Docket No. 50-~~398~~ 389

December 16, 1988

Mr. W. F. Conway
Senior Vice President-Nuclear
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Conway:

SUBJECT: ST. LUCIE UNIT 2 - ISSUANCE OF AMENDMENT RE: CONTAINMENT LEAKAGE RATE TEST PRESSURE CHANGE (TAC NO. 69324)

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application dated September 7, 1988.

This amendment changes the value of Pa in Technical Specifications Section 3/4.6.1, entitled "Containment Systems." Pa is defined in Appendix J (Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors) to 10 CFR Part 50 as the calculated peak containment pressure related to the design basis accident. The value of Pa is being changed from 43.4 psig to 41.8 psig. The value of 41.8 psig represents the postulated Loss of Coolant Accident peak containment internal pressure.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY Herbert N. Berkow FOR

E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 36 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

[Amend Tac No. 69324]

*LA:PDII-2	*PM:PDII-2	D:PDII-2	*DEST:
DMiller	ETourigny/jd	HBerkow	JCraig
11/03/88	11/03/88	12/16/88	11/17/88
*RPEP	*OGC		
LCunningham	SHLewis		
12/02/88	12/12/88		

*SEE PREVIOUS CONCURRENCE

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PDR ADOCK 05000389
P PDC

cp/cc

Mr. W. F. Conway
Florida Power & Light Company

St. Lucie Plant

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DATED: December 16, 1988

AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

Docket File
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PDII-2 Reading
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G. Lainas, 14/H/3
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D. Miller
E. Tourigny
OGC-WF
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Wanda Jones, P-130A
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ACRS (10)
GPA/PA
ARM/LFMB
PD Plant-specific file [Gray File]
B. Wilson, R-II

cc: Plant Service list

DF01
1/1

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.36
License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated September 7, 1988, 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, Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.2 to read as follows:

    2. Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 36 , 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

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 16, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 36

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-9	3/4 6-9
3/4 6-10	3/4 6-10

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1*, 2*, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.4.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a, 41.8 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 41.8 psig, or

2. Less than or equal to L_t, 0.35 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 20.9 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

c. A combined bypass leakage rate of less of than or equal 0.12 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.12 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 L_a, and the bypass leakage rate to less than or equal to 0.12 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50: using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during

SURVEILLANCE REQUIREMENTS (Continued)

shutdown at either P_a, 41.8 psig or at P_t, 20.9 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

   2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage rate at P_a, 41.8 psig or P_t, 20.9 psig.

d. Type B and C tests shall be conducted with gas at P_a, 41.8 psig at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.7.3 and 4.6.1.7.4.

SURVEILLANCE REQUIREMENTS (Continued)

f. The combined bypass leakage rate shall be determined to be less than or equal to 0.12 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 41.8 psig during each Type A test.

g. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 41.8 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable*:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying the seal leakage is < 0.01 L_a as determined by precision flow measurement when the volume between the door seals is pressurized to greater than or equal to:

   1. For the personnel air lock, greater than or equal to P_a, 41.8 psig for at least 15 minutes if not tested with the automatic tester.

   2. For the emergency air lock, greater than or equal to 41.8 psig for at least 15 minutes.

b. By conducting overall air lock leakage tests at not less than P_a, 41.8 psig, and verifying the overall air lock leakage rate is within its limit:

   1. At least once per 6 months,# and

   2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR 50.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 36

TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

INTRODUCTION

By application dated September 7, 1988, the Florida Power and Light Company, the licensee, requested a change to the value of Pa in Technical Specifications (TS) 3/4.6.1.1 entitled "Primary Containment Integrity," 3/4.6.1.2 entitled "Containment Leakage," and 3/4.6.1.3 entitled "Containment Air Locks." Pa is defined in Appendix J (Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors) to 10 CFR Part 50 as the calculated peak containment internal pressure related to the design basis accident. The current value of Pa is 43.4 psig and represents the Main Steam Line Break Inside Containment (MSLBIC) peak containment internal pressure. The proposed value of 41.8 psig represents the Loss of Coolant Accident (LOCA) peak containment internal pressure. Pa is used to measure and calculate containment leakage rates in order to assure that radiological consequences as a result of an accident will not exceed the guidelines specified in 10 CFR Part 100.

The licensee's containment leakage rate TS (3/4.6.1.2) also permits reduced pressure testing. In this case, the allowable minimum pressure is one half of Pa. Thus, the licensee is also proposing a test pressure of 20.9 psig instead of 21.7 psig when a reduced pressure test is conducted. Again, the reduced pressure testing is using the LOCA peak pressure as its basis, instead of the MSLBIC peak pressure.

EVALUATION

The containment structure at the St. Lucie Plant, Unit 2 is a steel containment vessel surrounded by a reinforced concrete shield building. The two structures are separated by an annular air space. The containment vessel is a low leakage cylindrical steel shell with hemispherical dome and ellipsoidal bottom.

The shield building is a concrete structure which protects the containment vessel from external missiles, provides biological shielding, and provides a means of controlling radioactive fission products that could leak from the containment vessel if an accident would occur.

The containment vessel is designed to withstand the pressure and temperature transients calculated to exist after a design basis accident. Post-accident conditions are determined by evaluating the combined influence of the energy sources, heat sinks, and engineered safety features operation. The design basis accidents for which the containment vessel is designed are the large break LOCA and the MSLBIC accident. The containment vessel design pressure is 44 psig and the design leak rate is 0.50 percent by weight of the containment air per day for the first 24 hours and 0.25 percent per day after 24 hours.

Since the containment vessel is not 100% leak tight, some radioactive nuclides will escape the containment vessel under design basis accident conditions. As such, containment vessel leakage is a significant factor to take into account and the radiological consequences of the design basis accidents must be within the guidelines of 10 CFR Part 100.

Licensees are required to follow 10 CFR 50.54(o) which states that primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to this part. Appendix J addresses primary reactor containment leakage testing for water-cooled power reactors. In order to measure and calculate the containment leakage rate, the containment must be pressurized. The pressure must be indicative of the pressure that would be expected to occur under design basis accident conditions. Thus, the term Pa is used in Appendix J and is defined as the calculated peak containment internal pressure related to the design basis accident and specified either in the Technical Specifications or associated Bases. In the case of the St. Lucie Plant, Unit 2, it is specified in the Technical Specifications.

The design basis accident as far as Appendix J is concerned has traditionally been the large break LOCA, and the Pa associated with this accident has traditionally been used at other pressurized water reactors, including the St. Lucie Plant, Unit 1. The value of Pa for the large break LOCA at St. Lucie, Unit 2 is 41.8 psig. However, the current value of Pa in the St. Lucie, Unit 2 Technical Specifications is 43.4 psig, which reflects the MSLBIC accident. This value was placed in the Unit 2 Technical Specifications when the unit was licensed in 1983. Thus, the licensee's proposal to use a Pa associated with the large break LOCA is acceptable because the large break LOCA is consistent with practice at other pressurized water reactors, including St. Lucie, Unit 1.

The proposed change to Pa will not change the accident analysis and resultant radiological consequences for the postulated LOCA and MSLBIC accidents. In the case of a LOCA, the radiological consequences are within the guidelines of 10 CFR Part 100 as illustrated in Table 15.6.6-12 (Radiological Consequences of a Major Loss of Coolant Accident) of the licensee's Updated Final Safety Analysis Report and in Table 15.3 of the St. Lucie Unit 2 Safety Evaluation Report (NUREG-0893) dated October 1981. The significant containment parameter for this analysis is the containment maximum allowable leakage rate (a Technical Specification value equal to the containment design leakage rate) and this will not change. Implicit with the containment leakage rate is the associated peak containment pressure-associated with the LOCA. The use of the LOCA peak pressure for Pa will ensure that the leakage rate is measured and calculated appropriately.

In the case of an MSLBIC, the radiological consequences are also well within the guidelines of 10 CFR Part 100, as illustrated in Table 15.0-4a (Summary of Chapter 15 Results) of the licensee's Updated Final Safety Analysis Report. The significant containment parameter for this analysis is the maximum allowable containment leakage rate (a Technical Specification value equal to the containment design leakage rate) and this will not change. Implicit with containment leakage rate is the associated MSLBIC peak containment pressure. Although the licensee will now use the LOCA peak pressure for Pa instead of the MSLBIC peak pressure, this will not affect the radiological consequences which are much smaller for the MSLBIC case versus the LOCA case. Thus, the licensee's proposal to use a Pa associated with the large break LOCA is acceptable because it represents the most appropriate pressure to use and the large break LOCA has the most severe radiological consequences.

SUMMARY

Based upon the above evaluation, the staff agrees with the licensee that the value of Pa should be the postulated LOCA peak containment internal pressure and not the postulated MSBLIC peak containment internal pressure. Thus, the Technical Specification changes proposed by the licensee are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change in a surveillance requirement. The staff has determined that the amendment involves 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets 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 the amendment.

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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 16, 1988

Principal Contributor:
E. Tourigny