March 9, 1987

Docket No. 50-289

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TMI Site Pouch
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PShemanski
JWermiel
DCrutchfield

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. 124 TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued the enclosed Amendment No.124 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 January 23, 1987, titled Technical Specification Change Request No. 166.

This amendment provides revised operability and surveillance requirements for the Emergency Feedwater (EFW) System at TMI-1. The revisions to the TSs were necessary due to significant system modifications made to comply with TMI-1 restart commitments and NUREG-0737, Action Item II.E.1. This included the installation of a Heat Sink Protection System to provide automatic control of main and emergency feedwater.

As discussed in the enclosed Safety Evaluation (SE), your proposal to allow an operational bypass of EFW auto start on low once-through steam generator (OTSG) water level, when reactor power is below 30%, is granted only for one cycle of operation and only under the conditions of a normal reactor startup or shutdown. Our principal objection is that removal of this operational bypass would be accomplished manually as reactor power increases, rather than automatically. Such a design is not in accordance with IEEE 279-1971 and increases the probability of an improper system lineup at higher reactor power levels. However, the Heat Sink Protection System is a new installation which may possibly experience problems resulting in inadvertent EFW initiation on low OTSG levels. While operational experience is obtained on this new system, we will allow the use of a bypass which is not automatically removed as the power level exceeds 30% (as required by IEEE 279-1971), under the very controlled conditions of a normal reactor startup or shutdown. Use of this bypass at any other time when reactor power is below 30% is denied. Authority to use this bypass is granted only for Cycle 6. Use of the bypass for future cycles will require modifying your system so that it satisfies the provisions of IEEE 279-1971 and a new amendment which justifies the setpoint for initiating the bypass.

Notice of Issuance will be included in the Commission's biweekly Federal Register notice. A separate Notice of Denial of Amendment is being forwarded to the Federal Register for publication.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 124 to DPR-50
2. Safety Evaluation
3. Notice of Denial

cc w/enclosures:
See next page

No legal objection, subject to changes noted.

*See previous white for concurrences

PBD-6*	PBD-6*	PBD-6	PBD-6*	PECISB*	AD:DPLB*	~~PBD-6~~	OGC
RIngram	TRoss	JThoma	RWeller	JCalvo	DCrutchfield	JStolz	SHLewis
2/26/87	2/26/87	03/05/87	2/27/87	2/25/87	2/26/87	03/05/87	3/5/87

Chgs. made 3/9/87

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As discussed in the enclosed Safety Evaluation (SE), your proposal to allow an operational bypass of EFW auto start on low once-through steam generator (OTSG) water level, when reactor power is below 30%, is hereby denied. Our principal objection is that removal of this operational bypass would be accomplished manually as reactor power increases, rather than automatically. Such a design is not in accordance with IEEE 279-1971 and increases the possibility of an improper system lineup at higher reactor power levels. If you choose to pursue this matter further, either through an appeal or a new amendment request, additional justification should also be provided for the actual setpoint selected, i.e., 30% reactor power.

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

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*SEE PREVIOUS CONCURRENCE

JO1	JO1	Raw					
PBD-6	PBD-6	PBD-6	PBD-6	PECISB*	AD:DPLB *	PBD-6	OGC
RIngram	TRoss for AW	JThoma	RWeller	JCalvo	DCrutchfield	JStolz	
/ /87	/ /87	02/26/87	2/27/87	/ /87	/ /87	2/27/87	/ /87

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/ /87	/ /87	02/19/87	/ /87	2/25/87	2/26/87	/ /87	/ /87

Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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Atomic Safety & Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Atomic Safety & Licensing Appeal
Board Panel (8)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
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Washington, D.C. 20555

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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.124
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 January 23, 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.

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. 124, 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

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 9, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3-25	3-25
3-26b	3-26b
3-27a	3-27a
3-28	3-28
3-28a	3-28a
3-32	3-32
3-32a	3-32a
---	3-32b
4-1	4-1
4-7	4-7
4-7a	4-7a
4-52	4-52
4-52a	4-52a
4-52b	---

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.

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.

---

*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.

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 EFW pumps, or the steam-driven EFW pump are required initially to remove decay heat with one EFW pump 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.

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 20% 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.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system bypass switch key permitted in the control room.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 460 volt sources. Redundant trip devices are employed in each of these sources. The AC Trip Breaker is one means to trip a source. The redundant means is a parallel configuration consisting of two DC Trip Breakers and five SCR power supplies. The SCRs are turned off by the "electronic trip relays."

Diverse trip features are provided on each breaker. These are the undervoltage relay and shunt trip attachment. Each trip feature is tested separately. Failure of one breaker trip feature does not result in loss of redundancy and a reasonable time limit is provided for corrective action.

Failure in the untripped state of a breaker or SCR electronic trip results in loss of redundancy and prompt action is required. Failure of both trip features on one breaker is considered failure of the breaker.

Power may be restored through the failed breaker (SCRs) for a limited time to perform required testing.

Automatic initiation of EFW is provided on loss of all reactor coolant pumps, loss of both main feedwater pumps, low OTSG level, and high reactor building pressure. High reactor building pressure would be indicative of a loss of coolant accident, main steam line or feedwater line break inside the reactor building. Operability of these instruments is required in order to assure that the EFW system will actuate and control at the appropriate OTSG level without operator action for those events where timely initiation of EFW is required.

Automatic isolation of main feedwater is provided on low OTSG pressure in order to maintain appropriate RCS cooling (minimize overcooling) following a loss of OTSG integrity and minimize the energy released to the Reactor Building atmosphere.

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

FSAR, Section 7.1

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]
C. Engineered Safety Features (cont'd)			
3. Reactor Building Isolation and Reactor Building Cooling System			
a. Reactor Bldg. 4 psig Instrument Channel	2	1	Hot Shutdown
b. Manual Pushbutton	2	1	Hot Shutdown
c. RPS Trip	2	1	Hot Shutdown
d. Reactor Building 30 psig	2	1	Hot Shutdown
e. RCS Pressure less than 1600 psig	2	1	Hot Shutdown
f. Reactor Bldg. Purge line Isolation (AHV-1A and AHV-1D) High Radiation	1	0	(f)
4. Reactor Building Spray System			
a. Reactor Bldg. 30 psig Instrument Channel	2[d]	1	Hot Shutdown
b. Spray Pump Manual Switches[c]	2	1	Hot Shutdown
5. 4.16KV ES Bus Undervoltage Relays			
a. Degraded Grid Voltage Relays	2	1	(e)
b. Loss of Voltage Relay	2	1	(e)

TABLE 3.5-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit

C. Engineered Safety Features (cont'd)

(a) If minimum conditions are not met within 24 hours, the unit shall then be placed in a cold shutdown condition.

(b) Also initiates Low Pressure Injection

(c) Spray valves opened by manual pushbutton listed in Item 3 above.

(d) Two out of three switches in each actuation channel operable.

(e) If a relay fails in the untripped state, it shall be placed in a tripped state within 12 hours to obtain a degree of redundancy of 1. The relay may be removed from the tripped state for up to 2 hours for functional testing pursuant to Table 4.1-1.

(f) Discontinue Reactor Building purging and close AHV-1A, 1B, 1C, and 1D. Note: (a) above does not apply if AHV-1A, 1B, 1C and 1D are closed.

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)]
D. Heat Sink Protection System			
1.) EFW Auto Initiation			
a. Loss of both Feedwater Pumps	N/A(b)	N/A(b)	Hot Shutdown
b. Loss of all RC Pumps	N/A(b)	N/A(b)	Hot Shutdown
c. OTSG A Low Level	2 (c)	1 (c)	Hot Shutdown
d. OTSG B Low Level	2 (c)	1 (c)	Hot Shutdown
e. High Reactor Building Pressure	2	1	Hot Shutdown
2.) MFW Isolation			
a. OTSG A Low Pressure	2	1	Hot Shutdown
b. OTSG B Low Pressure	2	1	Hot Shutdown
3.) EFW Level Control			
a. OTSG A Level Control	N/A(b)	N/A(b)	Hot Shutdown
b. OTSG B Level Control	N/A(b)	N/A(b)	Hot Shutdown

(a.) If minimum conditions are not met within 72 hours, the unit shall be placed in Hot Shutdown within the next 12 hours.

(b.) Operability requirements are specified in section 3.5.1.9.

(c.) For Cycle 6 operation only, bypass of the OTSG Low Level EFW train initiation may be placed in effect when indicated power is less than 30% for a normal reactor startup or shutdown. The bypass shall be removed when indicated reactor power is raised above 30%. If during a reactor startup, low power physics testing is being conducted, train actuation may be defeated as long as actual reactor power remains below 30%.

4. SURVEILLANCE STANDARDS

During Reactor Operational Conditions for which a Limiting Condition for Operation does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/component operable, the associated surveillance requirement must be current. The above applicability requirements assure the operability of systems/components for all Reactor Operating Conditions when required by the Limiting Conditions for Operation.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system, engineered safety feature protection system, and heat sink protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Each post accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies shown in Table 4.1-4.

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

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 T_{ave} 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 20% power
47a.	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

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
49.	Saturation Margin Monitor	S(1)	M(1)	R	(1) When T_{ave} is greater than 525°F
50.	Emergency Feedwater Flow Instrumentation	NA	M(1)	R	(1) When T_{ave} is greater than 250°F
51.	Heat Sink Protection System				(1) Includes logic test only
	a. EFW Auto Initiation Instrument Channels				
	1. Loss of both Feedwater Pumps	NA	Q(1)	R	
	2. Loss of all RC Pumps	NA	Q(1)	R	
	3. Reactor Building Pressure	NA	Q	R	
	4. OTSG Low Level	W	Q	R	
	b. MFW Isolation OTSG Low Pressure	NA	Q	R	
	c. EFW Control Valve Control System				
	1. OTSG Level Loops	W	Q	R	
	2. Controllers	W	NA	R	
	d. HSPS Train Actuation Logic	NA	Q(1)	R	
52.	Backup Incore Thermocouple Display	M(1)	NA	R	(1) When T_{ave} is greater than 250°F

S - Each Shift
D - Daily
W - Weekly
M - Monthly

T/W - Twice per week
B/M - Every 2 months
Q - Quarterly
P - Prior to each startup if not done previous week

R - Each Refueling Period
NA - Not applicable
B/W - Every two weeks

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 affected EFW local manual valves. 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 EFW System flowpath valve from both CSTs to the OTSGs via the motor driven pumps and the turbine driven pump shall be verified to be in the required status.

4.9.1.4 On a refueling interval basis:

a.) Verify that each EFW pump starts automatically upon receipt of an EFW test signal

b.) Verify that each EFW control valve responds upon receipt of an EFW test signal

c.) Verify that each EFW control valve responds in manual control from the control room and remote shutdown panel.

4.9.1.5 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.

4.9.1.6 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 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. 124 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 January 23, 1987, GPU Nuclear Corporation (GPU or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The proposed amendment would provide operability and surveillance requirements for the Emergency Feedwater (EFW) System, including the Heat Sink Protection System (HSPS), which are similar to those of safety-related systems.

EFW long-term upgrades were required by NUREG-0737, Item II.E.1. EFW System improvements, including the HSPS, were required by the Licensing Board to be installed prior to the startup following Cycle 6 Refueling (LBP-82-27, 15 NRC at 747 and TMI-1 Condition of Operation 3(a), dated October 2, 1985). The purpose of these modifications was to upgrade EFW to a safety grade system in order to provide increased EFW System reliability in mitigating the effects of design basis accidents when the Main Feedwater (MFW) System is not available. Since loss of MFW is an anticipated event, EFW System reliability should be high.

The HSPS provides for automatic initiation of EFW and once-through steam generator (OTSG) water level control independent of the Integrated Control System (ICS). As part of the modifications to improve the reliability of the EFW System, GPU has added redundant EFW control and block valves, and automatic EFW System initiation on OTSG low water level and high containment pressure. High containment pressure would be an indication of a main steam line break (MSLB), a main feedwater line break (MFLB) or a loss of coolant accident (LOCA). Automatic isolation of MFW on low OTSG pressure has been added such that MFW would be isolated in the event of a loss of OTSG integrity.

The NRC review of the TMI-1 EFW System is contained in the following documents:

1. NUREG-0680, TMI-1 Restart, June 1980
2. NUREG-0680, TMI-1 Restart, Supplement No. 3, April 1981
3. NRC letter to GPU dated April 24, 1984, forwarding a Safety Evaluation for NUREG-0737, Item II.E.1.2.
4. NRC letter to GPU dated February 18, 1987, forwarding a Safety Evaluation for the NUREG-0737, Item II.E.1.

The following Safety Evaluation is concerned with TSs to support the EFW System as a result of modifications made during the 6R refueling outage.

EVALUATION

Technical Specification Change Request No. 166 (TSCP 166) dated January 23, 1987, proposed TSs for the modified TMI-1 EFW System and associated HSPS. The NRC staff used, as review criteria, the Standard Technical Specifications (STSs) for auxiliary feedwater systems as contained in NRC Generic Letter 83-37 dated November 1, 1983. Additional review criteria were obtained from commitments made in the system reviews as documented in the introduction section of this Safety Evaluation. Individual changes to each TS page are discussed below:

Page 3-25

A footnote on this page which defines an operable flow path for the EFW System has been deleted. The EFW System has been upgraded to safety grade. EFW System operability is required in order to meet the conditions assumed in the safety analysis taking into account a single active failure. Therefore, the footnote definition of an operable EFW System is not applicable to the safety grade EFW System and should be deleted.

References to Section 3.0.1 of the TSs are deleted from this page. Section 3.0.1 concerns general action statements for when conditions of the TSs are not satisfied and no specific action is provided. Since specific action statements are provided concerning inoperability of various components of the EFW System, a reference to the general action statements is not needed.

Section 3.4.1.1 states that the EFW System shall be operable whenever the Reactor Coolant System temperature is greater than 250°F. A footnote being added to this page states that the HSPS operability is specified in Section 3.5.1 of the TSs. The HSPS provides automatic initiation and control functions for the EFW System. The automatic initiation functions are covered under other sections of the TSs. However, in order for the EFW pumps and associated flow paths to be considered operable, certain control subsections of the HSPS must also be considered operable. Therefore, this footnote is not interpreted to mean that the entire HSPS is allowed to be inoperable when EFW is required to be operable.

Based on the above considerations, the proposed changes to page 3-25 of the TSs are considered acceptable by the NRC staff.

Page 3-26b

The bases to Section 3.4.1.1 of the TSs are modified to state that either both motor driven EFW pumps or the single steam-driven EFW pump are required initially to remove decay heat, with one EFW pump eventually sufficing. Adequate technical justification is provided in the system reviews to support this statement. Therefore, this change is acceptable.

Page 3-27a

Sections 3.5.1.9 and 3.5.1.9.1 have been added to provide operability requirements for the HSPS. The operability requirements of the HSPS are similar to those of the Reactor Protection System (RPS), and allowed outage times are equivalent to those specified for trains of the EFW System. HSPS actuation logic trains are required to be operational whenever the reactor is critical or in the Startup Mode. However, subsections of the HSPS, which are necessary for manual control of the EFW System, are governed by TSs requiring EFW operability when Reactor Coolant System temperature is above 250°F. The proposed TS changes are similar to other TSs on safety-related systems at TMI-1 and are acceptable.

Pages 3-27a, 3-28, and 3-28a

The Basis for Section 3.5.1, Operational Safety Instrumentation, has been revised to include a section on the HSPS. The additions to the Basis section are satisfactory. However, part of the proposed addition has been deleted where it discusses bypassing EFW automatic initiation on low OTSG level when reactor power is below 30%. As discussed under changes to page 3-32b of the TSs, the operational bypass aspect of the amendment is being denied, and it is therefore deleted from the Basis statements.

Pages 3-32 and 3-32a

Table 3.5-1, functional item C.6 concerning EFW System automatic initiation features has been deleted. It is being replaced with a new functional item D which is discussed in our review of page 3-32b of the TSs. Also, a footnote on page 3-32a, which was only applicable for hot functional testing prior to Cycle 5 criticality, has been deleted as it is no longer effective. These changes are administrative in nature and are acceptable.

Page 3-32b

This page is proposed to add functional unit D, Heat Sink Protection System, to Table 3.5-1 of the TSs. This table establishes the minimum number of operable channels and minimum degree of redundancy for HSPS components. Channel operability and redundancy for EFW auto initiation on loss of both feedwater pumps or loss of all RC pumps is shown to be not applicable (N/A) since this instrumentation does not fit the definition of a "channel". Operability of these components is required whenever the HSPS is required to be operational (per TS Section 3.5.1.9). Therefore, showing these components as N/A, since they do not fit the definition of a channel, is acceptable because it does not change the overall operability requirements.

The licensee has also proposed adding an operational bypass of EFW auto initiation on low OTSG level during low power physics testing and when reactor power is less than 30%. The principal objection of the NRC staff is that this operational bypass is to be manually removed when the conditions which allow it to be implemented have been exceeded. In accordance with

NUREG-0737, Item II.E.1, the upgraded EFW auto initiation design should meet the criteria of IEEE 279-1971. This standard would require that operational bypasses are to be automatically removed when the conditions which allow them to be implemented have been exceeded. The licensee's proposed alternative of manual action on a permanent basis is not acceptable to the NRC staff.

Manual action does increase operational flexibility, but it also increases the risk that the system will be improperly aligned at higher power levels. However, the HSPS is a new system for controlling OTSG level. Our experience with other plants is that there may be a period of adjustment to the new system. There is a possibility of inadvertent EFW initiation at low power levels in various transients due to the close proximity of the EFW auto initiation setpoint on low OTSG level and the low level limit of the OTSG level control circuitry. Therefore, to allow system familiarization during one cycle of operation, we will accept the bypass of auto initiation of EFW on low OTSG level when reactor power is below 30% under the conditions of a normal reactor startup or shutdown. Although the bypass will not be automatically removed as power is increased, adequate design and procedural controls will insure that the bypass switch is in the appropriate position for operation at 30% power levels and above. First, when the EFW auto start circuitry is in the bypass position, an alarm is indicated in the control room. Second, the startup and shutdown procedures require the operators to acknowledge by signature the status of the EFW system at 30% reactor power. These factors insure positive control of the EFW initiation bypass switches during a startup or shutdown.

The NRC staff is not opposed to a permanent operational bypass that meets the criteria of IEEE 279-1971. Justification of operational bypasses to be utilized after Cycle 6 must be contained in an additional amendment application. This application should also include a detailed justification of the setpoint chosen for initiation of the bypass.

The other provisions of functional unit D are equivalent to provisions on other safety-related systems and are acceptable.

Page 4-1

The HSPS instrumentation is being added to the list of the minimum frequency and type of surveillance required for instrumentation when the reactor is critical. This addition is desired and is acceptable.

Page 4-7

This page contains changes to Table 4.1-1 of the TSs. Line Item 38 has been changed from Steam Generator Water Level to OTSG Full Range Level. Startup and operating range OTSG levels are part of the HSPS, and these instruments have been included with other HSPS instrumentation under a new Item 51, HSPS. Full range OTSG level is not part of the HSPS, so Item 38 has been revised to include only the Full range level instrument. This change is acceptable.

A footnote which was only applicable for Cycle 5 operation has been deleted from the page and an NA (not applicable) has been added to a blank space on the table. These changes are administrative in nature and are acceptable to the NRC staff.

Page 4-7a

This page adds a new Item 51 to Table 4.1-1 to add the HSPS. It provides for weekly cross checks of appropriate instrumentation, quarterly testing, and calibration on a refueling outage basis. It also deletes a reference requiring operability when T_{ave} is greater than 250°F to be consistent with operability requirements of the HSPS as specified in TS 3.5.1. These changes are consistent with the STSs and commitments made during the systems review. Therefore, these proposed changes are acceptable.

Page 4-52

A footnote on this page which defines an operable flowpath for the EFW System has been deleted. This is the same footnote as deleted from page 3-25, and this change is acceptable for the same reason.

This page is also modified to delete all reference to Table 4.9-1 and the table itself is being deleted from the TSs. Table 4.9-1 is a detailed listing of EFW valves and their correct status for proper alignment. The requirement to verify the position of EFW flowpath valves every 31 days is stated in TS 4.9.1.3. The licensee has procedures (specifically SP 1300-3F and SP 1300-3G) to implement this valve lineup requirement. Specifying the valve lineup in the TS is unnecessary, and it is acceptable to delete this table from the TSs.

TS 4.9.1.4 has been added to require EFW component testing on a refueling interval basis. The proposed testing is consistent with the STSs and is acceptable.

The manual control valve station (HIC-849/850), which was part of the interim system design, has been replaced as part of the long-term modifications during the Cycle 6 refueling outage. TS 4.9.1.4 testing requirements related to this temporary valve station have been deleted. Additionally, the redundant requirement for full-stroke testing of the EFW control valves in TS 4.9.1.5 is being deleted as the same testing requirement is contained in TS 4.2.2. These changes are acceptable.

Page 4-52a

The Bases portion of TS 4.9 has been revised to reflect system upgrades and additional review findings. The changes are consistent with the system review safety evaluations and are acceptable.

SUMMARY

With the exception of the denied portion concerning an operational bypass of the EFW auto initiation, the proposed TSs are consistent with the provisions of our Generic Letter 83-37 and the system review safety evaluations. Therefore, with the exception of the denied portion, the proposed TS changes 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: March 9, 1987

Principal Contributor:
John Thoma

NUREG-0737, Item II.E.1, the upgraded EFW auto initiation design should meet the criteria of IEEE 279-1971. This standard would require that operational bypasses are to be automatically removed when the conditions which allow them to be implemented have been exceeded. The licensee's proposed alternative of manual action on a permanent basis is not acceptable to the NRC staff.

Manual action does increase operational flexibility, but it also increases the risk that the system will be improperly aligned at higher power levels. However, the HSPS is a new system for controlling OTSG level. Our experience with other plants is that there may be a period of adjustment to the new system. There is a possibility of inadvertent EFW initiation at low power levels in various transients due to the close proximity of the EFW auto initiation setpoint on low OTSG level and the low level limit of the OTSG level control circuitry. Therefore, to allow system familiarization during one cycle of operation, we will accept the bypass of auto initiation of EFW on low OTSG level when reactor power is below 30% under the conditions of a normal reactor startup or shutdown. Although the bypass will not be automatically removed as power is increased, adequate design and procedural controls will insure that the bypass switch is in the appropriate position for operation at 30% power levels and above. First, when the EFW auto start circuitry is in the bypass position, an alarm is indicated in the control room. Second, the startup and shutdown procedures require the operators to acknowledge by signature the status of the EFW system at 30% reactor power. These factors insure positive control of the EFW initiation bypass switches during a startup or shutdown.

The NRC staff is not opposed to a permanent operational bypass that meets the criteria of IEEE 279-1971. Justification of operational bypasses to be utilized after Cycle 6 must be contained in an additional amendment application. This application should also include a detailed justification of the setpoint chosen for initiation of the bypass.

The other provisions of functional unit D are equivalent to provisions on other safety-related systems and are acceptable.

Page 4-1

The HSPS instrumentation is being added to the list of the minimum frequency and type of surveillance required for instrumentation when the reactor is critical. This addition is desired and is acceptable.

Page 4-7

This page contains changes to Table 4.1-1 of the TSs. Line Item 38 has been changed from Steam Generator Water Level to OTSG Full Range Level. Startup and operating range OTSG levels are part of the HSPS, and these instruments have been included with other HSPS instrumentation under a new Item 51, HSPS. Full range OTSG level is not part of the HSPS, so Item 38 has been revised to include only the Full range level instrument. This change is acceptable.

A footnote which was only applicable for Cycle 5 operation has been deleted from the page and an NA (not applicable) has been added to a blank space on the table. These changes are administrative in nature and are acceptable to the NRC staff.

Page 4-7a

This page adds a new Item 51 to Table 4.1-1 to add the HSPS. It provides for weekly cross checks of appropriate instrumentation, quarterly testing, and calibration on a refueling outage basis. It also deletes a reference requiring operability when T_{ave} is greater than 250°F to be consistent with operability requirements of the HSPS as specified in TS 3.5.1. These changes are consistent with the STSs and commitments made during the systems review. Therefore, these proposed changes are acceptable.

Page 4-52

A footnote on this page which defines an operable flowpath for the EFW System has been deleted. This is the same footnote as deleted from page 3-25, and this change is acceptable for the same reason.

This page is also modified to delete all reference to Table 4.9-1 and the table itself is being deleted from the TSs. Table 4.9-1 is a detailed listing of EFW valves and their correct status for proper alignment. The requirement to verify the position of EFW flowpath valves every 31 days is stated in TS 4.9.1.3. The licensee has procedures (specifically SP 1300-3F and SP 1300-3G) to implement this valve lineup requirement. Specifying the valve lineup in the TS is unnecessary, and it is acceptable to delete this table from the TSs.

TS 4.9.1.4 has been added to require EFW component testing on a refueling interval basis. The proposed testing is consistent with the STSs and is acceptable.

The manual control valve station (HIC-849/850), which was part of the interim system design, has been replaced as part of the long-term modifications during the Cycle 6 refueling outage. TS 4.9.1.4 testing requirements related to this temporary valve station have been deleted. Additionally, the redundant requirement for full-stroke testing of the EFW control valves in TS 4.9.1.5 is being deleted as the same testing requirement is contained in TS 4.2.2. These changes are acceptable.

Page 4-52a

The Bases portion of TS 4.9 has been revised to reflect system upgrades and additional review findings. The changes are consistent with the system review safety evaluations and are acceptable.

SUMMARY

With the exception of the denied portion concerning an operational bypass of the EFW auto initiation, the proposed TSs are consistent with the provisions of our Generic Letter 83-37 and the system review safety evaluations. Therefore, with the exception of the denied portion, the proposed TS changes 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: March 9, 1987

Principal Contributor:
John Thoma

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

NOTICE OF DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSE

AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied, in part, a request by GPU Nuclear Corporation, et al. (the licensees) for an amendment to Facility Operating License No. DPR-50 issued to GPU Nuclear Corporation, et al. for operation of the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1), located in Dauphin County, Pennsylvania. Notice of consideration of issuance of this amendment was published in the FEDERAL REGISTER on February 4, 1987 (52 FR 3515).

The purpose of the licensees' amendment request was to incorporate new and revised Technical Specification (TS) requirements for operability and surveillance for the Emergency Feedwater (EFW) System at TMI-1. The revisions were necessary due to significant system modifications made to comply with TMI-1 restart commitments and NUREG-0737, Action Item II.E.1. Included in this proposal was a request to manually bypass automatic initiation of EFW on low once-through steam generator level during low power physics testing and when reactor power is below 30%.

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The principal objection of the Commission is that this operational bypass would be manually removed when the conditions which allow it to be implemented were exceeded. In accordance with NUREG-0737, Item II.E.1, the upgraded EFW auto initiation design should meet the criteria of IEEE 279-1971. This standard would require that operational bypasses are to be automatically removed when the conditions which allowed them to be implemented were exceeded. The licensees' proposed alternative of manual action on a permanent basis is not acceptable. Manual action does increase operational flexibility, but it also increases the risk that the system will be improperly aligned at higher power levels. The HSPS is a new system for controlling OTSG level. There is a potential of inadvertent EFW initiation at low power levels during various transients due to the close proximity of the EFW auto initiation setpoint on low OTSG level and the low level limit of the OTSG level control. To allow system familiarization during one cycle of operation, the NRC will allow the licensee to manually insert a bypass of auto initiation of EFW on low OTSG level when reactor power is below 30% under the conditions of a normal reactor startup or shutdown. Although this bypass is not automatically removed as power level increases above 30%, adequate design and procedural controls will ensure that the bypass switch is in the proper position for operation at 30% power levels and above. When the EFW auto start circuitry is in the bypass position, an alarm is indicated in the control room. The startup and shutdown procedures require the operator to acknowledge by signature the status of the of the EFW system at 30% reactor power. These two factors will insure the bypass switch is not in the bypass position at higher power levels. The NRC staff is denying the use of the bypass switch under conditions other than normal startup or shutdown. Additionally, the NRC staff is granting permission to use the bypass switches under restrictive circumstances for Cycle 6 operation only. Use of any bypass capability after Cycle 6 operation will require a new amendment.

All other provisions of the amendment request have been approved by Amendment No. 124 dated March 9, 1987. Notice of Issuance of Amendment No. 124 will be published in the Commission's biweekly FEDERAL REGISTER notice.

The licensees were notified of the Commission's denial of the proposed TS change by letter dated March 9, 1987.

By April 13, 1987, the licensees may demand a hearing with respect to the denial described above and any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel-Bethesda, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Ernest L. Blake, Jr. of Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037, attorney for the licensees.

For further details with respect to this action, see (1) the application for amendment dated January 23, 1987, (2) the Commission's letter to GPU Nuclear Corporation dated March 9, 1987, and (3) the Commission's Safety Evaluation issued with Amendment No. 124 to DPR-50 dated March 9, 1987.

These documents are available for public inspection at the Commission's Public Document Room, 1717 H Street, 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 PWR Licensing-B.

Dated at Bethesda, Maryland, this 9th day of March, 1987.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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