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. 138
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 January 12, 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, 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.138 , 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, 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: April 25, 1988

ATTACHMENT TO LICENSE AMENDMENT NO.138

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 area of change.

Remove	Insert
4-1	4-1
4-2	4-2
4-10	4-10
5-6	5-6
5-7	5-7

4. SURVEILLANCE STANDARDS

During Reactor Operational Conditions for which a Limiting Condition for Operation does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/component operable, the associated surveillance requirement must be current. The above applicability requirements assure the operability of systems/components for all Reactor Operating Conditions when required by the Limiting Conditions for Operation.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system, engineered safety feature protection system, and heat sink protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Each post accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies shown in Table 4.1-4.

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required monthly on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protection channels is as follows:

Channels A, B, C & D	Before Startup, when shutdown greater than 24 hours
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested monthly. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protection channels and the regulating control rod power SCRs shall be trip tested prior to startup when the reactor has been shutdown for greater than 24 hours.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

TABLE 4.1-3 Cont'd.

	Item	Check	Frequency
4.	Spent Fuel Pool Water Sample	Boron concentration greater than or equal to 600 ppmb	Monthly and after each makeup.
5.	Secondary Coolant System Activity	Isotopic analysis for DOSE EQUIVALENT I-131 concentration	At least once per 72 hours when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F
6.	Boric Acid Mix Tank or Reclaimed Boric Acid Tank	Boron concentration	Twice weekly***
7.	Deleted		
8.	Deleted		
9.	Deleted		
10.	Sodium Hydroxide Tank	Concentration	Quarterly and after each makeup.
11.	Deleted		
12.	Deleted		

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** Deleted

*** The surveillance of either the Boric Acid Mix Tank or the Reclaimed Boric Acid Tank is not necessary when that respective tank is empty.

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 NEW FUEL STORAGE

a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions for the new fuel storage vault and the Spent Fuel Pool "A". The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 13-5/8 inches in both directions for the Spent Fuel Pool "B". This spacing is sufficient to maintain a K effective of less than .95 based on fuel assemblies with an enrichment of 4.3 weight percent U^{235}. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition. When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to ensure meeting the NRC maximum allowable reactivity value under the postulated accident condition of a misplaced fuel assembly.

b. New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in an 8 x 8 array storage rack having a nominal center to center distance of 21-1/8 inches. When fuel is being moved in or over the fuel transfer canal, a boron concentration of at least 600 ppmb must be maintained to ensure that, under the postulated accident condition of a misplaced fuel assembly, the maximum reactivity will be less than the NRC maximum allowable reactivity. This applies only when fuel is being stored in the canal.

c. New fuel may also be stored in shipping containers.

5.4.2 SPENT FUEL STORAGE

a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.

b. Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.

c. Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

d. The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:

	South End of Fuel Transfer Canal RB	Spent Fuel Pool A North End of Fuel Handling Building	Spent Fuel Pool B South End of Fuel Handling Building	Dry New Fuel Storage Area Fuel Handling Building
Fuel Assys	64 *	256 **	496 ***	66****
Cores	0.36	1.45	2.8	0.37

NOTES: * Includes space for accommodating a failed fuel detection container.
** Includes three spaces for accommodating failed fuel containers.
*** Spent Fuel Pool B contains spent fuel storage racks with a reduced center-to-center spacing of 13 5/8 inches to increase the storage capacity of the pool.
**** Includes twelve spaces which are required to be vacant of fissile or moderating material so that there is sufficient neutron leakage.

e. All of the fuel assembly storage racks provided are designed to Seismic Class 1 criteria to the accelerations indicated below:

	Fuel Transfer Canal in Reactor Building	Fuel Handling Building Dry New Fuel Storage Area And Spent Fuel Pool A	Fuel Handling Building Spent Fuel Pool B
Horiz.	0.76 g	0.38 g	*
Vertical	0.51 g	0.25 g	*

* The "B" pool fuel storage racks are designed using the floor response spectra of the Fuel Handling Building.

f. Fuel in the storage pool shall have a U-235 loading equal to or less than 57.8 grams of U-235 per axial centimeter of fuel assembly.

REFERENCES

(1) FSAR, Section 9.7