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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND 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. 93
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 licensees) dated April 21, 1982, 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 enclosure to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 93 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: May 18, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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 areas of change.

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

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

a. The local leak rate shall be measured for the following components using a type "B" test as defined in 10 CFR 50, Appendix J.

   1. Personnel air lock door gaskets and other seals

   2. Emergency air lock door gaskets and other seals

   3. The resilient seals on the equipment hatch and fuel transfer tube blind flanges
   4. Blind flanges on penetration No. 414 (L.R. Pressure Sensing)
   5. Reactor Building Purge valves (AH-V1A, B, C and D)

   6. Blind flanges on both ends of pipe through the following penetrations:

      6.1 No. 104 (S/G drains)
      6.2 No. 105 (S/G cleaning)
      6.3 No. 106 (S/G cleaning)
      6.4 No. 210 (S/G annulus drains)
      6.5 No. 211 (S/G annulus drains)
      6.6 No. 24[illegible] (Incore Inst. Transfer Tube)
      6.7 No. 415 (L.R. Test Bleed Line)
      6.8 No. 416 (L.R. Test Bleed Line)

b. The local leak rate shall be measured for the following isolation valves using a type "C" test as defined in 10 CFR 50, Appendix J.

   1. CA-V1, 2, 3, 13 (Primary Sampling)
      CA-V189, 192 (Reclaimed Water)
      CA-V4A, 4B, 5A, 5B (Secondary Sampling)

   2. CF-V2A, 2B, 12A, 12B, 19A, 19B, 20A, 20B (Core Flood)

   3. CM-V1, 2, 3, 4 (Containment Monitoring)

   4. DH-V64, 69 (Decay Heat)

   5. HP-V1, 6 (Hydrogen Purge)

   6. HR-2A, 2B, 4A, 4B, 22A, 22B, 23A, 23B (Hydrogen Recombiner)

   7. IA-V6, 20 (Instrument Air)

   8. IC-V2, 3, 4, 6, 16, 18 (Intermediate Cooling)

9. LR-V1, 4, 5, 6, 10, 49 (Leak Rate Test)

10. MU-V2A, 2B, 3, 18, 20, 25, 26, 116 (Make up and Purification)

11. NI-V27 (Nitrogen)

12. NS-V4, 11, 15, 35 (Nuclear Services Closed Cooling)

13. RB-V2A, 7 (R.B. Industrial Cooling System)

14. SA-V2, 3 (Service Air)

15. SF-V23 (Spent Fuel Cooling)

16. WDG-V3, 4 (Waste Gas Header)

17. WDL-V303, 304 (Waste Disposal Liquid)

18. WDL-V534, 535 (R.B. Sump Gravity Drains)

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 items listed in 4.4.1.2.1 shall not exceed .6 L_a (the maximum allowable leakage rate at P_a).

The minimum test pressure of 27.5 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at the reduced pressure. 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 (0.06 percent) of leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

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 0.06 percent 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 established 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.

Reference

(1) FSAR, Section 5.