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

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

December 16, 1988

MEMORANDUM FOR: Sholly Coordinator

FROM: E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II

SUBJECT: REQUEST FOR PUBLICATION IN BIWEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE (TAC NO. 69324)

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: September 7, 1988

Brief description of amendment: The amendment changed the value of Pa in Technical Specification 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 was 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.

Date of Issuance: December 16, 1988

Effective Date: December 16, 1988

Amendment No.: 36

Facility Operating License No. NPF-16: Amendment revised the Technical Specifications.

Date of initial notice in FEDERAL REGISTER: October 19, 1988 (53 FR 40986)
The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 1988

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No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virgina Avenue, Ft. Pierce, Florida.

ORIGINAL SIGNED BY Herbert N. Berkow FOR
E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II

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