March 21, 96

Mr. J. Knubel, Vice President
and Director - TMI
GPU Nuclear Corporation
Post Office Box 480
Middletown, PA 17057-0480

SUBJECT: ISSUANCE OF AMENDMENT - TSCR NO. 254 - (TAC NO. M93970)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 200 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 August 10, 1995, as supplemented on December 21, 1995 and February 22, 1996.

The amendment revises the Technical Specifications to incorporate several improvements from the Revised Standard Technical Specifications for Babcock & Wilcox Plants (NUREG-1430).

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

Sincerely,
Original signed by:

Ronald W. Hernan, 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. 200 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

March 21, 1996

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cc w/encls: See next page

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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. 200
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 August 10, 1995, as supplemented on December 21, 1995 and February 22, 1996, 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. 200, 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

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 21, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 200

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
3-32	3-32
4-2a	4-2a
4-5	4-5
4-10	4-10
4-46	4-46

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
C. <u>Engineered Safety Features (cont'd)</u>			
3. Reactor Building Isolation and Cooling System			
a. Reactor Bldg. 4 psig Instrument Channel	2	1(b)	(a)
b. Manual Pushbuttons			
i. 4 psig feature	2	N/A	(g)
ii. 30 psig feature	2	N/A	(g)
c. Deleted			
d. Reactor Building 30 psig pressure switches	2	1	(c)
e. RCS Pressure less than 1600 psig	2	1(b)	(a)
f. Reactor Building Purge Line Isolation (AH-V1A and AH-V1D) High Radiation	1	0	(f)
4. Reactor Building Spray System			
a. Reactor Building 30 psig pressure switches	2	1	(d)
b. Spray Pump Manual Switches	2	N/A	(g)
5. 4.16KV ES Bus Undervoltage Relays			
a. Degraded Grid Voltage Relays	2	1	(e)
b. Loss of Voltage Relay	2	1	(e)

Bases (Cont'd.)

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 (Reference 1).

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

a) Deleted
b) Monthly with one channel being tested per week on a continuous sequential rotation.

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.

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

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
19.	Reactor Building Emergency Cooling and Isolation System Analog Channels				
	a. Reactor Building 4 psig Channels	S(1)	M(1)	F	(1) When CONTAINMENT INTEGRITY is required.
	b. RCS Pressure 1600 psig	S(1)	M(1)	NA	(1) When RCS Pressure > 1800 psig.
	c. Deleted				
	d. Reactor Bldg. 30 psig pressure switches	S(1)	M(1)	F	(1) When CONTAINMENT INTEGRITY is required.
	e. Reactor Bldg. Purge Line High Radiation (AH-V1A/D)	W(1)	M(1)(2)	F	(1) When CONTAINMENT INTEGRITY is required.
	f. Line Break Isolation Signal (ICCW & NSCCW)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required.
20.	Reactor Building Spray System Logic Channel	NA	Q	NA	
21.	Reactor Building Spray 30 psig pressure switches	NA	M	F	
22.	Pressurizer Temperature Channels	S	NA	R	
23.	Control Rod Absolute Position	S(1)	NA	R	(1) Check with Relative Position Indicator.
24.	Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator.
25.	Core Flooding Tanks				
	a. Pressure Channels	S(1)	NA	F	(1) When Reactor Coolant system pressure is greater than 700 psig.
	b. Level Channels	S(1)	NA	F	
26.	Pressurizer Level Channels	S	NA	R	

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	Semi-Annually 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.

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability: Applies to periodic testing and surveillance requirement of the emergency power system.

Objective: To verify that the emergency power system will respond promptly and properly when required.

Specification:

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiate start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the name-plate rating (3000 kw). This test will be conducted every month on each diesel generator. Normal plant operation will not be effected.

b. Automatically start and loading the emergency diesel generator in accordance with Specification 4.5.1.1.b/c including the following. This test will be conducted every refueling interval on each diesel generator.

(1) Verify that the diesel generator starts from ambient condition upon receipt of the ES signal and is ready to load in ≤ 10 seconds.

(2) Verify that the diesel block loads upon simulated loss of offsite power in ≤ 30 seconds.

(3) The diesel operates with the permanently connected and auto connected load for ≥ 5 minutes.

(4) The diesel engine does not trip when the generator breaker is opened while carrying emergency loads.

(5) The diesel generator block loads and operates for ≥ 5 minutes upon reclosure of the diesel generator breaker.

c. Each diesel generator shall be given an inspection at least annually in accordance with the manufacturer's recommendations for this class of stand-by service.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded:
(1) every 92 days
(2) once within 24 hours after a battery discharge < 105 V
(3) once within 24 hours after a battery overcharge > 150 V
(4) If any cell parameters are not met, measure and record the parameters on each connected cell every 7 days thereafter until all battery parameters are met.

b. The voltage and specific gravity of a pilot cell will be measured and recorded weekly. If any pilot cell parameters are not met, perform surveillance 4.6.2.a on each connected cell within 24 hours and every 7 days thereafter until all battery parameters are met.

c. Each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 200 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

1.0 INTRODUCTION

By letter dated August 10, 1995, as supplemented on December 21, 1995 and January XX, 1996, the GPU Nuclear Corporation (the licensee) submitted a request for changes to the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1) Technical Specifications (TS). The requested changes would revise the TS to make them more consistent with the Revised Standard Technical Specifications for Babcock & Wilcox Plants (RSTS or NUREG-1430). The requested amendment would also change the Bases for TS 4.1.1. The December 21, 1995, letter provided a minor change in one station battery parameter value. The February 22, 1996, letter clarified actiion to be taken if the requirements of TS surveillance 4.6.2.b on station battery pilot cells are not within the specified values. The letters did not change the initial proposed no significant hazards consideration determination.

Specifically, the following changes have been requested:

A. Deletion of one reactor building isolation system (RBIS) instrumentation input signal from the TS TMI-1 (Table 3.5-1, item C.3.c Table 4.1-1, item 19.c). This input signal is the reactor protection system (RPS) trip that would cause the RBIS circuitry to actuate. The RSTS do not include this parameter.

B. Revision of the sodium hydroxide tank sampling frequency from quarterly (TS Table 4.1-3 Item 10) to semiannually (184 days) as specified in the RSTS surveillance requirements (SRs) (SR 3.6.7.3). The requirement for sampling the sodium hydroxide tank after each makeup would be retained.

C. Revision of the station battery individual cell surveillance frequency from monthly (TS 4.6.2.a) to quarterly (92 days) as specified in the RSTS SRs (SR 3.8.6.2) and clarification of action to be taken if the requirments of TS surveillance 4.6.2.b on station battery pilot cells are not within the specified values.

D. Revision of the Bases for TS Surveillance 4.1.1 to delete references to testing requirements for the reactor protection channels and the regulated control rod power supplies prior to reactor startup that are not required by Surveillance Table 4.1-1 or by the RSTS.

2.0 BACKGROUND

Section 50.36 of Title 10 of the Code of Federal Regulations (10 CFR 50.36) establishes the regulatory requirements related to the content of TS. The rule requires that TS include items in specific categories, including safety limits, limiting conditions for operation, and surveillance requirements; however, the rule does not specify the particular requirements to be included in a plant's TS. The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," (58 FR 39132, July 22, 1993) to determine which of the design conditions and associated surveillances need to be located in the TS because the requirement is "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." Briefly, those criteria are (1) detection of abnormal degradation of the reactor coolant pressure boundary, (2) boundary conditions for design basis accidents and transients, (3) primary success paths to mitigate design basis accidents and transients, and (4) functions determined to be important to risk or operating experience. The Commission's final policy statement acknowledged that its implementation may result in the relocation of existing TS requirements to licensee-controlled documents and programs. The criteria stated in the policy statement have also been recently codified in NRC regulations in a change to 10 CFR 50.36 (60 FR 36953, July 19, 1995). The FEDERAL REGISTER Notice related to this rulemaking stated the following:

> ... Each licensee covered by these regulations may voluntarily use the criteria as a basis to propose the relocation of existing technical specifications that do not meet any of the criteria from the facility license to licensee-controlled documents. The voluntary conversion of current technical specifications in this manner is expected to produce an improvement in the safety of nuclear power plants through a reduction in unnecessary plant transients and more efficient use of NRC and industry resources.

3.0 EVALUATION

Reactor Building Isolation Instrumentation (pp. 3-32, 4-5)

The current TMI-1 TS contains a Limiting Condition for Operation (LCO) (Table 3.5-1, Item C.3.c) and a surveillance requirement (SR) (Table 4.1-1, Item 19.c) for an RPS trip signal which would result in an RBIS actuation. The proposed changes would delete these requirements. The RPS trip signal provides a diverse backup to other containment isolation signals. These other signals (4 psig reactor building pressure, 1600 psig reactor coolant system (RCS) pressure, and 30 psig reactor building pressure) are redundant and have LCOs and SRs of their own specified in TS. The TMI-1 Updated Final Safety Analysis Report (UFSAR) describes the RBIS design basis and operation and states that the RPS input to the RBIS logic as a "conservative, diverse containment isolation signal." Most of the lines isolated by this signal are normally isolated during plant operation, such as the RCS and steam generator sample line. No valves associated with the emergency core cooling system (ECCS) are actuated by this signal. The RPS trip signal is not included in the B&W RSTS. It is not a part of the primary success path for containment isolation and neither operating experience nor the TMI-1 probabilistic risk analysis has shown it to be significant to public health and safety.

The staff has reviewed the licensee's request and finds that including these requirements in the TS does not meet the criteria in the NRC "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," as codified by the revision to 10 CFR 50.36. They are, therefore, not required to be maintained in the TS and can be deleted as requested. The proposed amendment is expected to produce an improvement in safety through reduced potential for LCO-induced plant transients.

<u>Sodium Hydroxide Tank Sampling Frequency</u> (p. 4-10)

The sodium hydroxide tank at TMI-1 is part of the reactor building spray system and provides a caustic solution to remove radioiodine from the reactor building atmosphere following a loss-of-coolant accident and controls reactor building sump pH. The current TMI-1 TS contain a SR (Table 4.1-3, Item 10) that requires verification of the sodium hydroxide tank concentration quarterly and after each makeup. The proposed change would require verification of the sodium hydroxide tank concentration semiannually and after each makeup. This requested change is more conservative than RSTS (SR 3.6.7.3), since RSTS does not specify sampling after each makeup. As stated in RSTS Bases for SR 3.6.7.3, "a 184 day frequency is sufficient to ensure that the concentration level of [sodium hydroxide] NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected." GPUN has concluded that this basis is applicable to TMI-1. The licensee concluded, therefore, that revision of the surveillance specifying the sampling for the sodium hydroxide tank from quarterly to the RSTS (SR 3.6.7.3) frequency of every 184 days (retaining the current "after each makeup") is consistent, if not more conservative, with the RSTS and is justified.

The staff has reviewed this request and the licensee's conclusion and finds them to be consistent with the requirements of the B&W RSTS. The requirements are therefore, acceptable.

<u>Station Battery Individual Cell Surveillance Requirements</u> (p. 4-46)

The TMI-1 TS contain a SR (4.6.2.a) that requires that the voltage, specific gravity, and liquid level of each cell be measured and recorded monthly. The proposed change would extend the surveillance interval to 3 months and add checks following battery discharge or battery overcharge events. The quarterly checks of liquid level, specific gravity, and voltage are consistent with IEEE-450 and the B&W RSTS (SR 3.8.6.2).

In addition to quarterly checks, the B&W RSTS has a requirement to check the battery voltage, specific gravity, and liquid level within 24 hours of a severe battery discharge when battery voltage drops to <110 volts or a battery overcharge results in battery voltage of >150 volts. The licensee's original proposal requested this lower voltage limit to be 100 volts. Based on subsequent discussions with the staff, the licensee decided to raise this value to 105 volts and submitted a revised TS in letter dated December 21, 1995. A voltage lower than 110 volts (RSTS) is appropriate for TMI-1 because the number of battery cells is lower than in some plants and this value is not absolute because of the margin to inoperability. The RSTS allows for a plant specific value to be used for this parameter. This a minor change to the original request and has no significant effect on the surveillance testing program. The checks following a severe battery discharge or overcharge are also consistent with IEEE-450, which recommends such checks following a severe discharge or overcharge to ensure that no significant degradation of the battery has occurred as a consequence.

The staff has reviewed the licensee's rationale for this request and finds that it is generally consistent with the current staff position and with IEEE-450. However, the licensee's submittal did not include changes to the same level of detail as the RSTS (NUREG-1430). For example, the licensee did not request to incorporate RSTS Table 3.8.6-1, which specifies the acceptable values (acceptance criteria) for the parameters measured by this surveillance test. The staff was supplied information that indicated where the TMI-1 acceptance criteria are located and what the values are. These criteria are located in plant procedures SP 1301-4.6, SP 1301-5.8, SP 1303-11.11, and PM E-72. Without exception, these criteria are equivalent to or more conservative than the RSTS criteria. Likewise, a RSTS requirement (LCO 3.8.6.A.2) to check the subject parameters for all connected cells within 24 hours if the pilot cells do not meet the weekly surveillance acceptance criteria was not requested by the licensee. This requirement was added to the proposed TS by a GPUN letter dated January XX, 1996. The staff also reviewed the overall TMI-1 station battery maintenance and surveillance test program to ensure that other elements of the RSTS not in the TMI-1 TS are part of the program. The staff found that TMI-1 has an equivalent program and in some areas has enhancements that are not included in the RSTS, such as infra-red scanning of all battery connections during the battery discharge test.

The staff concluded that, with the changes discussed above, the revision of the surveillance frequency for individual battery cell voltage, specific gravity, and liquid level to quarterly is acceptable.

<u>Revision to Bases for Surveillance of Reactor Protection Channels</u> (p. 4-2a)

This change revises the Bases for TMI-1 TS SR 4.1 to delete a reference to testing the reactor protection channels coincidence logic, the control rod drive trip breakers, and the regulating control rod power electronic trips before startup, when shutdown greater than 24 hours. The B&W RSTS do not include this requirement. This reference is not appropriate because there is no such requirement in either TS 4.1, "Operational Safety Review," or in TS Table 4.1-1. The Basis section is not part of the TS, as defined in 10 CFR 50.36, which states "A summary statement of the bases or reasons for such specifications ... shall also be included in the application, but shall not become part of the technical specifications." The staff finds that these tests need not be added to the TS as requirements in view of the monthly channel testing currently required by TS, with one channel being tested once per week on a continuous sequential rotation. The monthly testing provides adequate verification of reactor protection channel availability. There are also no such additional requirements in the B&W RSTS.

Therefore, the staff agrees that it is appropriate to delete the subject references from the Bases for TS 4.1.

3.0 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 comments.

4.0 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 20 and changes surveillance requirements. 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 58401). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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 Contributor: Ronald W. Hernan

Date: March 21, 1996