June 15, 1999

Mr. James W. Langenbach, Vice President
and Director, TMI
GPU Nuclear, Inc.
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: CERTAIN IMPROVEMENTS FROM THE STANDARD TECHNICAL SPECIFICATIONS FOR BABCOCK AND WILCOX PLANTS (TAC NO. M97389)

Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No. 211 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) in response to your application dated December 3, 1996.

The amendment will incorporate certain improvements from the Standard Technical Specifications for Babcock and Wilcox plants (NUREG-1430).

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

Sincerely,

Original signed by:

Ronald B. Eaton, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No211 to DPR-50
2. Safety Evaluation

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DATED: June 15, 1999

AMENDMENT NO. 211 TO FACILITY OPERATING LICENSE NO. DPR-50 THREE MILE ISLAND

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

June 15, 1999

Mr. James W. Langenbach, Vice President
and Director, TMI
GPU Nuclear, Inc.
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: CERTAIN IMPROVEMENTS FROM THE STANDARD TECHNICAL SPECIFICATIONS FOR BABCOCK AND WILCOX PLANTS (TAC NO. M97389)

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Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 211 to DPR-50
2. Safety Evaluation

cc w/encls: See next page

Three Mile Island Nuclear Station, Unit No. 1

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DATED: June 15, 1999

AMENDMENT NO. 211 TO FACILITY OPERATING LICENSE NO. DPR-50 THREE MILE ISLAND

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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, INC.

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 211
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear, Inc., et al. (the licensee) dated, December 3, 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.

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PDR ADOCK 05000289
P PDR

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. 211 are hereby incorporated in the license. GPU Nuclear, Inc. 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 15, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 211

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 areas of change.

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3.1.10 CONTROL ROD OPERATION

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(Page 3-18b deleted)

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the operating status of the emergency core cooling, reactor building emergency cooling, and reactor building spray systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

3.3.1 The reactor shall not be made critical unless the following conditions are met:

3.3.1.1 Injection Systems

a. The borated water storage tank shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, restore the BWST to OPERABLE within 8 hrs. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

b. Two makeup pumps are operable in the engineered safeguards mode powered from independent essential buses. Specification 3.0.1 applies.

c. Two decay heat removal pumps are operable. Specification 3.0.1 applies.

d. Two decay heat removal coolers and their cooling water supplies are operable. (See Specification 3.3.1.4) Specification 3.0.1 applies.

e. Two BWST level instrument channels are operable.

f. The two reactor building sump isolation valves (DH-V6A, DH-V6B) shall be either manually or remote-manually operable. Specification 3.0.1 applies.

3.3.1.2 Core Flooding System

a. Two core flooding tanks each containing 1040 $\pm$ 30 ft^3 of borated water at 600 $\pm$ 25 psig shall be available. Specification 3.0.1 applies.

b. Core flooding tank boron concentration shall not be less than 2,270 ppm boron.

c. The electrically operated discharge valves from the core flood tank will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

d. One core flood tank pressure instrumentation channel and one core flood tank level instrumentation channel per tank shall be operable.

e. Core flood tank (CFT) vent valves CF-V3A and CF-V3B shall be closed and the breakers to the CFT vent valve motor operators shall be tagged open, except when adjusting core flood tank level and/or pressure. Specification 3.0.1 applies.

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The sodium hydroxide (NaOH) tank shall be maintained at 8 ft. $\pm$6 inches lower than the BWST level as measured by the BWST/NaOH tank differential pressure indicator. The NaOH tank concentration shall be 10.0 $\pm$.5 weight percent (%). If the NaOH concentration is not within limits, restore to OPERABLE within 72 hours. If the BWST/NaOH tank level differential is not within limits, restore to OPERABLE within 72 hours.

c. All manual valves in the discharge lines of the sodium hydroxide tank shall be locked open.

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.

b. Two nuclear service river water pumps must be OPERABLE.

c. Two decay heat closed cycle cooling water pumps must be OPERABLE.

d. Two decay heat river water pumps must be OPERABLE.

e. Two reactor building emergency cooling river water pumps must be OPERABLE.

3.3.1.5 Engineered Safeguards Valves and Interlocks Associated with the Systems in Specifications 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4 are OPERABLE. Specification 3.0.1 applies.

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, CFT pressure instrumentation, CFT level instrumentation, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

3.4 DECAY HEAT REMOVAL CAPABILITY

Applicability

Applies to the operating status of systems and components that function to remove decay heat when one or more fuel bundles are located in the reactor vessel.

Objective

To define the conditions necessary to assure continuous capability of decay heat removal.*

Specification

3.4.1 Reactor Coolant System temperature greater than 250°F.

3.4.1.1 With the Reactor Coolant System temperature greater than 250°F, three independent EFW pumps and associated flow paths shall be OPERABLE ** with:

a. Two EFW pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW pump capable of being powered from an OPERABLE steam supply system.

(1) With one pump or flow path inoperable, restore the inoperable pump or flow path to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours.

(2) With more than one EFW pump or flow path inoperable, restore the inoperable pumps or flow paths to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 12 hours.

NOTE: When EF-P-1 and EF-P-2A or EF-P-2B become inoperable due to TS surveillance, entry into this LCO may be delayed for up to 8 hours.

b. Four of six turbine bypass valves OPERABLE. With more than two turbine bypass valves inoperable, restore operability of at least four turbine bypass valves within 72 hours.

c. The condensate storage tanks (CST) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST.

(1) With a CST inoperable, restore the CST to operability within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours.

(2) With more than one CST inoperable, restore the inoperable CST to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

---

* These requirements supplement the requirements of Sections 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.
**HSPS operability is specified in Section 3.5.1.

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall not be less than one percent $\Delta K/K$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod as defined in Specification 4.7.1 and 4.7.2.3 in the safety or regulating rod banks shall not be permitted Verify SDM $\geq 1\%$ $\Delta k/k$ or initiate boration to restore within limits within 1 hour. The reactor shall be brought to HOT SHUTDOWN within 6 hours.

b. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent $\Delta k/k$ hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, and once per 12 hours thereafter, it is not determined that a one percent $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2 and cannot be aligned per 3.5.2.2.f, power shall be reduced to $\leq 60\%$ of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to $\leq 70\%$ of the thermal power allowable within 10 hours. Verify the potential ejected rod worth (ERW) is within the assumptions of the ERW analysis and verify peaking factor ($F_Q(Z)$ and $F^N_{\Delta H}$) limits per the COLR have not been exceeded within 72 hours.

f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation may continue provided that within 1 hour the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

g. If the inoperable rod in Paragraph "e" above is in groups 5, 6, 7, or 8, the other rods in the group may be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that within 1 hour the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality is limited by the restriction of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Tilt:

a. Except for physics tests, the quadrant tilt, as determined using the full incore system (FIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

The FIS is OPERABLE for monitoring quadrant tilt provided the number of valid symmetric string individual SPND signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore system is not OPERABLE and except for physics tests quadrant tilt as determined using the power range channels for each quadrant (out of core detector system)(OCD), shall not exceed the values in CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests, quadrant tilt as determined using the minimum incore system (MIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if quadrant tilt exceeds the tilt limit, allowable power shall be reduced 2 percent for each 1 percent tilt in excess of the tilt limit. For less than four pump operation, thermal power shall be reduced 2 percent below the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of the tilt limit.

e. If quadrant power tilt exceeds the tilt limit then within a period of 10 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following verifications and/or adjustments in setpoints and limits shall be made:

1. Verify $F_Q(Z)$ and $F^N_{\Delta H}$ are within limits of the COLR once per 2 hours and restore QPT to ≤steady state limit within 24 hours, or perform steps 2,3, &4 below.

2. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt, in excess of the tilt limit, or when thermal power is equal to or less than 50% full power with four reactor coolant pumps running, set the nuclear overpower trip setpoint equal to or less than 60% full power.

3. The control rod group withdrawal limits in the CORE OPERATING LIMITS REPORT shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

4. The operational imbalance limits in the CORE OPERATING LIMITS REPORT shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

f. Except for physics or diagnostic testing, if quadrant tilt is in excess of the maximum tilt limit defined in the CORE OPERATING LIMITS REPORT and using the applicable detector system defined in 3.5.2.4.a, b, and c above, reduce thermal power to ≤15% FP within 2 hours. Diagnostic testing during power operation with a quadrant tilt is permitted provided that the thermal power allowable is restricted as stated in 3.5.2.4.d above.

g. Quadrant tilt shall be monitored on a minimum frequency of once every 12 hours when the QPT alarm is inoperable and every 7 days when the alarm is operable during power operation above 15 percent of rated power. When QPT has been restored to ≤steady state limit, verify hourly for 12 consecutive hours, or until verified acceptable at ≥95% FP.

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ± 5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and $F_Q(Z)$ and $F^N_{\Delta H}$ shall be verified within limits once every 2 hours, or power shall be reduced to $\leq$power allowed by insertion limits.

2. If regulating rods are inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to $\geq 1\% \Delta K/K$, and restore regulating rods to within restricted region within 2 hours or reduce power to $\leq$power allowed by rod insertion limits.

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Director, Operations and Maintenance, TMI.

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify $F_Q(Z)$ and $F^N_{\Delta H}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

e. If an acceptable axial power imbalance is not achieved within 24 hours, reactor power shall be reduced to ≤40% FP within 2 hours.

f. Axial power imbalance shall be monitored on a minimum frequency of once every 12 hours when axial power imbalance alarm is OPERABLE, and every 1 hour when imbalance alarm is inoperable during power operation above 40 percent of rated power.

3.5.2.8 A power map shall be taken at intervals not to exceed 31 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in the CORE OPERATING LIMITS REPORT.

Bases

The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. The effects of the APSRs are included in the limit development. Additional conservatism included in the limit development is introduced by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors

e. Postulated fuel rod bow effects

f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

The incore instrumentation system uncertainties used to develop the axial power imbalance and quadrant tilt limits accounted for various combinations of invalid Self Powered Neutron Detector (SPND) signals. If the number of valid SPND signals falls below that used in the uncertainty analysis, then another system shall be used for monitoring axial power imbalance and/or quadrant tilt.

For axial power imbalance and quadrant power tilt measurements using the incore detector system, the minimum incore detector system consists of OPERABLE detectors configured as follows:

Axial Power Imbalance

a. Three detectors in each of three strings shall lie in the same axial plane with one plane in each axial core half.

b. The axial planes in each core half shall be symmetrical about the core mid-planes.

c. The detectors shall not have radial symmetry.

Quadrant Power Tilt

a. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

A system of 52 incore flux detector assemblies with seven detectors per assembly has been provided primarily for fuel management purposes. The system includes data display and record functions and is also used for out-of-core nuclear instrumentation calibration and for core power distribution verification.

a. The out-of-core instrumentation calibration includes:

1. Calibrations of the split detectors at initial reactor startup, during the power escalation program, and periodically thereafter.

2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during operation.

3. Confirmation that the out-of-core axial power splits are as expected.

b. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation to ensure that power distribution is consistent with calculations.

3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

c. The minimum requirement for 23 individual incore detectors is based on the following:

1. An adequate axial imbalance indication can be obtained with nine individual detectors. Figure 3.5-1 shows a typical set of three detector strings with three detectors per string that will indicate an axial imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

2. Figure 3.5-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from two detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. Figure 3.5-3 combines Figures 3.5-1 and 3.5-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions.

The 25 ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping rod bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5, 6 and 7 are overlapped 25 %. The normal position at power is for group 7 to be partially inserted.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (Reference 1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% delta k/k at rated power. These values have been shown to be safe by the safety analysis of the hypothetical rod ejection accident (Reference 2). A maximum single inserted control rod worth of 1.0% delta k/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% delta k/k at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than 0.65% delta k/k ejected rod worth at rated power.

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, then manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed as specified until the computer is returned to service.

Reduction of the nuclear overpower trip setpoint to 60% full power when thermal power is equal to or less than 50% full power maintains both core protection and an operability margin at reduced power similar to that at full power.

During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

Test Power	Test Setpoint
0	<5%
≤80	90%
>80	105.1%

REFERENCES

(1) UFSAR, Section 3.2.2.1.2 - "Reactivity Control Distribution"

(2) UFSAR, Section 14.2.2.2 - "Rod Ejection Accident"

3.5.4 INCORE INSTRUMENTATION

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TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2.	Control Rod Movement	Movement of each rod	Every 92 days, when reactor is critical
3.	Pressurizer Safety Valves	Setpoint	In accordance with the Inservice Testing Program
4.	Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5.	Refueling System Interlocks	Functional	Start of each refueling period
6.	Main Steam Isolation Valves	(See Section 4.8)	
7.	Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8.	(Deleted)	–	–
9.	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10.	Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation-Visual inspection of Intake Pump House Floor	Not to exceed 24 months
		(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11.	Pressurizer Block Valve (RC-V2)	Functional*	Quarterly

* Function shall be demonstrated by operating the valve through one complete cycle of full travel.

TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

ITEM	CHECK	FREQUENCY
1. Reactor Coolant	a. Specific Activity Determination to compare to the 100/ÉµCi/gm limit	At least once each 7 days during POWER OPERATION, HOT STANDBY, STARTUP, and HOT SHUTDOWN.
	b. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	i) 1 per 14 days during power operations. ii) One Sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a one hour period during power operation, start-up, and hot standby. iii) # Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/ÉµCi/gram during all modes but refueling.
	c. Radiochemical for É Determination	1 per 6 months* during power operation
	d. Chemistry (C1, F and 0_2)	5 times/week when T_{avg} is greater than 200°F.
	e. Boron concentration	2 times/week
	f. Tritium Radioactivity	Monthly
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 T_{avg} 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.

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the axial power shaping rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at hot reactor coolant full flow conditions or 1.40 seconds for the hot no flow conditions (Reference 1). For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has actuated the 25% withdrawn reference switch during insertion from the fully withdrawn position. The specified trip time is based upon the safety analysis in UFSAR, Chapter 14 and the Accident Parameters as specified therein.

Each control rod drive mechanism shall be exercised by a movement of a minimum of 3% of travel at a minimum of every 92 days. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

4.7.2 CONTROL ROD PROGRAM VERIFICATION (Group vs. Core Positions)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.211 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR, INC.

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated December 3, 1996, the GPU Nuclear, Inc. (the licensee) submitted a request for changes to the Three Mile Island Nuclear Station, Unit 1 (TMI-1) Technical Specifications (TSs). This change would incorporate certain improvements from the Standard Technical Specifications (STSs) for Babcock and Wilcox (B&W) Plants (NUREG-1430). Additionally, there are some administrative, spelling and typographical corrections included.

2.0 BACKGROUND

The licensee's application includes the removal of information from the TSs. Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. That regulation requires that the TSs include items in five specific categories, including:

(1) safety limits, limiting safety system settings and limiting control settings;

(2) limiting conditions for operation;

(3) surveillance requirements;

(4) design features; and

(5) administrative controls.

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On July 19, 1995, the Commission published revisions to 10 CFR 50.36 specifying what must be included in limiting conditions for operation (LCO) in the TSs (60 FR 36953). The four criteria added to 10 CFR 50.36 for determining whether a particular matter is required to be included in the TSs, are as follows:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier;

(3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; and

(4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As a result, existing TS LCOs which fall within or satisfy any of the criteria in 10 CFR 50.36 must be retained in the TSs, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

3.0 EVALUATION

3.1 Table of Contents, pages ii, iii, and vii

The corrections to the Table of Contents pages incorporate changes to the TSs, correct typographical errors, and correct an omission associated with the issuance of TS Amendment 170. The NRC staff finds these corrections acceptable.

3.2 Limiting Condition for Operation 3.1.10 and associated Figure 3.1-3

The licensee proposed to delete this LCO on the basis that 1) the LCO does not meet any of the requirements of Criterion 1, 2, 3, or 4 in 10 CFR 50.36(c)(2)(ii) for retention in TS; 2) they monitor the allowable combinations of pressure and temperature for control rod operation and dissolved gas limits for the reactor coolant system (RCS) in controlled procedures specific to TMI-1; and 3) it is not a part of the STS. The requirement that dissolved gas concentration remains below 100 standard cubic centimeters per liter was designed to preclude the formation of gas bubbles in significant amounts during depressurization. By maintaining the reactor coolant temperature and pressure within specified limits, any dissolved gases in the RCS are maintained in solution. This request and the reasons given are similar to those granted to other B&W plants. During startup, power operation, and cooldown, the gas content of the RCS is controlled through operating procedures and practices such as venting and temperature/pressure control. Instrumentation, sampling and procedures are in place to ensure that the appropriate actions are taken to preclude conditions that would result in "gas-out" problems. In view of the unlikelihood of a "gas-out" incident, and installed monitoring instrumentation and plant-specific controlled procedures to preclude such an occurrence, the

staff finds the amendment to delete LCO 3.1.10 and associated Figure 3.1-3 acceptable and does not meet the criteria necessary for retention in the TSs.

3.3 Limiting Condition for Operation 3.3 Emergency Core Cooling and Reactor Building Spray System

The proposed TS change in this section adds allowed outage time (AOT) to: 1) restore the Borated Water Storage Tank (BWST) boron concentration, water temperature, and volume to operable status; 2) provide a timeclock to restore to open the core flood tank discharge valve; and 3) restore NaOH concentration and BWST/NaOH tank differential limits.

The TMI-1 TSs have not had AOTs for these functions. The BWST AOT of 8 hours to restore boron concentration and water temperature and 1 hour to restore tank volume are consistent with the STS and are acceptable. The AOT of 1 hour to open the Core Flood Tank discharge valve is also consistent with STSs and is acceptable. Establishing an AOT for the proposed changes associated with the NaOH tank concentration is more conservative than having none and the time allowed is reasonable. This proposed change is acceptable to the staff.

3.4 Limiting Condition for Operation 3.4, Decay Heat Removal Capability

The proposed changes in this section provide two editorial changes and one timeclock change, from 6 to 12 hours, for the licensee to be in cold shutdown with respect to the operability of the emergency feedater (EFW) pumps or flow paths. The editorial corrections are acceptable. The AOT change allows for a more controlled power reduction and extends the time allowed to be in cold shutdown without altering the requirement to be in hot shutdown within 6 hours, and is consistent with the STSs and, therefore, is also acceptable.

3.5 Limiting Condition for Operation 3.5.2, Control Rod Group and Power Distribution Limits

The correction of the misspellings in the Objective section and subsection 3.5.2.2 are acceptable.

3.5.1 Subsection 3.5.2.2.a

There are currently no timeclocks or actions associated with more than one inoperable rod in the safety or regulating rod banks. This proposed change establishes both actions to bring the reactor to a hot shutdown condition and verification of shutdown margin (SDM) and establishes timeclocks to do so that are consistent with the STSs. This change allows for orderly action while permitting adequate time for operator actions and is acceptable.

3.5.2 Subsection 3.5.2.2.c

There is currently no timeclock associated with bringing the reactor to a hot shutdown condition and no follow-up determination of adequate SDM for this subsection. This change establishes the addition of an SDM determination 12 hours after the initial determination and establishes a 6-hour timeclock to reach a hot shutdown condition. This change is more conservative in that it establishes a timeclock to determine the adequacy of SDM and time to reach a hot shutdown condition and it is consistent with the STSs and is acceptable to the staff.

3.5.3 Subsection 3.5.2.2.e

The proposed changes to this subsection include: 1) an editorial clarifying change linking this Subsection to Subsection 3.5.2.2.f with respect to rod group alignment; 2) a procedural change for modification of the power reduction for misaligned rod configurations from 60 percent to less than or equal to 60 percent; and 3) the establishment of timeclocks for power reduction, overpower trip setpoint reduction, and verification that ejected rod worth (ERW) within the assumptions of the ERW analysis to verify that the power peaking factors are within the limits specified by the Core Operating Limits Report.

These changes are additional requirements that are consistent with requirements already in the STS and provide reasonable and sufficient time to perform the required actions and evaluations and are acceptable.

3.5.4 Subsection 3.5.2.2.f

There currently is no timeclock associated with the positioning of rods within a group that has an inoperable rod per Specification 4.7.1.2. This proposed change establishes a 1-hour timeclock to perform the required action which is reasonable and is consistent with the STSs and is acceptable.

3.5.5 Subsection 3.5.2.2.g

There is no timeclock associated with the actions required by this Subsection. The proposed changes establishes a 1-hour timeclock, which is acceptable because local xenon redistribution during this short interval will not cause a significant increase in local linear heat rate and it is consistent with the STSs.

3.6 Limiting Condition for Operation 3.5.4, Incore Instrumentation

Existing LCO 3.5.4 provides the specification for the functional and operational requirements for the incore detectors. The incore detector system is used to provide detailed information on the reactor core neutron flux distribution. The incore detector system is used for periodic surveillance of the power distribution, which verifies that the peaking factors are within their design envelope. Other (retained) TSs address the required power distribution limits, peaking factors, and excore detector calibrations. The incore detectors map the spatial neutron flux distribution of the core and provide this information to the plant computers. The provisions of the existing incore detector LCO, establishing incore detector operational requirements, are relocated in plant-specific procedures. The power range (excore) neutron flux instrumentation is also available to measure axial power distribution (axial imbalance) and quadrant power tilt. The reactor protection system does not use the incore detector instrumentation but instead uses the power range (excore) neutron flux instrumentation to initiate a reactor trip as a result of unacceptable axial core power distribution. No automatic actions result from the incore detector system. The incore detector instrumentation does not provide an indication of a degradation of the reactor coolant pressure boundary or an indication of a challenge to the integrity of a fission product boundary, nor does it mitigate the consequences of a design-basis accident (DBA), and therefore, it does not meet 10 CFR 50.36(c)(2)(ii) for retention in the improved TSs. The staff finds this change acceptable.

3.7 Surveillance Requirement, Table 4.1-2 and Surveillance 4.7-1

The frequency of the surveillance to verify rod freedom of movement (trippability), Surveillance Requirement (SR) 4.7-1 and Table 4.1-2, has been extended from monthly to quarterly in accordance with the recommendation in NUREG-1366, "Improvements to TS Surveillance Requirements." This SR frequency has been extended because of the unlikelihood of a stuck rod, the potential for a dropped rod during the performance of the SR, and the fact that an untrippable rod is most often discovered when starting up or when performing rod drop tests. The licensee's proposed extension change is, therefore, acceptable.

3.8 Surveillance Requirement, Table 4.1-3

In improved TS 3.4.16, "RCS Specific Activity," the frequency of SR 3.4.16.1, to measure RCS gross specific activity, has been extended from 72 hours to 7 days because of the time required to perform the analysis and the more immediate availability of data from the Dose Equivalent I-131 analysis (SR 3.4.16.2) which is performed within 2 to 6 hours of power changes of 15 percent or more of rated thermal power.

The proposed less restrictive requirement has been reviewed by the staff and has been found to be acceptable because it does not present a significant safety question in the operation of the plant and, similarly, the TMI-1 plant-specific TSs require the Dose Equivalent I-131 analysis within 2 to 6 hours of power changes of 15 percent or more of rated thermal power. The TS requirements that remain are consistent with current licensing practices, operating experience and plant accident and transient analyses, and provide reasonable assurance that public health and safety will be protected and are, therefore, acceptable.

3.9 Surveillance Requirement, 4.7-2

The licensee proposed to delete SR 4.7-2, which is a surveillance to verify that the designated control rod is operating in its functional position and group. The type of errors this surveillance would detect is addressed in the TMI-1 plant-specific procedures. This SR is not included in the STSs nor is there a LCO associated with this SR in the existing TSs. The licensee has applied the four criteria pursuant to 10 CFR 50.36(c)(2)(ii) to this SR and determined that none of the criteria are either applicable or are met and, therefore, the standard to establish or retain this SR is not met. Additionally, proper rod programming is addressed in TMI-1 plant-specific procedure 1301-9.2. In that this SR is addressed in a controlled plant procedure and it does not meet the standards for an inclusion in the TSs per 50.36(c)(3), the staff finds this proposed change acceptable.

3.10 Bases Pages

The licensee has made changes to various bases pages which are not subject to staff approval, but are included for incorporation with the amendment being issued.

4.0 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.

5.0 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 (61 FR 66708). 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.

6.0 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: R. Eaton

Date: June 15, 1999