternative to the after-each-opening requirement of Appendix J (provided that results are conservatively extrapolated to Pa). No exemption from the requirements of Appendix J is required because of the revision to Section III.D.2 effective October 22, 1980.

2) FPL's proposed change to Technical Specification 4.4.2 is acceptable since it conforms to the requirements of Appendix J except for airlock testing which has been found to be an acceptable alternative.

3) The changes in terminology and other miscellaneous administrative changes not already incorporated in reference 3 are acceptable.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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: November 4, 1981

4.0 REFERENCES

(1) NRC generic letter regarding implementation of 10 CFR Part 50, Appendix J, at TMI-1, dated August 7, 1975.

(2) FPL letter from Dr. R. E. Uhrig to Mr. K. R. Goller dated September 12, 1975.

(3) FPL letter from Dr. R. E. Uhrig to Mr. E. G. Case dated September 20, 1974.

(4) FPL letter from Dr. R. E. Uhrig to Mr. V. Stello dated July 27, 1977.

(5) FPL letter from Dr. R. E. Uhrig to Mr. S. A. Varga dated November 26, 1981.

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4

NRC DOCKET NO. 50-250, 50-251

NRC TAC NO. 08779, 08780 — FRC PROJECT C5257

NRC CONTRACT NO. NRC-03-79-118 — FRC TASK 53/54

Prepared by

Franklin Research Center
The Parkway at Twentieth Street
Philadelphia, PA 19103

Author: T. J. DelGaizo

FRC Group Leader: T. J. DelGaizo

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: Y. S. Huang

June 5, 1981

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway Phila., Pa. 19103 (215) 448-1000

DUPE OF
~~8106110369~~

CONTENTS

1. BACKGROUND

On August 7, 1975 [1], the NRC requested Florida Power & Light Company (FPL) to review the containment leakage testing programs at Turkey Point Units 3 and 4 and to provide a plan for achieving full compliance with 10CFR50, Appendix J, including appropriate design modifications, changes to Technical Specifications, or requests for exemption from the requirements pursuant to 10CFR50.12, where necessary.

FPL responded on September 12, 1975 [2], stating that the containment leakage testing program at Turkey Point Units 3 and 4 conformed to the requirements of Appendix J except for the frequency and method of testing containment airlocks, the frequency of performing Type B electrical penetration leak tests, and minor differences in terminology between the Technical Specifications and Appendix J. FPL indicated that the minor differences in terminology were eliminated by its proposed Technical Specification change of September 20, 1974 [3].

FPL's letter of July 27, 1977 [4] provided additional information regarding proposed testing of containment airlocks. This letter also indicated that Type B electrical penetrations would be tested every refueling outage, leaving the question of testing of containment airlocks as the only remaining request for exemption from the requirements of Appendix J.

The purpose of this report is to conduct technical evaluations of outstanding issues regarding the implementation of 10CFR50, Appendix J, at Turkey Point Units 3 and 4. Consequently, technical evaluations are provided for FPL's request for exemption from the requirements of Appendix J regarding the testing of containment airlocks as submitted in References 2 and 4, as well as a proposed revision to Technical Specification 4.4.2 submitted in Reference 4.

2. EVALUATION CRITERIA

Code of Federal Regulations, Title 10, Part 50 (10CFR50), Appendix J, Containment Leakage Testing, provided the criteria used in conducting the technical evaluations. Where applied to the following evaluations, the criteria are either referenced or briefly stated, where necessary, in support of the results of the evaluations. Furthermore, in recognition of the plant-specific conditions that could lead to requests for exemption not explicitly covered by the regulations, the NRC directed that the technical reviews constantly emphasize the basic intent of Appendix J, that potential containment atmospheric leakage paths be identified, monitored, and maintained below established limits.

3. TECHNICAL EVALUATION

3.1 REQUESTS FOR EXEMPTION FROM THE REQUIREMENTS OF 10CFR50, APPENDIX J

3.1.1 Testing of Containment Airlocks

In Reference 2, FPL requested an exemption from the requirements of 10CFR50, Appendix J with regard to the testing frequency and method of testing containment airlocks. This exemption request would permit continued testing in accordance with Turkey Point Technical Specification 4.4.2.2, which required pressure testing of the personnel and emergency airlocks either annually, if not used, or every 4 months if used periodically. FPL's basis for this request was given as follows:

> Personnel and emergency airlocks are leak tested in accordance with Turkey Point Operating Procedure 13514.1. Leak tightness of the inner door is tested by pressurizing the annulus between the two O-rings. The outer door O-rings are then tested by pressurizing the entire airlock. However, since the inner door opens into containment, both tests tend to unseat the inner door.
>
> Therefore, if the inner door O-rings are to be meaningfully tested, the door must be held shut by a clamping arrangement which takes a minimum of about 12 man-hours to install. A similar arrangement is not required on the outer door because that door opens into the airlock and the test differential pressure is in the direction which seats the door. Thus, a simple positive-pressure test of the personnel and emergency airlocks is not possible because of the design and arrangement of the doors.
>
> Both containments are entered approximately once each week for performance of routine inspections and minor maintenance. If we were to perform the inspection program required by Operating Procedure 13514.1 after each airlock opening, routine entry of the containment would become impractical due to the many man-hours which would be necessary for leak testing. Therefore, in order to continue a viable containment inspection program, and at the same time achieve compliance with the intent of Appendix J, we submitted a proposed Technical Specification change on September 20, 1974, which provided for the performance of an O-ring vacuum test instead of a pressure test. We have designed and built a vacuum test device which could be duplicated and permanently installed on all airlock outer doors and used to

leak test the doors after each opening. Pending disposition of the proposed change, however, we are currently complying with the existing Technical Specification 4.4.2.2 requirement which requires airlock testing once every 4 months.

In Reference 4, FPL withdrew its request for exemption with regard to the frequency of testing airlocks, but continued to request an exemption in order to use the vacuum testing technique to verify airlock door seals after each opening. In this letter, FPL provided a revised Technical Specification 4.4.2.2 which required airlocks to be tested as follows:

4.4.2 LOCAL PENETRATION TESTS

Test Procedure and Frequency

Local leak detection tests of the following components shall be performed at a pressure not less than 50 psig using pressure decay, soap bubble, halogen detection or equivalent methods at the frequency listed, unless otherwise noted:

2. Personnel and Emergency Airlocks

a. Within 3 days of every first of a series of openings when containment integrity is required, verify that door seals have not been damaged or seated improperly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. At least once per 6 months, conduct an overall airlock leakage test to verify that the overall airlock leakage rate is within its limit.

FRC Evaluation:

Sections III.B.2 and III.D.2 of Appendix J require that containment airlocks be tested at peak calculated accident pressure (Pa) at 6-month intervals and after each opening in the interim between 6-month tests. These requirements were imposed because airlocks represent potentially large leakage paths which are more subject to human error than other containment penetrations. Type B penetrations (other than airlocks) require testing in accordance with Appendix J at intervals not to exceed 2 years.

Appendix J was published in 1973. A compilation of airlock events from Licensee Event Reports submitted since 1969 shows that airlock testing in accordance with Appendix J has been effective in the prompt identification of airlock leakage, but that rigid adherence to the after-each-opening requirement may not be necessary.

Since 1969, there have been approximately 70 reported airlock leakage tests in which measured leakage exceeded allowable limits. Of these events, 25% were the result of leakage other than from improper seating of airlock door seals. These failures were generally caused by leakage past door-operating mechanism handwheel packing, door-operating cylinder shaft seals, equalizer valves, or test lines. These penetrations resemble other Type B or C containment penetrations except that they may be operated more frequently. Since airlocks are tested at a pressure of Pa every 6 months, these penetrations are tested, at a minimum, four times more frequently than typical Type B or C penetrations. The 6-month test is, therefore, considered to be both justified and adequate for the prompt identification of this leakage.

Improper seating of the airlock door seals, however, is not only the most frequent cause of airlock failures (the remaining 75%), but also represents a potentially large leakage path. While testing at a pressure of Pa after each opening will identify seal leakage, it can also be identified by alternative methods such as pressurizing between double-gasketed door seals (for airlocks designed with this type of seal) or pressurizing the airlock to pressures other than Pa. Furthermore, experience gained in testing airlocks since the issuance of Appendix J indicates that the use of one of these alternative methods may be preferable to the full-pressure test of the entire airlock.

Reactor plants designed prior to the issuance of Appendix J often do not have the capability to test airlocks at Pa without the installation of strongbacks or the performance of mechanical adjustments to the operating mechanism of the inner doors. The reason for this is that the inner doors are designed to seat with accident pressure on the containment side of the door, and therefore, the operating mechanisms were not designed to withstand accident pressure in the opposite direction. When the airlock is pressurized for a local airlock test (i.e., pressurized between the doors), pressure is exerted on the airlock side of the inner door, causing the door to unseat and preventing the performance of a meaningful test. The strongback or mechanical adjustments prevent the unseating of the inner door, allowing the test to proceed. The installation of strongbacks or performance of mechanical adjustments is time consuming (often taking several hours), may result in additional radiation exposure to operating personnel, and may also cause degradation of the operating mechanism of the inner door, with consequential loss of reliability of the airlock. In addition, when conditions require frequent openings over a short period of time, testing at Pa after each opening becomes both impractical (tests often take from 8 hours to several days) and accelerates the rate of exposure of personnel and the degradation of mechanical equipment.

For these reasons, the intent of Appendix J is satisfied, and the undesirable effects of testing after each opening are reduced if a satisfactory test of the airlock door seals is performed within 3 days of each opening or every 3 days during periods of frequent openings, whenever containment integrity is required. The test of the airlock door seals may be performed by pressurizing the space between the double-gasketed seals (if so equipped) or by pressurizing the entire airlock to a pressure less than Pa that does not require the installation of strongbacks or performance of other mechanical adjustments. If the reduced pressure airlock test is to be employed, the results of the leakage test must be conservatively extrapolated to equivalent Pa test results.

In view of the foregoing discussion, FPL's proposed Technical Specification 4.4.2.2 is acceptable. Furthermore, no exemption from the requirements of Appendix J is necessary because FPL's proposed testing is within the revised version of Section III.D.2 (effective October 22, 1980). FPL should ensure that its airlock testing program is in complete conformance with the revised rule.

With regard to the extrapolation of the reduced pressure test to equivalent Pa test results, comments on FPL's proposed extrapolation method submitted on November 26, 1980 [5] are contained in Appendix A to this report.

3.2 PROPOSED TECHNICAL SPECIFICATION CHANGE

In Reference 4, FPL proposed to revise Specification 4.4.2 to incorporate its proposed exemption from Appendix J with regard to the testing of containment airlocks. In addition, this specification provided for testing at Pa using pressure decay, soap bubble, halogen detection, or equivalent methods of the following components:

Containment purge valves - each refueling
Equipment access openings - annually and after use
Fuel transfer tube flange - each refueling
Electrical penetrations - each refueling.

The proposed specification also required that repairs and tests be made whenever the sum of the local leak rate tests, including isolation valves, exceeds 60% of the total containment allowable leak rate.

FRC Evaluation:

In Section 3.1 of this report, FRC found FPL's proposal for testing of containment airlocks to be acceptable, provided that the results of the vacuum testing between airlock door seals are conservatively extrapolated to Pa results. The remainder of the proposed specification conforms to Section III.B of Appendix J. Consequently, Proposed Specification 4.4.2 is acceptable in meeting the requirements and intent of Appendix J.

4. CONCLUSIONS

FPL's request for exemption from the requirements of Appendix J regarding testing of containment airlocks as submitted in References 2 and 4 and FPL's proposed change to Technical Specification 4.4.2 as submitted in Reference 4 were technically evaluated. The conclusions of these evaluations are as follows:

- FPL's proposal to verify that airlock door seals have not been damaged or seated improperly by vacuum testing the volume between the seals within 3 days of every first of a series of openings, when containment integrity is required in the interim between full-pressure 6-month, tests is acceptable. No exemption is required because of the revision to Section III.D.2, effective October 22, 1980.

- FPL's proposed change to Technical Specification 4.4.2 is acceptable since it conforms to the requirements of Appendix J.

5. REFERENCES

1. K. R. Goller (NRC)
Letter to FPL
August 7, 1975.

2. R. E. Uhrig (FPL)
Letter to K. R. Goller (NRC)
September 12, 1975.

3. R. E. Uhrig (FPL)
Letter to E. G. Case (NRC)
September 20, 1974.

4. R. E. Uhrig (FPL)
Letter to V. Stello (NRC)
July 27, 1977.

5. R. E. Uhrig (FPL)
Letter to S. A. Varga (NRC)
November 26, 1980.

APPENDIX A – EXTRAPOLATION OF REDUCED PRESSURE LEAKAGE MEASUREMENTS TO EQUIVALENT FULL PRESSURE LEAKAGE

1. FPL's CORRELATION

In Reference 5, FPL provided the following information:

"The test will begin at a absolute pressure of 12.92" Hg (17" Hg vacuum) and alarm if pressure increases to 14.42" Hg (15.5" Hg vacuum).

To determine the leak rate at 50 psig (64.7 psia), the design basis accident pressure, the following derivation was used:

Flow for a compressible fluid may be calculated as follows:

$$F = K\,Y\sqrt{\Delta P} \qquad (1)$$

where F = Flow or leakage
K = Coefficient of resistance
Y = Expansion factor
ΔP = Pressure drop across seal

The maximum valve for Y is 1.0 and calculates the leakage for a non-compressible fluid. The coefficient of resistance is constant for each seal tested. Therefore:

$$F = K\sqrt{\Delta P} \qquad \text{or} \qquad L = K\sqrt{\Delta P}$$

A ratio between the leak rate at L_{50} and L_{test} becomes:

$$\frac{L_{50}}{L_{test}} = \frac{K\sqrt{P_{64.7} - P_{14.7}}}{K\sqrt{P_{14.7} - P_{test}}}$$

$$L_{50} = L_{test}\left[\frac{\sqrt{P_{64.7} - P_{14.7}}}{\sqrt{P_{14.7} - P_{test}}}\right]$$

where $P_{64.7}$ = 131.73" Hg

$P_{14.7}$ = 29.92" Hg

$$L_{50} = L_{test}\left[\frac{\sqrt{131.73 - 29.92}}{\sqrt{29.92 - P_{test}}}\right]$$

[1]Chemical Engineer's Handbook, McGraw-Hill, Inc., 1963, Section 5 (Fluid Mechanics, Flow Measurement), Pages 5-8 & 5-9

$$L_{50} = \sqrt{102.52}\left[\frac{L_{test}}{29.92 - P_{test}}\right]$$

$$\therefore \quad L_{50} = 10.13\left[\frac{L_{test}}{29.92 - P_{test}}\right] \text{ with pressures in inches of Hg"}$$

By substituting 12.92 in. Hg for P_{test}, this formula yields the following correlation:

$$\frac{L_{50}}{L_{test}} = 2.45$$

EVALUATION:

The Licensee dropped the value of Y from the formula $F = KY\sqrt{\Delta P}$ because the maximum value of Y is 1.0. If the value of Y is retained, the correlation would be:

$$\frac{L_{50}}{L_{test}} = \frac{KY_{50}\sqrt{P_{64.7} - P_{14.7}}}{KY_{test}\sqrt{P_{14.7} - P_{test}}}$$

Although the maximum value of Y is 1.0, it does not follow that the ratio of Y_{50} to Y_{test} is necessarily ≤ 1.0. Consequently, the Licensee's correlation is not necessarily conservative.

2. VISCOUS FLOW

For viscous flow, mass flow rate ($\dot{m}$) is proportional to the difference of the square of inlet pressure and the square of outlet pressure:

$$\dot{m}_{50} = (64.7^2 - 14.7^2) \times \text{const.}$$

$$\dot{m}_{test} = (14.7^2 - P_{test}{}^2) \times \text{const.}$$

$$L_{50} \sim F_{50} = \frac{\dot{m}_{50}}{P_{50}} = \frac{64.7^2 - 14.7^2}{64.7} \times \text{const.}$$

$$L_{test} \sim F_{test} = \frac{\dot{m}_{test}}{P_{test}} = \frac{14.7^2 - P_{test}^2}{14.7} \times \text{const}$$

$$\therefore \frac{L_{50}}{L_{test}} = \frac{64.7^2 - 14.7^2}{64.7} \times \frac{14.7}{14.7^2 - P_{test}^2}$$

Using in. Hg units: 64.7 = 131.73

14.7 = 29.92

P_{test} = 12.92

then $\frac{L_{50}}{L_{test}} = 5.13$

3. CHOKED FLOW

For choked flow, $P_{orifice} \simeq P_{source}$. Therefore, apart from Reynold's Number effects, mass flow rate ~ $P_{source,\ abs}$.

F = volumetric flow rate (at source density) is independent of P_{outlet}. Therefore, since volumetric flow rate is proportional to the percent of mass per unit time (denoted by L),

$$\frac{L_{50}}{L_{test}} = \frac{P_{64.7}}{P_{test}} = \frac{131.73 \text{ in. Hg}}{29.92 \text{ in. Hg}} = 10.2$$

CONCLUSION:

The above analysis yields the following results for the correlation of L_{50}/L_{test}:

FPL's $\sqrt{\Delta P}$ Method	Viscous Flow	Choked Flow
2.45	5.13	10.2

Since the choked flow correlation is the most conservative, this correlation should be used.

It should be noted that FPL stated in Reference 5 that the allowable local leakage rate at Turkey Point is 0.25% wt/day or 45,000 cc/min. At the same time, FPL calculated airlock leakage rates (including instrument errors) which will cause an alarm to sound after an elapsed time of 1 minute as follows:

$$L_{test} = 31.93 \text{ cc/min (personnel airlock)}$$
$$L_{test} = 7.98 \text{ cc/min (emergency airlock).}$$

It can be seen that even using the most conservative correlation (choked flow), the alarm will detect leakage which is a very small percentage of total allowable local leakage (less than 1%).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 73 to Facility Operating License No. DPR-31, and Amendment No. 67 to Facility Operating License No. DPR-41 issued to Florida Power and Light Company (the licensee), which revised Technical Specifications for operation of Turkey Point Plant, Unit Nos. 3 and 4 (the facilities) located in Dade County, Florida. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to: (1) include the air lock testing according to Appendix J to 10 CFR Part 50; (2) make certain corrections in terminology to be consistent with Appendix J; and (3) make certain administrative corrections.

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 these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated September 20, 1974, as supplemented July 27, 1977, (2) Amendment Nos. 73 and 67 to License Nos. DPR-31 and DPR-41, 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, NW., Washington, D. C. and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of November 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Robert E. Uhrig
Florida Power and Light Company

NOV 4 1981

cc: Mr. Robert Lowenstein, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, N.W.
Suite 1214
Washington, D. C. 20036

Environmental and Urban Affairs Library
Florida International University
Miami, Florida 33199

Mr. Norman A. Coll, Esquire
Steel, Hector and Davis
1400 Southeast First National
Bank Building
Miami, Florida 33131

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

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

Bureau of Intergovernmental Relations
560 Apalachee Parkway
Tallahassee, Florida 32304

Resident Inspector
Turkey Point Nuclear Generating Station
U. S. Nuclear Regulatory Commission
Post Office Box 1207
Homestead, Florida 33030

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.W.
Atlanta, Georgia 30308

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

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