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. 95
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 24, 1981, 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:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 95 , 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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 25, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 95

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
iv	iv
4-9	4-9
4-10	4-10
4-35	4-35
4-51	4-51
4-53	4-53*
4-54	4-54

*Overleaf page included for document completeness.

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TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

Item		Check	Frequency
1. Reactor Coolant	a.	Radio-Chemical Analysis(1)	Monthly
	b.	$\overline{E}$ determination (2)	Semiannually
	c.	15 Min. Gross Degassed Beta-Gamma Activity (1)	5 times/week when Tavg is greater than 200°F
	d.	Tritium Radioactivity	Monthly
	e.	Chemistry (Cl, F and O_2)	5 times/week when Tavg is greater than 200°F
	f.	Boron Concentration	2 times/week
2. Borated Water Storage Tank Water Sample		Boron Concentration	Weekly and after each makeup when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F
3. Core Flooding Tank Water Sample		Boron Concentration	Monthly and after each makeup when RCS pressure is greater than 700 psig
4. Spent Fuel Pool Water Sample		Boron Concentration	Monthly and after each makeup
5. Secondary Coolant	a.	Gross Activity	Weekly when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F
	b.	Iodine Analysis (3)	
6. Boric Acid Mix Tank or Reclaimed Boric Acid Tank		Boron Concentration	Twice weekly (4)
10. Sodium Hydroxide Tank		Concentration	Quarterly and after each makeup

TABLE 4.1-3 (Continued)

Item	Check	Frequency
11. Deleted		
12. Condenser Partition Factor	I^{131} Partition Factor	Once if primary/secondary leakage develops, i.e.: Gross Beta-Gamma on secondary side of OTSG is greater than 2×10^{-8} micro curies per cc and evidence of fission products is present

(1) When radioactivity level is greater than 10 percent of the limits of Specification 3.1.4, the sampling frequency shall be increased to a minimum of 5 times per week.

(2) $\bar{E}$ determination will be started when the 15 minute gross degassed beta-gamma activity analysis indicates greater than 10 μCi/ml and will be redetermined each 10 μCi/ml increase in the 15 minute gross degassed beta-gamma activity analysis. A radio chemical analysis for this purpose shall consist of a quantitative measurement of 95 percent of radionuclides in reactor coolant with half lives of >30 minutes.

(3) When the gross activity increases by a factor of two above background, an iodine analysis will be made and performed thereafter when the gross activity increases by 10 percent.

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

4.4.2 Structural Integrity

Specification

4.4.2.1 Inservice Tendon Surveillance Requirements

The surveillance program for structural integrity and corrosion protection conforms to the recommendations of the U. S. NRC Regulatory Guide 1.35, proposed Revision 3, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures." The detailed surveillance program for the prestressing system tendons shall be based on periodic inspection and mechanical tests to be performed on selected tendons, as specified hereafter.

4.4.2.1.1 Containment Tendons

Tendon surveillance was completed for one, three and five years following initial structural integrity using a Tech. Spec. based on Regulatory Guide 1.35 Rev. 1. The containment tendon structural integrity shall be demonstrated at five year intervals thereafter by:

a. Determining that for a representative sample* of at least 23 tendons (6 dome, 7 vertical, and 10 hoop) each tendon has a lift off force equalling, or exceeding, its lower limit predicted for the time of the test as defined in NRC Regulatory Guide 1.35, "Inservice Inspection for Ungrouted Tendons in Prestressed Concrete Containments", Proposed Revision 3, April, 1979.

   If the lift off force of a selected tendon in a group lies between the prescribed lower limit and 90% of that limit, one tendon on each side of this tendon shall be checked for their lift off forces. If the lift off forces of the adjacent tendons are equal to, or greater than, their prescribed lower limits at the time of the test, the single deficiency shall be considered unique and acceptable. If the lift off force of either of the adjacent tendons lies below the prescribed lower limit for that tendon, the condition is reportable per T.S. 6.9.2.A3.

   If the lift off force of any one tendon lies below 90% of its prescribed lower limit, the tendon shall be considered a defective tendon. It shall be completely detensioned and a determination made as to the cause of the occurrence. The condition is reportable per T.S. 6.9.2.A3.

   If the inspections performed at one, three, and five years indicate no abnormal degradation of the post-tensioning system, the number of tendons checked for lift off force during subsequent tests may be reduced to a representative sample of at least 11 tendons (3 dome, 3 vertical, and 5 hoop).

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection.

4.8 MAIN STEAM ISOLATION VALVES

Applicability

Applies to the periodic testing of the main steam isolation valves.

Objective

To specify the minimum frequency and type of tests to be applied to the main steam isolation valves.

Specification

4.8.1 A check of valve stem movement, up to 10 percent, shall be performed on a monthly basis when the unit is operational and under normal flow and load conditions.

4.8.2 The main steam isolation valves shall be tested at intervals not to exceed the normal refueling outage. Closure time of < 120 seconds shall be verified. This test will be performed under no flow and no load conditions.

Bases

Since a portion of the main steam lines and the steam lines to the main feed pump turbines are located in the turbine hall which is not protected against hypothetical tornado, missile, or aircraft incident; main steam isolation stop check valves are provided and located in the hardened portion of the intermediate building. These stop check valves are remotely closed by the operator from the control room, close in less than two minutes, and are tight closing [1] for long term containment isolation. Their ability to close upon signal should be verified at intervals not to exceed each scheduled refueling shutdown, and valve stem freedom should be checked on a monthly basis.

References

(1) FSAR, Section 10.3.1.2

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.

Specification

4.10.1 Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation will be made to determine the cause of the discrepancy.

Bases

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10 percent of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than one percent would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of one percent is considered a safe limit since a shutdown margin of at least one percent with the most reactive rod in the fully withdrawn position is always maintained.

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