August 31, 1989

Docket No. 50-289

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 73748)

The Commission has issued the enclosed Amendment No. 151 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, in response to your letter dated June 13, 1989.

The amendment revises the Technical Specifications (TS) by removing the listing of penetration components and valves requiring local leak rate tests per 10 CFR 50, Appendix J and providing this listing in the Updated Final Safety Analysis Report (USAR). The amendment also corrects various editorial errors in the TS, deletes periodic monitoring of rotameter readings in the Penetration Pressurization System from the TS and revises the listing of valves requiring Appendix J leak testing.

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

Sincerely,

/s/

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 151 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

[AMEND TAC 73748]

LA:PDI-4	PM:PDI-4	PD:PDI-4	NRR/ECEB	OGC
SNorris	RHernan:lm	JStolz	McCracken	
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DATED: August 31, 1989
AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-50

Docket File
NRC & Local PDRs
Plant File
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J. Stolz
S. Norris
R. Hernan
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meek(4) (P1-137)
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ACRS (10)
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Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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

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. 151
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated June 13, 1989 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 paragraphs 2.c.(2) of Facility Operating License No. DPR-50 are hereby amended to read as follows:

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No.151 , 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

Ronald W. Hernan for

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 31, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 151

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Facility Operating License and the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
4-32	4-32
4-33	4-33
4-34	4-34
4-34a	4-34a
4-34b	4-34b
4-34c	-

detection tests. Sufficient data and analysis shall be included to show that a stabilized leak rate was attained and to identify all significant required correction factors such as those associated with humidity and barometric pressure, and all significant errors such as those associated with instrumentation sensitivities and data scatter. This report shall be titled "Reactor Containment Building Integrated Leak Rate Test" and shall be submitted to the NRC within 3 months of the test.

4.4.1.2 Local Leakage Rate Tests

4.4.1.2.1 Scope of Testing

Local Leakage Rate tests of penetrations and valves identified in the FSAR shall be performed in accordance with 10CFR 50 Appendix J except as provided in 4.4.1.2.5.f.

4.4.1.2.2 Conduct of Tests

a. Local leak rate tests shall be performed pneumatically at a pressure of not less than P_a, with the following exception: The access hatch door seal test shall normally be performed at 10 psig and the test every six months specified in 4.4.1.2.5.b shall be performed at a pressure not less than P_a.

b. Acceptable methods of testing are halogen gas detection, pressure decay, pneumatic flow measurement, or equivalent.

c. The pressure for a valve test shall be applied in the same direction as that when the valve would be required to perform its safety function unless it can be determined that the direction will provide equivalent or more conservative results.

d. Valves to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

4.4.1.2.3 Acceptance Criteria

The combined leakage from all penetrations and valves subject to Local Leak Rate tests shall not exceed .6 L_a (the maximum allowable leakage rate at P_a).

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed at a frequency as required by 10CFR 50 Appendix J, except that:

a. The equipment hatch and fuel transfer tube seals shall be tested every other refueling period but in no case at intervals greater than 3 years. If they are opened they will be tested after being closed.

b. The entire personnel and emergency airlocks shall be tested once every six months. When the airlocks are opened during the interim between six month tests, the airlock door resilient seals shall be tested within 72 hours of the first of each of a series of openings. This requirement exists whenever containment integrity is required.

c. The reactor building purge isolation valves shall be leak tested per 10CFR 50, Appendix J, Item III.D.3.

d. An interspace pressurization test (See T.S. 4.4.1.7.1) shall be performed for reactor building purge isolation valves every 3 months. This requirement is not in effect during cold shutdown.

e. Deleted.

f. Where an exemption from the frequency specified by 10CFR 50 Appendix J has been granted by the NRC, the frequency specified by the exemption shall apply.

4.4.1.3 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The valves not stroked every three months shall be stroked during each refueling period.

4.4.1.4 Annual Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, nondestructive tests, and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the reactor building integrity shall be followed by either an integrated leak rate test or a local leak test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.6 and 4.4.1.2.3, respectively.

4.4.1.6 Operability of Access Hatch Interlocks

1. At least once per six months the operability of the personnel and emergency hatch door interlocks and the associated control room annunciator circuits shall be determined. 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.

2. During periods when containment integrity is required and an interlock is inoperable, each entry and exit via that airlock shall be locally supervised by a member of the unit operating maintenance or technical staffs, to assure that only one door is open at any time and that both doors are properly closed following use. A record of supervision and verification of closure shall be maintained during periods of interlock inoperability in an appropriate station log.

3. If an interlock is inoperable for more than 14 days following determination of inoperability, use of the airlock, except for emergency purposes, shall be suspended until the interlock is returned to operable status.

4.4.1.7 Operability of Purge Valves

1. A periodic pressurization of the purge valve interspaces to 50.6 psig per Specification 4.4.1.2.5.d shall be performed to help assure timely detection and resolution of valve and/or actuator degradation. The acceptance criteria is that total local leakage when updated for the new purge valve leakage shall be less than $0.6L_a$. See Specification 3.6.8 for further action.

2. The rubber seats on purge valves shall be visually examined each refueling interval to detect degradation (e.g. cracking, brittleness, etc.) and to assure timely cleaning, lubrication, and seat replacement. As a minimum, seats shall be replaced at the first refueling following 5 years of seat service.

Bases [1]

The reactor building is designed for an internal pressure of 55 psig and a steam-air mixture temperature of 281°F. Prior to initial operation, the containment was strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment was also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests established the acceptance criteria of 4.4.1.1.3.

The performance of periodic integrated and local leakage rate tests during the plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions "as found" local leakage results must be documented for correction of the integrated leakage rate test results. Containment isolation valves are to be closed in the normal manner prior to local or integrated leakage rate tests. Containment Isolation Valves are addressed in the FSAR.

The minimum test pressure of 30 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it exceeds the pre-operational leakage rate test at the reduced pressure of 27.5 psig. The specification provides a relationship for relating the measured leakage of air at the reduced pressure to the potential leakage of 55 psig. The minimum of 24 hours was specified for the integrated leakage rate test to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.10 percent leakage rate at 55 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation and the low value of leakage that is specified as acceptable from penetrations and isolation valves, 0.6 L_a.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the reactor building liner due to the mechanical closure involved. The basis for specifying a total leakage rate of 0.6 L_a from those penetrations and isolation valves is that more than one-half of the allowable integrated leakage rate will be from these sources.

Valve operability tests are specified to assure proper closure or opening of the reactor building isolation valves to provide for isolation or functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is establish that such testing is not practical during operation. Valves that cannot be full-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

Periodic surveillance of the airlock interlock systems 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.

Purge valve interspace pressurization test operability requirements and inspections provide a high degree of assurance of purge valve performance as containment isolation barriers. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

Reference

(1) FSAR, Chapter 5.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

INTRODUCTION

GPU Nuclear Corporation (GPU) submitted Technical Specification Change Request (TSCR) No. 191 by letter dated June 13, 1989. The amendment requested by this TSCR would remove Section 4.4.1.2.1, Scope of Testing, from the Technical Specifications (TS) and relocate the listing of containment penetration components and valves requiring type "B" and "C" leak rate testing to the Updated Final Safety Analysis Report (USAR). The request also included changes to these lists resulting from modifications to the facility and reevaluation of 10 CFR 50, Appendix J requirements as well as various other editorial changes. Finally, the TSCR requested removal, from the TS, of a periodic surveillance regarding the Penetration Pressurization System.

EVALUATION

The operability of containment isolation valves and leak integrity of containment penetration components ensure that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. To ensure operability and integrity of this hardware, Appendix J requires periodic leak testing of three different types: type A for overall containment integrated leakage, type B for various types of containment penetrations and type C for containment isolation valve leakage. Presently the specific components requiring type B and C leak tests are specifically listed in TS Section 4.4.1.2.5. Each time the content of this list changes due to facility modifications or regulation interpretation a formal amendment to the facility operating license is required. GPU has proposed removal of the list from the TS and incorporating it in the USAR as Table 5.7-2 (type B testing) and Table 5.7-3 (type C testing). Final draft copies of these tables were included with the TSCR.

The Commission's Interim Policy Statement on Technical Specification Improvements recognized the advantages of improved TS and endorsed the recommendations of the nuclear industry and the NRC staff for a program to develop improvements in TS. An important part of that program is the implementation of line-item improvements in TS. This change has been implemented in the TS for new licenses and is consistent with previous guidance provided by Generic Letter 84-13 on removing the list of snubbers from TS. Guidance to licensees regarding removal of this list from TS will be provided to all power reactor licensees in a future generic letter based on issuance of a similar amendment for Crystal River Unit 3 in May 1989. Relocation of the listing of components requiring Appendix J type B and C leak testing would allow future changes to be made without a license amendment. This would relieve both the NRC and the licensee of an administrative burden but would not change any requirements to perform Appendix J leak testing. Maintaining the table in the USAR would also ensure that the information is still available to the operators. Changes to the table would be controlled under 10 CFR 50.59 as a change to the facility. Therefore, adequate measures exist to control changes to the facility without having these components listed in the TS. Due to the proposed relocation of Section 4.4.1.2.5 from the TS to the USAR, references to the section would be deleted from TS 4.4.1.2.3.

The TSCR also discusses three changes to the local leak rate test listing itself. The first is moving containment purge valves AH-VIA/B/C/D from the type B list to the type C list. Type B test requirements apply to containment penetrations whose design incorporates resilient seals, airlock doors, and doors with resilient seals or gaskets. In accordance with 10 CFR 50 Appendix J, the type C test is clearly specified to apply to valves such as AH-VIA/B/C/D that "provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operations, such as purge and ventilation, ... valves." Apparently, because the design of the purge valves incorporates the use of resilient seals, these valves were inappropriately included in the original TMI-1 Technical Specifications as type B components. The purge valves were the only valves included in the list of type B components.

Special tests and inspections of the purge valves would continue to exceed the Appendix J test requirements for containment isolation valve type C tests. Moving the purge valves from the list of valves requiring a type B test to the type C list could result in a change in purge valve test requirements in that it would no longer be required to test the valves prior to startup if opened following a type A or B test if the startup were to occur less than 92 days since the last test. The existing special tests of the purge valves each quarter and examinations each refueling interval provide adequate assurance of timely detection of purge valve seat degradation and inoperability. The staff agrees with this change. A resultant editorial change to TS 4.4.1.2.5.c is required to reflect the appropriate Appendix J section (III.D.3).

The second change to the local leak rate test listing is deletion of valve LR-V10 from the type C list. LR-V10 has been included in the list of valves requiring a type C test ever since the original TMI-1 Technical Specifications was issued even though such a test is not needed in meeting the Local Leakage Rate test requirements of 10 CFR 50 Appendix J. LR-V10 is a manual handwheel operated test connection valve between containment isolation valve LR-V49 and a containment isolation blind flange associated with containment penetration #417. LR-V10 has a blind flange installed at its outlet resulting in double manual isolation of the test connection. Therefore LR-V10 should be deleted from the list of vlaves requiring a type C test. The staff agrees with this change.

The third change to the listing is addition of valves PP-V210, 211, 212 and 213 to the list of type C valves to be tested. During a recent modification, these globe valves were added to the facility to replace four check valves in the Penetration Pressurization System. The staff agrees with this change.

The TSCR also requested deletion of TS 4.4.1.2.5.e which provides for quarterly monitoring of the Penetration Pressurization System. The TSCR stated that this system is not required by Appendix J or any codes and that the TMI-1 safety analysis takes no credit for the active function of this system. The system, in fact, is disabled during the Appendix J type A containment integrated leak rate test. The staff notes, however, that this system interconnects with the instrument air system. The original purpose for the Penetration Pressurization System and related surveillance tests would have been to permit a reduced leak rate test program or justify exemptions from the 10 CFR 50 Appendix J. However, all containment penetrations with resilient seals, process system flanges, valves, and gaskets requiring periodic leak test per Appendix J are now tested using Type B or C tests with no credit or exemptions for the use of a fluid blocking system.

The apparent purpose of TS 4.4.1.2.5.e is to detect abnormal leakage in the Penetration Pressurization System. Since credit is not taken for this system the staff agrees that this surveillance serves no useful purpose. The staff reviewed the possible detrimental effects this system may have on the instrument air system should a major rupture occur in the system, thereby potentially causing depletion of instrument air. We concluded that the possibility of this occurrence is very remote in light of existing low pressure alarms, the ability to isolate the system and recent instrument air system enhancements under the Safety Performance Improvement Program. We therefore conclude that deletion of this TS is appropriate and has no detrimental effect on plant operation.

SUMMARY

The staff finds that the requested changes will maintain conservative limiting conditions for plant operation and adequate surveillance requirements. Thus, the staff finds the proposed changes to be acceptable.

ENVIRONMENTAL CONSIDERATION

The 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 2, and also relates to changes in recordkeeping, reporting or administrative procedures or requirements. We have 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 off site, and that there is no significant increase in individual or cumulative occupational exposure. The staff has previously issued a proposed finding that this 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) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this 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 (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Ronald W. Hernan

Dated: August 31, 1989