May 27, 1997

Mr. James W. Langenbach
Vice President and Director, TMI
GPU Nuclear Corporation
Route 441 South
PO Box 480
Middletown, PA 17057-0480

SUBJECT: THREE MILE ISLAND - ISSUANCE OF AMENDMENT RE: 10 CFR 50, APPENDIX J, OPTION B (TAC NO. M96029)

Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) in response to your letter dated June 28, 1996, as supplemented March 11, 1997.

The amendment revises the TMI-1 Technical Specifications to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing.

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

This completes our effort on this issue and we are, therefore, closing out TAC No. M96029.

Sincerely,
Original signed by
Bart C. Buckley, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 201 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

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

May 27, 1997

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Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) in response to your letter dated June 28, 1996, as supplemented March 11, 1997.

The amendment revises the TMI-1 Technical Specifications to permit the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing.

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Bart C. Buckley

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Docket No. 50-289

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2. Safety Evaluation

cc w/encls: See next page

Three Mile Island Nuclear Station, Unit No. 1

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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 201
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated June 28, 1996, as supplemented March 11, 1997, 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. DPR-50 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 201, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Patrick D. Milano, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 27, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove	Insert
3-41a	3-41a
3-41b	3-41b
3-41c	3-41c
3-41d	3-41d
4-29	4-29
4-30	4-30
4-31	---
4-32	---
4-33	---
---	6-11c

3.6.8 While containment integrity is required (see T.S. 3.6.1), if a 48" reactor building purge valve is found to be inoperable perform either 3.6.8.1 or 3.6.8.2 below.

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage, close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared OPERABLE. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (see Specification 6.8.5), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below:

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of Specification 6.8.5 and perform either (1) or (2) below.

(1) Restore the leaking valve to OPERABILITY within the following 72 hours.

(2) Maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test in accordance with the Reactor Building Leakage Rate Testing Program. In order to accomplish repairs, one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V-1A&D) shall be limited to less than 31° and (AH-V-1B&C) shall be limited to less than 33° open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons:

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

3.6 REACTOR BUILDING (Continued)

(3) Performance of Technical Specification required surveillances.

(4) Radiation Surveys.

(5) Engineering support of safety-related modifications for pre-outage planning.

(6) Purging prior to shutdown to prevent delaying of outage.commencement (24 hours prior to shutdown).

c. Containment purging is permitted for Reactor Building pressure control.

d. To the extent practicable the above containment entries shall be scheduled to coincide, in order to minimize instances of purging.

3.6.11 When the reactor is in COLD SHUTDOWN or REFUELING SHUTDOWN, continuous purging is permitted with the Reactor Building purge isolation valves opened fully.

3.6.12 Personnel or emergency air locks:

a. At least one door in each of the personnel or emergency air locks shall be closed and sealed during personnel passage through these air locks.

b. One door of the personnel or emergency air lock may be open for maintenance, repair or modification provided the other door of the air lock is verified closed within 1 hour, locked within 24 hours, and verified to be locked closed monthly. Air lock doors in high radiation areas may be verified locked closed by administrative means.

c. Entry and exit is permissible to perform repairs on the affected personnel or emergency air lock components. With both air locks inoperable due to inoperability of only one door in each airlock, entry and exit is permissible for 7 days under administrative controls. With the personnel or emergency air lock door interlock mechanism inoperable, entry and exit is permissible under the control of a dedicated individual.

d. With one or more air locks inoperable for reasons other than "b" or "c" above, initiate action immediately to evaluate the overall containment leakage rate with respect to the requirements of Specification 6.8.5, verify a door is closed in the affected air lock within 1 hour, and restore the affected air lock(s) to operable status within 24 hours or the reactor shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

Bases

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30° open position.

Allowing purge operations during STARTUP, HOT STANDBY and POWER OPERATION (T.S. 3.6.10) is more beneficial than requiring a cooldown to COLD SHUTDOWN from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. The recombiner is designed in accordance with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971, the acceptance criteria of the Standard Review Plan (S.R.P.) 6.2.5., and NUREG 0578, July 1979. In addition to the installed hydrogen recombiner, a second recombiner including all piping, electrical, and structural provisions is available on site.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Reference 1), and the Reactor Building Leakage Rate Testing Program. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events.

Entry and exit is allowed to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock. With both air locks inoperable due to inoperability of one door in each of the two air locks, entry and exit is allowed for use of the air locks for 7 days under administrative controls. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was 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 the containment during the short time that the OPERABLE door is expected to be open.

With one or more air locks inoperable for reasons other than those described in 3.6.12."b" or "c," Section 3.6.12.d requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour would otherwise be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Section 3.6.12.d requires that one door in the affected containment air locks(s) must be verified to be closed within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. 24 hours is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

References

(1) 10 CFR 50, Appendix J.

4.4 REACTOR BUILDING

4.4.1 CONTAINMENT LEAKAGE TESTS

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Testing (ILRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program at test frequencies established in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2 Local Leakage Rate Testing (LLRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program. LLRT shall be performed at a pressure not less than peak accident pressure P_{ac} with the exception that the airlock door seal tests shall normally be performed at 10 psig and the periodic containment airlock tests shall be performed at a pressure not less than P_{ac}. LLRT frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Operability of the personnel and emergency air lock door interlocks and the associated control room annunciator circuits shall be determined at least once per six months. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable.

Bases (1)

The Reactor Building is designed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design internal pressure of 55 psig with a coincident temperature of 281°F at accident conditions. The peak calculated Reactor Building pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig. The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}. Containment Isolation Valves are addressed in the UFSAR (Reference 2).

The Reactor Building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program (See Section 6.8.5). This program is contained in the surveillance procedures for Reactor Building inspection, Integrated Leak Rate Testing, and Local Leak Rate Testing. These periodic testing requirements verify that Reactor Building leakage rate does not exceed the assumptions used in the safety analysis. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. 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 leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

Periodic surveillance of the airlock interlock systems (Reference 4) is specified to assure continued operability and preclude instances where one or both doors are inadvertently left open. When an airlock is inoperable and containment integrity is required, local supervision of airlock operation is specified.

Reference

(1) UFSAR, Chapter 5.7.4 - "Post Operational Leakage Rate Tests"
(2) UFSAR, Tables 5.7-1 and 5.7-3
(3) DELETED.
(4) UFSAR, Table 5.7-2

(Pages 4-31 through 4-34, 4-34a, and 4-34b deleted)

6.8.5 Reactor Building Leakage Rate Testing Program

The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:

A program shall be established to implement the leakage rate testing of the Reactor Building 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 accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig.

The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}.

Reactor Building leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests and $\leq$ 0.75 L_a for the Type A tests.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated June 28, 1996, as supplemented March 11, 1997, GPU Nuclear Corporation, the licensee, requested changes to the Technical Specifications (TS) for Three Mile Island, Unit 1. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Reactor Building Leakage Rate Testing Program" and proposed adding this program to the TS. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B.

The March 11, 1997, letter provided clarifying information that did not change the scope of the June 28, 1996, submittal and the proposed no significant hazards consideration.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

RG 1.163, dated September 1995, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Three Mile Island Unit 1 TS.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's June 28, 1996, letter to the NRC proposes to establish a "Reactor Building Leakage Rate Testing Program" and proposes to add this program to the TS . The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires revising existing TS sections 3.6.8.2, 3.6.10, 3.6.12, 4.4.1.1, 4.4.1.1.4, 4.4.1.2, and 4.4.1.2.5, and relocating applicable information from TS sections 4.4.1.1.1, 4.4.1.1.2, 4.4.1.1.3, 4.4.1.1.5, 4.4.1.1.6, 4.4.1.1.7, 4.4.1.2.1, 4.4.1.2.2, 4.4.1.2.3, 4.4.1.2.4, 4.4.1.3, 4.4.1.4, 4.4.1.5, and 4.4.1.7 to a new TS section 6.8.5 which requires a Reactor Building Leakage Rate Testing Program. Section 4.4.1.3 was deleted. This section is redundant to 10 CFR 50.55a(f) and TS 4.2.2. Corresponding Bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

The TS changes proposed by the licensee to adopt 10 CFR 50, Appendix J, Option B are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995 letter, and are, therefore, acceptable to the staff.

The licensee has also proposed to modify the TS for the 48-inch reactor building purge valves by requiring leakage rate testing in accordance with the Reactor Building Leakage Rate Testing Program. This changes the leakage rate test frequency from every 3 months to 30 months.

As a result of reports of unsatisfactory performance of resilient seals in butterfly-type isolation valves, such as containment purge valves, due to seal deterioration, the NRC established Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" to study this issue and propose a regulatory resolution of the problem. IE Circular 77-11 "Leakage of Containment Isolation Valves With Resilient Seals" was issued and the final resolution imposed augmented leakage rate testing requirements which required more frequent testing than Appendix J for containment purge and vent line isolation valves that used resilient seal materials. The TMI TS contain these augmented testing frequency requirements at a frequency of every 3 months.

The licensee was requested to demonstrate good leakage rate performance over an extended period of time for these valves. The licensee responded by letter dated March 11, 1997, providing TMI-1 containment purge valve test results from 1986 to the present. The staff has reviewed these data and concludes that the data justify a change to a leakage rate test frequency of every 30 months.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (61 FR 40019). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 55.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Richard Lobel
Bart Buckley

Date: May 27, 1997