December 19, 198

Docket Nos. 50-335
and 50-389

DISTRIBUTION
See attached page

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 UNITS 1 AND 2 - CHANGES TO THE TECHNICAL SPECIFICATIONS - USE OF "MASS POINT" METHOD FOR LEAK RATE TESTING (TAC NOS. 68479 AND 68480)

The Commission has issued the enclosed Amendment Nos. 99 and 37 to Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated June 23, 1988.

These amendments change Technical Specification 4.6.1.2, "Containment Leakage" by deleting the reference to ANSI Standard N45.4-1972. The above change for each unit allows the use of the "mass point" method, which is described in ANSI/ANS Standard 56.8-1981 (revised 1987), for determination of containment leakage rates.

You also requested an exemption from certain requirements of Appendix J to 10 CFR Part 50, in conjunction with the Technical Specification change. However, on November 15, 1988, a Final Rule was published in the Federal Register which amended 10 CFR Part 50, Appendix J, to explicitly permit the use of the "mass point" method. Therefore, an exemption is no longer necessary.

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 H.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. 99 to DPR-67
2. Amendment No. 37 to NPF-16
3. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000335
P PDC

[AMEND 68479/80]

DF01
1/1

*See previous concurrence

LA:PDII-2*	PM:PDII-2*	D:PDII-2*	PSB*	OGC-WF*
DMiller	ETourigny:bd	HBerkow	JCraig	SHLewis
08/31/88	11/28/88	11/15/88	09/07/88	09/15/88

ML013550

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

St. Lucie Plant

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

DATED: December 19, 1988

AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1
AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
E. Tourigny
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
T. Meek(8), P1-137
Wanda Jones, P-130A
E. Butcher, 11/F/23
ACRS (10)
GPA/PA
ARM/LFMB
B. Wilson, RII

cc: Plant Service list

DFol
1|1

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. DPR-67

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 June 23, 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. DPR-67 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. 99 , 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 19, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 99

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 6-2	3/4 6-2
B 3/4 6-1	B 3/4 6-1

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT VESSEL

CONTAINMENT VESSEL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without CONTAINMENT VESSEL INTEGRITY, restore CONTAINMENT VESSEL INTEGRITY within one 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 CONTAINMENT VESSEL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

   1. All containment vessel 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 3.6.3.1, and

   2. All containment vessel equipment hatches are closed and sealed.

b. By verifying that each containment vessel air lock is OPERABLE per Specification 3.6.1.3.

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. $\leq L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a, (39.6 psig), or

2. $\leq L_t$, 0.32 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (19.8 psig).

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests as identified in Table 3.6-1 when pressurized to P_a.

c. A combined leakage rate of $\leq 0.27\ L_a$ 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 $0.27\ L_a$, restore the leakage rate(s) to within the limit(s) 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:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a (39.6 psig) or at P_t (19.8 psig) during each 10-year

3/4.6.1 CONTAINMENT VESSEL

3/4.6.1.1 CONTAINMENT VESSEL INTEGRITY

CONTAINMENT VESSEL 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 accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits 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 accident analyses at the peak accident pressure, P_a (39.6 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq$ 0.75 L_a or $\leq$ 0.75 L_t (as applicable) during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the option of using any NRC-approved method for performing the leak rate testing and calculating the results.

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.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structural is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.70 psi and 2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break accident conditions.

The maximum peak pressure obtained from a steam line break accident is 41.6 psig. The limit of 2.4 psig for initial positive containment pressure will limit the total pressure to 44.0 psig which is the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment vessel temperature does not exceed the design temperature of 264°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 41.6 psig in the event of a steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capbility will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

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. 37
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 June 23, 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. 37 , 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 19, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 37

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 6-2	3/4 6-2
B3/4 6-1	B3/4 6-1

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 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:

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

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 limits of 10 CFR 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 accident 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 or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50 with the option of using any NRC-approved method for performing the leak rate testing and calculating the results.

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.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.7 psi and (2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 43.4 psig. The limit of 0.4 psig for initial positive containment pressure will limit the total pressure to 43.99 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment temperature does not exceed the design temperature of 264°F during steam line break conditions and is consistent with the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 44.0 psig in the event of a steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 48-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes devices to lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 99 AND 37

TO FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

INTRODUCTION

By application for license amendment dated June 23, 1988, the Florida Power and Light Company (the licensee) requested changes to the St. Lucie Plant, Unit Nos. 1 and 2 Technical Specification (TS) 4.6.1.2, "Containment Leakage" by deleting the reference to ANSI Standard N45.4-1972. The above change to the TS has been proposed by the licensee to allow for use of the "mass point" method, which is described in ANSI/ANS Standard 56.8-1981 (revised 1987), for determination of containment leakage rate.

DISCUSSION AND EVALUATION

The licensee has requested that the reference to ANSI N45.4-1972 in TS 4.6.1.2 be deleted in order to utilize the "mass point" method for integrated containment leak rate testing.

It has been recognized by the professional community that the mass point method is an acceptable means for calculation of containment leakage in addition to the two other methods, point-to-point and total time, which are referenced in ANSI N45.4-1972 and endorsed by the present regulations. The mass point method calculates the air mass at each point in time, and plots it against time. A linear regression line is plotted through the mass-time points using a least square fit. The slope of this line is proportional to the leakage rate. The mass point method has some advantages when it is compared with the other methods. In the total time method, a series of leakage rates is calculated on the basis of air mass differences between an initial data point and each individual data point thereafter. If for any reason (such as instrument error, lack of temperature equilibrium, ingassing, or outgassing) the initial data point is not accurate, the results of the test will be affected. In the point-to-point method, the leak rates are based on the mass difference between each pair of consecutive points which are then averaged to yield a single leakage rate estimate. Mathematically, this can be shown to be the difference between the air mass at the beginning of the test and the air mass at the end of the test expressed as a percentage of the containment air mass.

It follows from the above that the point-to-point method does not make use of any mass readings taken during the test and thus the leakage rate is calculated on the basis of the difference in mass between two measurements taken at the beginning and at the end of the test, which are 24 hours apart.

ANSI/ANS 56.8-1981 (revised 1987), which was intended to replace ANSI N45.4-1972, specifies the use of the mass point method, to the exclusion of the two older methods. However, the staff has determined that these three methods (mass point, total time and point-to-point) are acceptable methods which may be used to calculate containment leakage rates. On February 29, 1988, the NRC staff published for comment a proposed revision to Appendix J that would permit the use of the mass point method (see 53 FR 5985). On November 15, 1988, the Final Rule was published and became effective (53 FR 45890).

Based upon the above, we conclude that the use of the "mass point" methodology is acceptable as is the proposed change to TS 4.6.1.2.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

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 19, 1988

Principal Contributor:

E. Tourigny