December 1, 1987

Docket No. 50-289

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: AMENDMENT NO. 135 TO FACILITY OPERATING LICENSING NO. DPR-50 (TAC #64822)

The Commission has issued the enclosed Amendment No. 135 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 March 5, 1987 (Technical Specification Change Request (TSCR) No. 154).

The amendment revises the Technical Specifications (TSs) to: (1) increase the setpoint for reactor trips on high pressure from 2300 to 2355 psig; (2) raise the arming threshold for anticipatory reactor trip on turbine trip from the curretn 20% reactor power level to 45% reactor power level; (3) improve the language of the bases of the Technical Specification Safety Limit Section.

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

Sincerely,

ORIGINAL SIGNED BY:

Gordon E. Edison, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 135 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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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. 135
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 March 5, 1987, 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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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. 135 , 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, Director
Project Directorate I-4
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 1, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 135

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
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
Figure 2.3-1	Figure 2.3-1
3-27a	3-27a
3-28a	3-28a
3-30	3-30
4-7	4-7

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure (2). The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established(2). The maximum settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig)(3) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by the presence of a pressurizer electromatic relief valve (4).

References

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained below the safety limit (2750 psig) for any design transient (6). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated overpressurization transients (8). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

The low pressure (1800 psig) and variable low pressure (11.75 T_{OUT}-5103) trip setpoint were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (3,4). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. Figure 2.3-1 shows the high pressure, low pressure, and variable low pressure trip setpoints.

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618.8F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RPS is 520° to 620°F. The trip setpoint of the channel is 618.8F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1.2°F.This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip setpoint of 620°F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip setpoint must be reduced to value $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip setpoint of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

References

(1) FSAR, Section 14.1.2.3
(2) FSAR, Section 14.1.2.2
(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

(6) Technical Specification Change Request No. 31, January 16, 1976, and Technical Specification Change Request No. 84, June 23, 1978.

(7) "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103, Rev. 2, Babcock and Wilcox, April 1976.

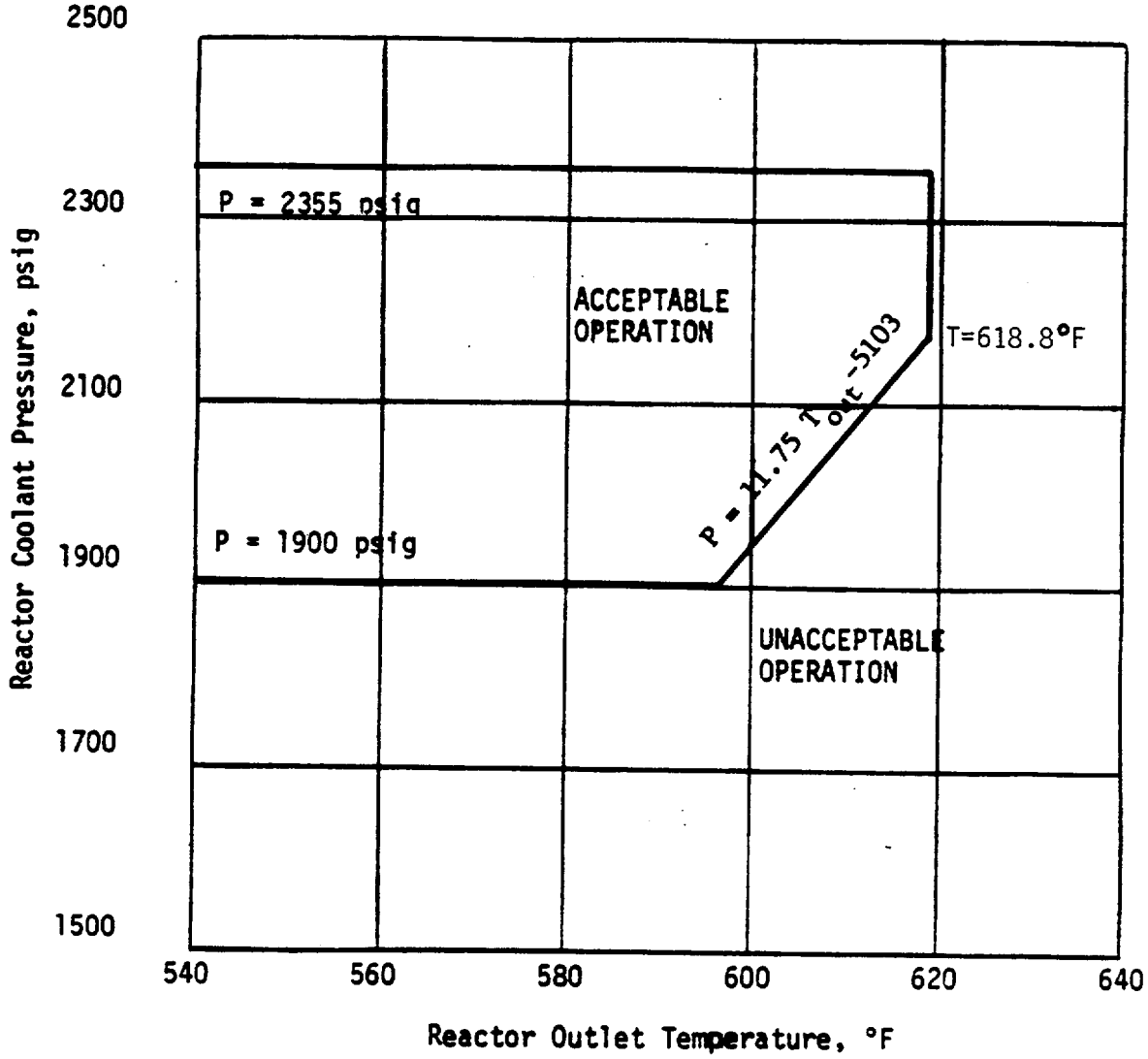
(8) "Justification for Raising Setpoint for Reactor Trip on High Pressure", BAW-1890, Rev. 0, Babcock and Wilcox, September 1985.

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%	Shutdown Bypass
1. Nuclear power, Max. % of rated power	105.1	105.1	105.1	5.0(3)
2. Nuclear power based on flow (2) and imbalance max. of rated power	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (5) on pump monitors, Max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2355	2355	2355	1720(4)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Variable low reactor coolant system pressure, psig min.	11.75 Tout-5103)(1)	11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	Bypassed
7. Reactor coolant temp. F., Max.	618.8	618.8	618.8	618.8
8. High Reactor Building pressure, psig max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F).
(2) Reactor coolant system flow, %.
(3) Administratively controlled reduction set only during reactor shutdown.
(4) Automatically set when other segments of the RPS (as specified) are bypassed.
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.
(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable connectors

Amendment No. 13, 17, 28, 39, 48, 78, 90, 126, 135



TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1

3.5.1.7.1 Power may be restored through the breaker with the failed trip feature for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.8 During STARTUP, HOT STANDBY or POWER OPERATION, in the event that one of the two regulating control rod power SCR electronic trips is inoperable, within one hour:

a. Place the inoperable SCR electronic trip in the tripped condition or
b. Remove the power supplied to the associated SCRs. Specification 3.0.1 applies.

3.5.1.8.1 Power may be restored through the SCRs with the failed electronic trip for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.9 The reactor shall not be in the Startup mode or in a critical state unless both HSPS actuation logic trains associated with the Functional units listed in Table 3.5-1 are operable except as provided in Table 3.5-1,D.

3.5.1.9.1 With one HSPS actuation logic train inoperable, restore the train to OPERABLE or place the inoperable device in an actuated state within 72 hours or be in HOT SHUTDOWN within the next 12 hours. With both HSPS actuation logic trains inoperable, restore one train to OPERABLE within 1 hour or be in HOT SHUTDOWN within the next 6 hours.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and four high reactor building pressure instrument channels. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. the bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power. The safety feature actuation system must have two analog channels functioning correctly prior to startup.

The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other instrumentation channels is one out of two.

HSPS instrument operability specified meets the single failure criterion for the EFW system. Four instrument channels are provided for automatic EFW initiation on OTSG low level and high reactor building pressure, and for automatic main feedwater isolation on low OTSG pressure. Normal trip logic is two out of four. With one of the 4 channels in bypass, a second channel may be taken out of service (placed in the tripped position) and no single active failure will prevent actuation of the associated HSPS train actuation logic. No single active failure of either HSPS train will prevent the other HSPS train from operating to supply EFW to both OTSGs.

REFERENCE

(1) FSAR, Section 7.1

(2) "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, Rev. 0, Babcock & Wilcox, October 1985.

Table 3.5-1 Continued

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
A. Reactor Protection system (Cont'd.)			
9. Power/number of pumps instrument channels	2	1	Maintain hot shutdown
10. High reactor building pressure channels	2	1	Maintain hot shutdown

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, or 2 of 4 power range instrument channels are greater than 10 percent full power, hot shutdown is not required.

B. Other Reactor Trips			
1. Loss of Feedwater	2(a)(c)	1(a)(c)	Maintain less than 7% indicated reactor power.
2. Turbine Trip	2(b)(c)	1(b)(c)	Maintain less than 45% indicated reactor power.

(a) Bypass of the feedwater pump trip signal may be placed in effect when indicated reactor power is less than 7%. The bypass will be removed when reactor power is raised above 7%.

(b) The main turbine trip bypass may be placed in effect when indicated reactor power is less than 45%. The bypass will be removed when reactor power is raised above 45%.

(c) Trip may be defeated during low power physics tests.

Table 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
38.	OTSG Full Range Level	W	NA	R	
39.	Turbine Overspeed Trip	NA	R	NA	
40.	BWST/NaOH Differential Pressure Indicator	NA	NA	R	
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R	
42.	Diesel Generator Protective Relaying	NA	NA	R	
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)				
	a. Degraded Grid	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons
	b. Loss of Voltage	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Reactor Trip	S(1)	M(1)	R	(1) When reactor power exceeds 7% power
46.	Turbine Trip/Reactor Trip	S(1)	M(1)	R	(1) When reactor power exceeds 45% power.
47.	a. Pressurizer Code Safety Valve and PORV Tailpipe Flow Monitors	S(1)	NA	R	(1) When T_{ave} is greater than 525°F
	b. PORV - Acoustic/Flow	NA	M(1)	R	(1) When T_{ave} is greater than 525°F
48.	PORV Setpoints	NA	M(1)	R	(1) Per Specification 3.1.12 excluding valve operation.

ENCLOSURE 1

EG&G IDAHO REVIEW
OF TECHNICAL SPECIFICATION CHANGE REQUEST NO. 154
FOR THE GPU NUCLEAR CORPORATION PLANT
THREE-MILE ISLAND UNIT 1 DOCKET NO. 50-289
OPERATING LICENSE NUMBER DPR-50

1.0 INTRODUCTION

In a letter from H. D. Hukill, GPU Nuclear Corporation (GPU), to the U.S. Nuclear Regulatory Commission (NRC) dated March 5, 1987 (Reference 1), GPU proposed license amendments to Facility Operating License DPR-50 for Three Mile Island Unit 1 (TMI-1). The proposed amendment would: (1) increase the setpoint for trip (i.e., shutdown) of the reactor on high pressure in the reactor coolant system from 2300 psig to 2355 psig; and (2) increase the arming threshold for anticipatory reactor trip (ART) on turbine trip from 20% of full power to 45% of full power. In response to the NRC request, GPU provided additional supporting information by letter, H. D. Hukill to the NRC dated August 4, 1987 (Reference 2). This report presents the EG&G Idaho review of the proposed amendment and the supporting information.

2.0 BACKGROUND

The Babcock and Wilcox (B&W) Nuclear Steam Supply System (NSSS) was designed with the capability to adjust to minor plant upsets and certain anticipated events such as feedwater transients, rapid load changes and turbine trips without a reactor trip. The system was designed to initiate a plant runback, upon detection of an upset or equipment malfunction, to a power level consistent with the plant condition and to limit the rise in Reactor Coolant System (RCS) pressure to less than the reactor trip setpoint by opening the

pressurizer Power Operated Relief Valve (PORV). Subsequent to the TMI-2 accident, the Nuclear Regulatory Commission (NRC), by IE Bulletin 79-05B (Reference 3), required licensees for all B&W Pressurized Water Reactor (PWR) facilities to make modifications to reduce the number of automatic actuation of the PORVs. The modifications proposed by B&W on behalf of the owners group and accepted by the NRC included (1) raising the PORV setpoint from 2255 psig to 2450 psig, (2) decreasing the reactor trip on high RCS pressure from 2390 psig to 2300 psig and (3) providing an ART for turbine trips above 20% power. In addition, the NRC required that B&W demonstrate that these modifications: (1) limited the frequency of PORV openings to less than 5% of the total number of overpressure transients (NUREG 0737 Item II.K.3.7 Reference 4) and (2) limited the probability of a small-break Loss of Coolant Accident (LOCA) caused by a stuck-open PORV to less than .001 per reactor-year (NUREG 0737 Item II.K.3.2 Reference 4). B&W submitted a report (Reference 5) to demonstrate that the modifications did meet the requirements. The NRC issued a Safety Evaluation Report (Reference 6) which concluded that the requirements were met.

Although these modifications have met the objectives of reducing challenges to and opening of the PORV, they have increased the frequency of reactor trips. Each reactor trip results in a challenge to plant safety systems and any reduction in reactor trip frequency will contribute to overall plant safety as well as plant availability. On behalf of the B&W Owners Group, B&W has submitted two reports: (1) "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, September 1985 (Reference 7) and (2) "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985 (Reference 8). These reports present the B&W justification for raising the high pressure reactor trip setpoint from 2300 psig to 2355 psig and increasing the ART for turbine trips from 20% power to 45% power. These reports state that with these changes, the NRC requirements for limiting the frequency of PORV openings and limiting the probability of a small-break LOCA due to a stuck open PORV will still be met. Further, these reports state that with these changes, the number of reactor trips per reactor year will be reduced by approximately 10%.

In April 1986, the NRC staff completed its review of these two B&W reports. In their Safety Evaluation Reports (SERs), the staff: (1) reviewed the basis for the proposed changes; (2) reviewed B&W's method of analysis of the effect of the proposed high pressure trip setpoint on PORV openings; (3) compared the results of Monte Carlo simulation for PORV openings with the NRC requirements contained in NUREG-0737 (Reference 4); and (4) reviewed the results of B&W's analysis of the arming threshold for ART. The NRC requirements include: (1) the PORV will open in less than 5% of all anticipated overpressure transients (NUREG 0737, Item II.K.3.7 Reference 4); and (2) the probability of a small-break loss of coolant accident caused by a stuck-open PORV will be less than 0.001 per reactor-year (NUREG 0737, Item II.K.3.2 Reference 4). In the SERs (References 9 and 10), the staff concluded on a generic basis that the proposed changes met the NRC requirements, and should benefit plants by potentially reducing the reactor trip frequency. Accordingly, the NRC concluded that the B&W reports could be referenced in licensing submittals by the B&W Owners Group members.

3.0 EVALUATION

The licensee is a member of the B&W Owners Group and in their March 5, 1987 proposal (Reference 1), referenced the B&W reports BAW-1890 (Reference 7) and BAW-1893 (Reference 8) as justification for raising the high pressure reactor trip setpoint from 2350 psig to 2355 psig and increasing the ART for turbine trip to 45% power. The licensee states these reports are applicable to TMI-1 because: (1) TMI-1 is a B&W 177FA plant type for which the reports apply, (2) the mix of transients used to develop the historical data and the pressure overshoot distribution included data from TMI-1 for pre-TMI and similar B&W plants for post-TMI, (3) the actual measured high pressure reactor trip setpoint of 2294 psig (nomnally set at 2300 psig) and the maximum expected string error at the pressure bistable of ±3.52 psi for TMI-1 are within the range of the plants used to develop the statistical data and (4) the available steam by-pass, including the first bank of main steam safety valves, is 76.9% of full steam flow which is conservatively higher than the 43% used in the analysis. Based on the information provided and a review of the B&W reports, EG&G concluded that data and parameters used in the B&W reports are a conservative bound for TMI-1 and the use of these reports for TMI-1 is

acceptable.

The important conclusion of the B&W reports which justifies raising the high pressure trip and the ART for turbine trip is that even though both changes will result in a small increase in the probability of opening the PORV, the increase is insignificant compared to the total openings of the PORV from all events. The estimate made in the B&W reports (References 7 and 8) for PORV openings from high pressure events with the reactor high pressure trip set at 2355 psig and the ART for turbine trip set at 45% is 1.96×10^{-5} PORV openings per reactor year. The estimate made in the B&W report (Reference 6) for PORV openings from all events is 8.06×10^{-2} PORV openings per reactor year with the major contributor to the PORV openings identified as actions by the operators in carrying out plant operation in accordance with the Abnormal Transient Operating Guidelines (ATOG). The 1.96×10^{-5} PORV openings per reactor year from high pressure events applies to TMI-1 as discussed in the preceeding paragraph. The 8.06×10^{-2} PORV openings per reactor year from all events applies to TMI-1 because the TMI-1 plant specific Abnormal Transient Procedures (ATPs) are based on the ATOG. Therefore, for TMI-1, the PORV openings from high pressure events with the proposed setpoints are insignificant compared with the openings from all events.

GPU states (Reference 1) that the proposed changes are within the bounds of the current TMI-1 accident analysis of Chapter 14 of the Final Safety Analysis Report (FSAR). These analyses use the initial high pressure trip of 2390 psig for evaluating high pressure events, which is a conservative bound to the proposed 2355 psi. The licensee also states (Reference 2) that the FSAR analyses do not take credit for the ART on turbine trip and are therefore still valid and a conservative bound for the proposed ART on turbine trip of 45% power.

The licensee in their letter of August 4, 1987 (Reference 2) confirmed that with the proposed setpoint changes more Main Steam Safety Valves (MSSVs) would open during an overpressure transient or a turbine trip compared to those opening with the current settings. However, they contend that with the proposed changes there would be fewer overpressure transients and turbine trips

that challenge the MSSVs and the net number of openings would be expected to be reduced. Also, the MSSVs have a good record of performance. Data from TMI-1 and similar B&W plants show the MSSVs have lifted many times without failures to reseat. Based on the information provided by the licensee and the information from a similar plant, EG&G concurs that the anticipated openings of the MSSVs are acceptable.

3.0 CONCLUSIONS

EG&G has reviewed the proposed changes to TS 2.2 and Table 3.5-1 for TMI-1. The proposed changes would (1) increase the setpoint for trip of the reactor on high pressure in the reactor coolant system from 2300 psig to 2355 psig; and (2) increase the arming threshold for ART on turbine trip from 20% of full power to 45% of full power. As discussed in the preceding section, the EG&G finds the proposed changes meet the NRC positions established in their review of the B&W topical reports and therefore meet the applicable regulatory guidance and requirements and are, therefore, acceptable.

REFERENCES

1. Letter from H. D. Hukill, GPU Corporation, to U.S. Nuclear Regulatory Commission, Letter number 5211-87-2053 dated March 5, 1987.
2. Letter from H. D. Hukill, GPU Corporation, to U.S. Nuclear Regulatory Commission, Letter number 5211-87-2142 dated August 4, 1987.
3. IE Bulletin 79-05B," Nuclear-Incident at Three Mile Island - Supplement". April 2, 1979.
4. NUREG-0737, "Clarification of TMI Action Plan Requirements". November 1980.
5. Babcock & Wilcox Report 12-1122779 Rev. 1, "Report on PORV Opening Probability and Justification for Present Systems and Setpoints". January 1981.
6. NRC Memorandum from F. H. Rowsome to G. C. Laines, "Safety Evaluation of the B&W Licensees' Responses to TMI Action II.K.3.2". August 24, 1983.
7. BAW-1890, "Justification for Raising Setpoint for Reactor Trip on High Pressure." September 1985.

8. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip". October 1985.
9. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, B&W, Acceptance for Referencing of Licensing Topical Report BAW-1890, "Justification for Raising Setpoint for Reactor Trip on High Pressure". April 27, 1986.
10. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, B&W, Acceptance for Referencing of Licensing Topical Report BAW-1893 "Basis for Raising Threshold for Anticipatory Reactor Trip on Turbine Trip". April 25, 1986.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 135 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letter dated March 5, 1987 (Ref. 1), General Public Utilities (GPU) Nuclear Corporation (the licensee) made application to amend the license of the Three Mile Island Unit 1 (TMI-1) plant. The proposed amendment would raise the reactor protection system (RPS) high reactor coolant system (RCS) pressure trip setpoint from 2,300 psig to 2,355 psig. Also, the arming threshold for the Anticipatory Reactor Trip (ART) on turbine trip would be raised from the current 20% reactor power level to a level of 45%. These proposed changes were to improve the operational performance of the plant.

The licensee, in providing justification for the requested changes, referred to studies made in 1985 by the Babcock and Wilcox Owners Group (B&WOG). As part of the Transient Assessment Program the B&WOG had studies performed to improve operational safety and performance through a reduction in the frequency of reactor trips. This effort resulted in two reports submitted for review to the NRC. These reports are:

(1) "Justification for Raising Setpoint for Reactor Trip on Pressure," BAW-1890, September 1985 (Ref. 2).

(2) "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985 (Ref. 3).

The staff accepted the two above listed reports by evaluations issued on April 22, 1986 (Ref. 4) and April 25, 1986 (Ref. 5), respectively.

2.0 EVALUATION

The staff was assisted in the evaluation of the TMI-1 Technical Specification changes by their consultants at EG&G. The EG&G evaluation included a review of the GPU Nuclear Corporation request for changes (Ref. 1) and comparison with

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the B&WOG reports (Refs. 2 and 3) which were approved by the staff (Refs. 4 and 5). Additional supporting information was requested by the staff. The evaluation of the above information was prepared by EG&G and is presented in Enclosure 1. The staff agrees with the EG&G evaluation which concludes that adequate justification has been provided to verify that NRC requirements for the number of PORV openings and the probability of a small-break loss of coolant from PORV failure have been met. Also, it was found that the accidents analyzed in the FSAR for TMI-1 bound the proposed Technical Specification changes. The staff therefore concludes that the RCS high pressure trip setpoint can be increased from 2,300 psig to 2,355 psig and that the arming threshold for anticipatory reactor trip on turbine trip can be raised from 20% reactor power to 45% reactor power without adverse effect on safety.

3.0 TECHNICAL SPECIFICATION CHANGES

The Technical Specification changes for TMI-1 are as follows:

1. TS 2.2 Bases, Page 2-4. Safety Limits - Reactor System Pressure. The value of the setting for the reactor high pressure trip is changed from 2,300 psig to 2,355 psig as found acceptable in Section 2 above. Also the reference to the setting of the pressurizer electromatic relief valve is deleted as no credit is taken for it in the safety analysis.

2. TS 2.2 Bases, Page 2-7. Reactor Coolant System Pressure. Parts of two paragraphs were modified for better grammatical and historical background information. Also a new paragraph was added to refer to the B&W analysis made in September 1985 (BAW-1890) which justified raising the setpoint for reactor trip on high pressure. This is acceptable.

3. TS 2.2, Page 2-8. The B&W report "Justification for Raising Setpoint for Reactor Trip in High Pressure," BAW-1890, Rev. 0, Babcock and Wilcox, September 1985, is added as a reference.

4. Table of Reactor Protection System Trip Setting Limits, Page 2-9. The value for the high reactor coolant system pressure was changed from 2,300 psig to 2,355 psig and is acceptable as explained in Section 2 above.

5. Figure 2.3-1 - Protection System Maximum Allowable Setpoints. The reactor coolant pressure was raised from 2,300 psig to 2,355 psig. This is acceptable as explained in Section 2 above.

6. TS 2.2, Page 3-28. The value shown for the arming threshold for the anticipatory reactor trip (ART) was changed from 20% reactor power to 45% reactor power. This is acceptable as explained in Section 2 above. Also, a reference to B&W report, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, Rev. 0, Babcock and Wilcox, October 1985, was added.

7. Table 3.5-1, Page 3-30. Instruments Operating Conditions. The value shown for the arming threshold for the ART was changed from 20% reactor power to 45% reactor power. This is acceptable as explained in Section 2 above.

8. Table 4.1-1, Page 4-7. The value shown for the arming threshold for the ART was changed from 20% reactor power to 45% reactor power. This is acceptable as explained in Section 2 above.

The staff with assistance from EG&G has reviewed the proposed changes to TS 2.2 and Table 3.5-1 for TMI-1. The proposed changes would (1) increase the setpoint for trip of the reactor on high pressure in the reactor coolant system from 2,300 psig to 2,355 psig; and (2) increase the arming threshold for ART on turbine trip from 20% of full power to 45% of full power. As discussed in Section 2, the proposed changes meet the NRC positions established in the review of the B&W topical reports and therefore meet the applicable regulatory guidance and requirements and are, therefore, acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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 this amendment.

6.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from H. D. Hukill, GPU Nuclear Corporation, to U.S. Nuclear Regulatory Corporation, dated March 5, 1987.

2. "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, September 1985.

3. "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985.

4. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, Babcock and Wilcox Company, April 22, 1986.

5. Letter from D. M. Crutchfield, NRC, to J. H. Taylor, Babcock and Wilcox Company, April 25, 1986.

Enclosure:
EG&G Evaluation

Dated: December 1, 1987

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