FEB 28 1983

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Mr. Henry D. Hukill
Vice President
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

The Commission has issued the enclosed Amendment No.81 to Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your request dated November 10, 1981.

The amendment adds TS 3.20 concerning Special Test Exceptions which permits suspension of certain TSs during performance of the Low Power Natural Circulation Test to be conducted during the restart of TMI-1, if it is authorized. We have incorporated certain revisions to the TS pages provided in your November 10, 1981, submittal. These revisions were agreed to by C. Stephenson of your licensing staff per phone call on January 12, 1983. Issuance of this amendment, together with our letter dated December 21, 1982, finding your restart testing program acceptable, completes the Office of Nuclear Reactor Regulation review of your response to NUREG-0737, Item I.G.1.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

"ORIGINAL SIGNED BY:"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No.81
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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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. 81
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensees), dated November 10, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 81, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: FEB 28 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 81

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following revised page and add the new pages of the Appendix "A" Technical Specifications with the enclosed pages. These pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iii	iii
--	3-95a (new)
--	3-95b (new)
--	3-95c (new)

TABLE OF CONTENTS

3.20 SPECIAL TEST EXCEPTIONS

3.20.1 Low Power Natural Circulation Test (LPNCT)

Applicability

During the performance of the Low Power Natural Circulation Test for Cycle 5 Restart. This Technical Specification is cancelled following the completion of this test.

Objective

To provide meaningful technical information concerning natural circulation at TMI-1 and enhance operator training under both normal and certain degraded conditions.

Specification:

3.20.1.1 The limitations of Specifications 3.1.3.1 and 3.1.3.3 may be suspended during the LPNCT - Determination of Indicated Reactor Power Correction Factor provided:

a. The RPS overpower trip is less than or equal to 7% full power.

b. With the RCS temperature (T (cold)) below 525°F, continuous visual monitoring will be initiated within 15 minutes. A manual trip will be initiated if the RCS temperature (T (cold)) drops below 520°F.

3.20.1.2 The limitations of Specifications 2.1.2, 2.3.1, 3.5.1.1, 3.5.1.3 and 3.1.1.1 a. and b., may be suspended during the LPNCT - Establishment of Natural Circulation Flow/Determination of the Effect of Decreased OTSG Levels on Natural Circulation Flow. Suspension of Specification 2.3.1 is limited to Items 1 (max. power), 2 (flux/imbalance/flow), and 3 (pump power) on Table 2.3-1. Suspension of Specifications 3.5.1.1 and 3.5.1.3 is limited only to bypassing the automatic trip functions of the following instrumentation in Table 3.5-1: Item A.7 (flux/imbalance/flow instruments), Item A.9 (pump power instruments) and Items B.1 and B.2 (other reactor trips). Suspension of these specifications is permitted provided:

a. The control rod index shall be maintained within the limits required for 100% FP operation specified in T.S. Figure 3.5-2A.

b. The subcooling margin is greater than or equal to 50°F.

c. The RPS Overpower trip is less than or equal to 7% full power.

d. The average of the five highest core exit thermocouple temperatures is less than or equal to 610°F.

e. T hot is maintained less than or equal to 600°F.

With the reactor coolant system outside the limits of any of the limits of a. through e., a manual trip of the reactor shall be initiated.

3.20.1.3 The limitations of Specifications 2.1.1, 2.1.2, 2.3.1, 3.1.1.1 a. & b., 3.5.1.1 and 3.5.1.3 may be suspended during the LPNCT - Verification of the adequacy of the Pressurizer Heaters on the Emergency Bus. Suspension of Specification 2.3.1 applies to all items on Table 2.3-1 except Item 4 (High RCS pressure), Item 7 (RCS temperature, max.) and Item 8 (High RB pressure). Suspension of Specifications 3.5.1.1 and 3.5.1.3 is limited only to bypassing the automatic trip functions of the following instrumentation in Table 3.5-1: Item A.6 (pressure-temperature instrument), Item A.7 (flux/imbalance/flow instrument), Item A.9 (pump power instrument) and Items B.1 and B.2 (other reactor trips). Suspension of these specifications is permitted provided:

a. **The control rod index shall be maintained within the limits required for 100% full power operation specified in T.S. Figure 3.5-2A.**

b. **The subcooling margin is greater than or equal to 20° F.**

c. **The RPS Overpower trip is less than or equal to 7% full power.**

d. The average of the five highest core exit thermocouple temperatures is less than or equal to 610° F.

e. RCS pressure is between 1700 psig and 2300 psig.

f. The low RCS pressure trip is greater than or equal to 1700 psig.

g. T_{hot} is maintained less than or equal to 600°F.

With the reactor coolant system outside the limits of any of the above limits of a. - g., a manual trip to the reactor shall be initiated.

Bases

During the performance of the special tests, the combination of administrative limits on reactor power $\leq$5% FP and subcooling margin $\geq$50°F, RPS overpower trip limit $\leq$7% FP and manual trip limits of core exit thermocouple temperature $\leq$610°F, loop T (hot) $\leq$600°F, loop Tave $\geq$525°F, and subcooling margin $\geq$20°F will insure that the integrity of the fuel cladding will be maintained.

During the performance of the special tests, removal of forced flow will be initiated at a reactor thermal power level of approximately 3% FP with an administrative control limit of 5% FP and RPS overpower trip setting of 7% FP. Table 14-13 of the Three Mile Island, Unit 1 FSAR provides calculated values that indicate that the expected natural circulation flow rates at less than or equal to 5% FP will be in excess of flow rates required for the removal of core decay heat without formation of voids in the core or reactor outlet piping.

The power imbalance portion of overpower trip based on reactor coolant flow and reactor power imbalance protects the core from center-line fuel melt by limiting the maximum linear heat rate (Kw/ft) in the fuel at high reactor thermal power levels. The overpower trip setting at 7% FP will prevent exceeding the Kw/ft limit regardless of the core imbalance.

The low pressure and variable low pressure trip setpoints have been established to maintain the DNB ratio $\geq$ 1.3 for those design accidents that result in RCS pressure reductions. To prevent an inadvertent reactor trip during the performance of 3.20.1.3, the variable low pressure trip will be bypassed.

However, constant operator surveillance of reactor power level, reactor coolant system temperatures and pressure, and subcooling margin and operator action will be relied on to provide the protection function normally furnished by this RPS trip. This will be accomplished by manual reactor trip if any of the limits specified are reached.

The low pressure trip prevents operation at pressures which might reduce DNBR margin and provides for reactor trip prior to ESAS actuation on low RCS pressure. In order to retain some automatic low pressure protection and still provide operation flexibility and prevent an inadvertent reactor trip during the performance of 3.20.1.3 only, the low pressure trip setpoint will be lowered to 1700 psig.

Based on TMI-1, Cycle 5 Physics Test Manual predictions, it is expected that a slightly negative moderator temperature coefficient will exist for the conditions of the low power natural circulation testing. This prediction will be verified based on the hot zero power moderator temperature coefficient measured during the zero power physics testing program prior to power escalation.

This technical specification delineates the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety. Technical Specification 3.5.1.1 states that the reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5-1, columns A and B are met. Parameter indications from these specified instrument channels will be available at all times but the trip functions for flux/imbalance/flow, power/number of pumps, and pressure/temperature (variable low pressure) will be defeated during previously specified tests.

Technical Specification 3.5.1.3 states that if the number of protection channels operable is less than the limit given in Table 3.5-1, column A, operation shall be limited as specified in Column C. In the cases of the above three protection channels defeated, operation would be limited to Hot Shutdown.

It should be noted that during natural circulation testing, the sensitivity of the reactor power (power range) indicator will be increased such that the actual power is 10 times less than the "indicated" reactor power (i.e. 50% power read on the meter equals 5% power). This will necessitate bypassing the anticipatory trips for loss of turbine (<20% indicated reactor power) and loss of both main feed pumps ($\leq$ 7% indicated reactor power).

Technical Specification Section 3.1.9 "Low Power Physics Testing Restrictions" does not apply during this test.

References

(1) FSAR Chapter 14

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 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 NO. 1

DOCKET NO. 50-289

Introduction

By letter dated November 10, 1981 (L1L 314, TSCR No. 108), GPU Nuclear Corporation (the licensee) requested a change to the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1) Technical Specifications (TSs). This change would add additional TSs to permit performance of a Low Power Natural Circulation Test following the restart of TMI-1. The licensee committed to perform this test in response to NUREG-0694, Item I.G.1, which was imposed by the NRC staff on TMI-1 by letter dated November 25, 1980, and reaffirmed by the Atomic Safety and Licensing Board (ASLB) in its December 14, 1981, Partial Initial Decision (PID) in the TMI-1 restart proceeding (paragraph 1132).

Discussion and Evaluation

As part of the low power testing program for the restart of TMI-1, the licensee proposes to operate the reactor outside the range of the current plant TSs and requests that the TSs be modified to encompass the conditions of the proposed tests. The tests are designed to demonstrate the ability of the primary and secondary system to remove energy from the core at decay heat power levels and to provide operator training. Since the decay heat generation rate at TMI-1 is very low because of the long shutdown period, the reactor will be brought critical and operated at a power level of approximately 3% of design (2535 MWT) to simulate decay heat. This modification to the TSs will be cancelled at the completion of the test program.

Six tests are proposed which are briefly summarized below:

1. Determination of indicated reactor power conversion factor. This test will be performed with the reactor coolant pumps in operation. The average reactor system temperature will be reduced to 526°F. The current TSs require that the reactor be subcritical if the reactor coolant temperature is below 525°F. Although the licensee does not believe that this limit will be violated, a reduction of the limit to 520°F is requested to reduce the potential for a TS violation.

2. Emergency feedwater actuation test. This test is designed to demonstrate that following a loss of main feedwater event, the emergency feedwater pumps will start automatically and emergency feedwater flow will be properly indicated within the control room. The reactor coolant pumps will remain in operation. No TS changes are required for this test.

3. Verification of two-hour air supply capability of the bottled backup air supply. In this test the ability of the backup air supply to control air operated emergency feedwater system valves will be verified. The reactor coolant pumps will remain in operation. No TS changes are required for this test.

4. Establishment of Natural Circulation Flow. This test will verify that the reactor can be controlled manually in natural circulation. For this test, the reactor trip logic will be modified so that the reactor will not trip on loss of forced coolant flow. TS changes are required to permit bypassing this trip function.

5. Verification of the adequacy of the pressurizer heaters on the emergency bus. This test will verify the adequacy of the pressurizer heaters to maintain natural circulation at hot standby with only emergency power available. The heaters will then be de-energized and the effect on natural circulation will be monitored as the reactor system decreases to 1800 psig. At that pressure, the test will be terminated. The reactor trip logic will be modified so that the reactor may be operated at low power without forced reactor coolant flow and so that the reactor will not trip on low pressure. TS changes are required to permit bypassing the automatic trip functions associated with loss of forced flow or low pressure.

6. Determination of the effect of decreased once-through steam generator levels on natural circulation flow. In this test, the secondary side water level will be slowly decreased to 90 inches on the startup range and the effect on natural circulation recorded. If natural circulation is lost, the secondary level will be increased so that natural circulation is reestablished. The reactor trip logic will be modified so that the reactor can be operated at low power without forced reactor coolant flow. TS changes are required to permit bypassing this trip function.

The low power tests proposed at TMI-1 are designed to be performed at power levels simulating decay heat. Since the amount of decay heat at TMI-1 is very low, the reactor will be operated at approximately 3% of full power. Administrative controls will require that the reactor be manually tripped at 5%, of full power.

For the natural circulation tests, the licensee will lower the power level trip setpoint to 7% of full power, and additional administrative controls will be placed on reactor system pressure and temperature. A 7% power level compares to the decay heat level immediately after reactor trip from full power operation. Similarily, a 5% power level would be present after 15 seconds and a power level of 3% would be present after 200 seconds following a reactor trip from full power. Analyses in

Chapter 14 of the TMI-1 Final Safety Analysis Report (FSAR) indicate that forced reactor coolant flow is not required at these power levels to prevent fuel damage. The analyses further demonstrate that with the steam generator water level at the design value of 50% on the operating range, natural circulation will be adequate to transfer the heat generated by the core to the steam generators.

In Test Number 6, the steam generator water level will gradually be lowered to determine the effect on natural circulation. During this test, natural circulation may be lost. Loss of natural circulation will be indicated to the operator by increasing primary system temperature and pressure and by reduced steam and feedwater flowrate in the steam generators. The operator will increase steam generator level if natural circulation is lost and will trip the reactor if the subcooled margin is lost. Tripping the reactor will reduce the reactor heat generation rate to a very low level since little fission product inventory will be built up in the core. Even in the event that the reactor system became saturated and boiling occurred in the core, fuel damage would not occur at the test power levels. This is demonstrated by analyses of small break Loss of Coolant Accidents which show that following a reactor trip from full power, the core will be adequately cooled as long as it remains covered with water or a two-phase mixture[1]. We have evaluated potential design basis transients and accidents and believe that with the additional requirement discussed below, the consequences of any design basis event at the low power levels of the proposed tests will be less than the consequences of transients and accidents analyzed in Chapter 14 of the FSAR. For the natural circulation tests (Tests 4, 5 and 6), reduction of the overpower setpoint to 7% will provide additional protection for transients and accidents which produce an increase in power level, such as inadvertent control rod withdrawal.

For Test Number 1, the reactor coolant temperature will be lowered below the values evaluated in the FSAR. The licensee has not evaluated the effect of increased system stiffness produced by reduced temperature for a potential control rod withdrawal event. We, therefore, require that the reactor overpower setpoint be set at 7% for this test as well as for Tests 4, 5 and 6. The licensee has agreed to implement this additional requirement and TS 3.20.1.1 has been revised from the licensee's proposed TS to reflect this.

Since the licensee has agreed to establish the reactor overpower trip setpoint at 7% for Test Number 1, the reduced trip setpoint will be available for all tests which require modification of the current TSs. This automatic protection, as well as constant operator surveillance combined with administrative controls on power level, temperature, pressure

[1] Letter from J. Taylor, B&W, to S. Varga, NRC, attaching additional Emergency Core Cooling System small break analyses, July 18, 1978.

and subcooling margin set forth in the test procedures will provide adequate protection for the reactor core and reactor system pressure boundary during the performance of these low power tests. On this basis, we find the proposed change to the TSs acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: FEB 28 1983

The following NRC personnel have contributed to this Safety Evaluation: W. Jensen, R. Jacobs.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-289

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 81 to Facility Operating License No. DPR-50, issued to Metropolitan Edison Company, Jersey Central Power and Light Company, Pennsylvania Electric Company, and GPU Nuclear Corporation (the licensees), which revised the Technical Specifications (TSs) for operation of the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1) located in Dauphin County, Pennsylvania. The amendment is effective as of its date of issuance.

The amendment adds TS 3.20 concerning Special Test Exceptions which permits suspension of certain TSs during performance of the Low Power Natural Circulation Test to be conducted during the restart, if authorized, of TMI-1.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 10, 1981, (2) Amendment No. 81 to License No. DPR-50, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555, and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 28th day of February 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing