August 14, 1986

DMB 016

Docket No. 50-289

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NThompson
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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. 119 TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued the enclosed Amendment No. 119 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated March 8, 1985, as supplemented May 14, 1985 and April 24, 1986. This amendment revises the TSs to provide decay heat removal capability in all modes of operation as requested in the NRC staff's generic letter dated June 11, 1980. This action completes Multiplant Action (MPA) B-57 for TMI-1.

Relative to our concerns about operation under conditions equivalent to Standard TS (STS) Modes 3 and 4 as expressed in our letter dated January 9, 1986, we agree that your letter dated April 24, 1986, provides sufficient justification to grant these Technical Specification changes as stated in our enclosed Safety Evaluation (SE). However, the broader question of whether your equivalent to STS Mode 3 and 4 operations is bounded by FSAR analyses will continue to be pursued as part of a generic effort by the NRC staff on this matter. Generic Letter 86-13, Potential Inconsistency Between Plant Safety Analyses and Technical Specifications, issued July 23, 1986 represents a separate follow-up action on this subject.

Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

"ORIGINAL SIGNED BY"
John O. Thoma, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 119 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

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*See previous white for concurrences.

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WPaulson*
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RIngram*	JThoma:jak*	RWeller*	GSchwenk*	TMarsh*	CThomas*	DCrutchfield*	JStolz*	MWagner*
8/13/86	8/13/86	7/25/86	7/9/86	7/9/86	7/10/86	7/14/86	7/28/86	7/31/86

Docket No. 50-289

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PWR Project Directorate #6
Division of PWR Licensing-B

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See next page

*See previous white for concurrences.

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WPaulson*
7/14/86

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Docket No. 50-289

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and Director - TMI-1
GPU Nuclear Corporation
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Sincerely,

John O. Thoma, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

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1. Amendment No. to DPR-50
2. Safety Evaluation

PBD#6
WPaulson
WP 7/14/86

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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Atomic Safety & Licensing Board Panel
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Atomic Safety & Licensing Appeal
Board Panel (8)
U.S. Nuclear Regulatory Commission
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UNITED STATES
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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. 119
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 March 8, 1985, as supplemented May 14, 1985, and April 24, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 119, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: August 14, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 119

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
ii	ii
iv	iv
3-25	3-25
3-26	3-26
3-26a	3-26a
--	3-26b
--	3-26c
4-52	4-52
4-52a	4-52a

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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. Specification 3.0.1 applies.

b. 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. With more than one EFW pump or flow path* inoperable, restore the inoperable pumps or flow paths* 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.

c. Four of six turbine bypass valves OPERABLE.

d. The condensate storage tanks (CST) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. 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. 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. Specification 3.0.1 applies.

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*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator.

**These requirements supplement the requirements of Sections 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.

3.4.1.2 With the Reactor Coolant System temperature greater than 250°F, all eighteen (18) main steam safety valves shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.4.2 Reactor Coolant System temperature 250°F or less.

3.4.2.1 With Reactor Coolant temperature 250°F or less, at least two of the following means for maintaining decay heat removal capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. Decay Heat Removal String "A".

b. Decay Heat Removal String "B".

c. Reactor Coolant Loop "A", its associated OTSG, and its associated emergency feedwater flowpath.

d. Reactor Coolant Loop "B", its associated OTSG, and its associated emergency feedwater flowpath.

3.4.2.2 Operation of the means for decay heat removal may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for decay heat removal required to be operable per 3.4.2.1 may be reduced to one provided that one of the following conditions is satisfied:

a. The Reactor is in a Refueling Shutdown condition with the Fuel Transfer Canal water level greater than 23 feet above the reactor vessel flange.

b. Reactor coolant temperature is less than 140°F with BWST level greater than 44 feet and an associated flow path through the RCS OPERABLE such that core outlet temperature can be maintained subcooled for at least 7 days.

c. Equipment Maintenance on one of the means for decay heat removal specified by 3.4.2.1 is required and the equipment outage does not exceed 7 days.

3.4.2.4 Specification 3.4.2.1 does not apply when either of the following conditions exist:

a. Decay heat generation is less than 188 KW with the RCS full.

b. Decay heat generation is less than 100 KW with the RCS drained down for maintenance.

3.4.2.5 With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when RCS temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If Main Steam Safety Valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.1.2 such that the remaining safety valves can prevent overpressure on a turbine trip.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust. (1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.1.1, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

When the RCS is below 250°F, a single DHR string, or single OTSG and its associated emergency feedwater flowpath is sufficient to provide removal of decay heat at all times following the cooldown to 250°F. The requirement to maintain two OPERABLE means of decay heat removal ensures that a single failure does not result in a complete loss of decay heat removal capability. The requirement to keep a system in operation as necessary to maintain the system subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

Limited reduction in redundancy is allowed for preventive or corrective maintenance on the primary means for decay heat removal to ensure that maintenance necessary to assure the continued reliability of the systems may be accomplished.

As decay heat loads are reduced through decay time or fuel off loading, alternate flow paths will provide adequate cooling for a time sufficient to take compensatory action if the normal means of heat removal is lost.

With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. The BWST with level at 44 feet provides an equivalent reservoir available as a heat sink. Operability of the BWST is to be determined using calculations based on actual plant data or through plant testing at the time the system is to be declared operable. At such times that either of these means is determined to be operable, removal of the redundant or diverse cooling system is permitted.

Following extensive outages or major core off loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capability is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident.

REFERENCES

(1) FSAR Section 10.2.1.3.

4.9 DECAY HEAT REMOVAL CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for decay heat removal are capable of performing their design function.

Specification

4.9.1 Emergency Feedwater System - Periodic Testing (Reactor Coolant System Temperature greater than 250°F.)

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the EFW pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3210. The test frequency shall be at least every 31 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 During testing of the EFW System when the reactor is in STARTUP, HOT STANDBY or POWER OPERATION, if one steam generator flow path* is made inoperable, a dedicated qualified individual who is in communication with the control room shall be continuously stationed at the EFW local manual valves (See Table 4.9-1). On instruction from the Control Room Operator, the individual shall realign the valves from the test mode to their operational alignment.

4.9.1.3 At least once per 31 days each valve listed in Table 4.9-1 shall be verified to be in the status specified in Table 4.9-1, when required to be OPERABLE.

4.9.1.4 On a quarterly basis, verify that the manual control (HIC-849/850) valve station functions properly.

4.9.1.5 On a quarterly basis, EFV-30A and B shall be checked for proper operation by cycling each valve over its full stroke.

4.9.1.6 Prior to start-up, following a refueling shutdown or a cold shutdown greater than 30 days, conduct a test to demonstrate that the motor driven EFW pumps can pump water from the condensate tanks to the Steam Generators.

*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a Steam Generator.

4.9.1.7 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 Decay Heat Removal Capability - Periodic Testing (Reactor Coolant System Temperature 250°F or less).*

4.9.2.1 On a daily basis, verify operability of the means for decay heat removal required by specification 3.4.2 by observation of console status indication.

*These requirements supplement the requirements of 4.5.2.2 and 4.5.4.

Bases

The 31-day testing frequency will be sufficient to verify that the turbine driven and two motor-driven EFW pumps are operable and that the associated valves are in the correct alignment. ASME Section XI, Article IWP-3210 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal acceptance criteria assures that the EFW pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. In the case of the EFW System flow, the flow shall be considered acceptable if under the worst case single pump failure, a minimum of 500 gpm can be delivered when steam generator pressure is 1050 psig and one steam generator is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient flow can be delivered to either Steam Generator. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Daily verification of the operability of the required means for decay heat removal ensures that sufficient decay heat removal capability will be maintained.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 119 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

A number of events have occurred at operating Pressurized Water Reactor (PWR) facilities where decay heat removal capability has been seriously degraded due to inadequate administrative controls during shutdown modes of operation. One of these events, described in IE Information Notice 80-20 dated May 8, 1980, occurred at the Davis-Besse Station, Unit No. 1, on April 19, 1980. In IE Bulletin 80-22 issued May 9, 1980, licensees were requested to immediately implement administrative controls which would ensure that proper means are available to provide redundant methods of decay heat removal. While the function of the bulletin was to effect immediate action with regard to this problem, the NRC considered it necessary that an amendment be made to each PWR license to provide for permanent long-term assurance that redundancy in decay heat removal capability will be maintained. By generic letter dated June 11, 1980, all PWR licensees were requested to propose Technical Specification (TS) changes that provide for redundancy in decay heat removal capability in all modes of operation. NRC model TSs were provided as guidance.

GPU Nuclear Corporation (the licensee) responded to the NRC generic letter by letters dated January 26, 1982, and October 10, 1984, in which the generic issue was discussed but no actual changes to the TSs were requested. By letter dated March 8, 1985, as supplemented by letter dated May 14, 1985, GPU Nuclear Corporation applied for TS changes designed to assure redundant decay heat removal capability for all modes of reactor operation for TMI-1. In an April 24, 1986 letter, the licensee provided additional information concerning the amendment request.

DISCUSSION AND EVALUATION

The proposed TSs provide for redundant means of decay heat removal in all modes of operation except during refueling when a large mass of water is above the core. These redundant means are outlined as follows:

Power Operation and Hot Standby (reactor critical)

These modes for TMI-1 are comparable to Standard Technical Specification (STS) Modes 1 and 2, Power Operation and Startup. For TMI-1, both reactor coolant (RC) loops with one reactor coolant pump (RCP) in each loop must be in operation when the reactor is critical. With less than four RCPs in operation, the TSs require a reduction in power.

Hot Shutdown (reactor sub-critical)

This mode for TMI-1 (RC temperature above 525°F) is comparable to STS Mode 3, Hot Standby (RC temperature above 305°F). For TMI-1 in Hot Shutdown, both RC loops are required to be operable. The STS requirements for Mode 4, Hot Shutdown (RC temperature below 305°F) are essentially covered by the proposed TMI-1 TSs that require two decay heat removal means to be available with one loop in operation when the RC temperature is below 250°F, i.e., either two decay heat removal loops, two RC loops, or one of each.

The proposed TSs would allow the plant to operate in the equivalent of STS Mode 3 with one RCP operating and STS Mode 4 with no RCP operating. By letter dated January 9, 1986, the NRC staff requested the licensee to verify that operation in such modes will be bounded by the conditions in the TMI-1 FSAR analysis of accidents. By letter dated April 24, 1986, the licensee provided information by which they concluded that the proposed amendment would allow operation in a manner bounded by their FSAR analysis. This letter provided confirmatory information to the NRC staff but did not change the application for amendment as it existed in their May 14, 1985 submittal. This letter provides sufficient justification for the staff to conclude that the proposed TS changes place additional restrictions on the licensee by designating systems which must be operational when RC temperature is between 200°F and 250°F. Furthermore, the proposed TS changes do not allow TMI-1 to operate in any condition less restrictive than current TSs and are therefore acceptable.

Cold Shutdown and Refueling with the Water Level Above the Core Less Than 23 Feet

Two decay heat removal loops are required to be operable with at least one loop in operation.

Refueling with the Water Level Above the Core Greater Than 23 Feet

At least one decay heat removal loop is required to be in operation. The other loop need not be operable. Under such conditions, the mass of water above the core provides adequate heat removal capability and therefore a redundant method is not necessary.

In addition to the above requirements for operability, the proposed TS revisions specify surveillance intervals for heat removal systems that are consistent with the STSs. The revised TSs provide an improvement over the existing ones since redundant decay heat removal will now be provided when the plant is in the equivalent of STS Modes 3, 4 and 5. During refueling with a large mass of water above the core, only a single heat removal path is required. The surveillance requirements that would identify any inoperable equipment or degraded performance are performed daily. The NRC staff, therefore, concludes that the proposed TSs meet the intent of the STSs with respect to redundant means of decay removal capability and are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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, this 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.

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

Dated: August 14, 1986

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