UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated June 15, 1999, 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 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 91 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Sheri R. Peterson

Sheri R. Peterson, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 17, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 91

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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 6-5	3/4 6-5
3/4 6-8	3/4 6-8
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
-----	6-19c

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 closed positions, except as provided in Table 3.6-1 of Specification 3.6.3;

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2a. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

* 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.

\# Valves CP-B3, CP-B7, and CM-B5 may be verified at least once per 31 days by manual remote keylock switch position.

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 within limits specified in the Containment Leakage Rate Testing Program.

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.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The Type A containment leakage rate tests shall be performed in accordance with the Containment Leakage Rate Testing Program described in Technical Specification 6.8.4.k. The Type B and Type C containment leakage rate tests shall be demonstrated at the test schedule and shall be determined in conformance with the criteria specified in 10 CFR 50 Appendix J, Option A.

SURVEILLANCE REQUIREMENTS (Continued)

a. Type B and C tests shall be conducted with gas at a pressure not less than P_a, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Containment purge makeup and exhaust isolation valves with resilient material seals;

b. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

c. Purge makeup and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2;

d. The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE by:

a. Performing required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions###. The acceptance criteria for air lock testing are:

   1. Overall air lock leakage rate is $\leq .05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq .01\ L_a$ when tested at $\geq P_a$.

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

1. An inoperable air lock door does not invalidate the previous successful performance of the overall airlock leakage test.
2. Results shall be evaluated against Specification 3.6.1.2.a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

** Only required to be performed upon entry or exit through the containment air lock. (If Surveillance Requirement 4.6.1.3.b has not been performed in the last 6 months, then perform Surveillance Requirement 4.6.1.3.b during the next containment entry through the associated air lock.)

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined, during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2), by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation. Additional inspections shall be conducted during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years.

4.6.1.6.2 Reports. Any abnormal degradation of the containment vessel structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a, during performance of the periodic test, to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50, Option A for Type B and C tests, and the Containment Leakage Rate Testing Program for Type A tests.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

Action statement "a" has been modified by a note. The note allows use of the air lock for entry and exit for seven days under administrative controls if both air locks have an inoperable door. This seven day restriction begins when a door in the second air lock is discovered to be inoperable. Containment entry may be required to perform Technical Specification surveillances and actions, as well as other activities on equipment inside containment that are required by Technical Specifications (TS) or other activities that support TS required equipment. In addition, containment entry may be required to perform repairs on vital plant equipment, which if not repaired, could lead to a plant transient or a reactor trip. This note is not intended to preclude performing other activities (i.e., non-TS required activities or repairs on non-vital plant equipment) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize containment during the short time that an OPERABLE door is expected to be open.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of a postulated main steam line break accident (41.2 psig). A visual inspection in conjunction with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment preentry purge makeup and exhaust isolation valves are required to be sealed closed during plant operations in MODES 1, 2, 3 and 4 since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during these MODES ensures that excessive quantities of radioactive materials will not be released via the Pre-entry Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Normal Containment Purge System is restricted to the 8-inch purge makeup and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during normal containment PURGING operation. The total time the Normal Containment Purge System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4.

Leakage integrity tests with a maximum allowable leakage rate for containment purge makeup and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before

PROCEDURES AND PROGRAMS (Continued)

k. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in conformance with the NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with the following exception noted:

1) The above Containment Leakage Rate Testing Program is only applicable to Type A testing. Type B and C testing shall continue to be conducted in accordance with the original commitment to 10 CFR 50 Appendix J, Option A.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident is 38.4 psig. The calculated peak containment internal pressure related to the design basis main steam line break is 41.2 psig. P_a will conservatively be assumed to be 41.2 psig for the purpose of containment testing in accordance with this Technical Specification.

The maximum allowable containment leakage rate, L_a at P_a, shall be 0.1 % of containment air weight per day.

The containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, test frequencies specified in this Program may be extended consistent with the guidance provided in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has this provision for test frequency extension:

1) Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

The provisions of Surveillance Requirement 4.0.3 are applicable to the Containment Leakage Rate Testing Program.