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June 14, 2000
1920-98-20681 B

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1),
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
REVISED TECHNICAL SPECIFICATION CHANGE REQUEST NO. 262
SECTIONS 1, 3, AND 4 IMPROVEMENTS

In accordance with 10 CFR 50.4 (b) (1), enclosed is Revised Technical Specification Change Request No. 262. Also enclosed is the Certificate of Service for this request certifying service to the chief executives of the township and county in which the facility is located, as well as the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection.

The purpose of this TSCR is to request that the TMI-1 Technical Specifications Sections 1.4.2, 1.4.3, 1.4.4, 3.1.12.3, 3.3.1.2.b and d, 3.3.1.3, 3.3.2, 3.3.2.1, Table 4.1-1 (Items 14, 31, and 32), Table 4.1-3 (Item 4), 4.5.2.1, 4.5.2.3, and 4.5.3.1, be revised to add LCO action statements and make surveillance requirements more consistent with the Revised Standard Technical Specifications for B&W Plants (NUREG-1430), to correct conflicts or inconsistencies caused by earlier Technical Specification revisions, and to revise spent fuel pool sampling from monthly and after adding chemicals to weekly.

Pursuant to 10 CFR 50.91 (a) (1), enclosed is our analysis, applying the standards in 10 CFR 50.92 to make a determination of no significant hazards considerations. As stated above, pursuant to 10 CFR 50.91(a), we have provided a copy of this letter, the proposed changes in the Technical Specifications, and our analyses of no significant hazards considerations to the designated representative of the Commonwealth of Pennsylvania.

Very truly yours,

John B. Cotton

John B. Cotton
Vice President, TMI Unit 1

A001

NRR-057

JBC/vlk

Enclosures: (1) Technical Specification Change Request No. 262
(2) TMI-1 Technical Specification revised pages
(3) Certificate of Service for Technical Specification Change Request No. 262

cc: Administrator Region I
TMI Senior Resident Inspector
TMI-1 Project Manager
File 97033

AMERGENENERGY, LLC

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 262

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF DAUPHIN)

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A to the License are also included. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

AMERGEN ENERGY COMPANY, LLC

BY: John B Cotton
Vice President, TMI Unit 1

Sworn and Subscribed to before me
this 14Th day of June, 2000.

Linda C. Witter
Notary Public

Notarial Seal
Linda C. Witter, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Sept. 25, 2000
Member, Pennsylvania Association of Notaries

Enclosure 1
TMI-1 Technical Specification Change Request No. 262
And No Significant Hazards Consideration Analysis

TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) No. 262

I. GPU Nuclear requests that the following changed replacement pages be inserted into the existing Technical Specification:

Technical Specification revised pages: 1-3, 3-21, 3-22, 4-4, 4-6, 4-10, 4-41, 4-42, and 4-43.

These pages are attached to this change request.

II. REASON FOR CHANGE WITH BRIEF DESCRIPTION OF CHANGES, SAFETY EVALUATION JUSTIFYING CHANGE, AND MARKUP

One purpose of this Technical Specification change request is to incorporate certain improvements from the Revised Standard Technical Specifications for B&W Plants (NUREG-1430) that would add limiting conditions for operation action statements, make surveillance requirements more consistent with the Revised Standard Technical Specifications, correct conflicts or inconsistencies caused by earlier Technical Specification revisions, to revise spent fuel pool sampling from monthly and after adding chemicals to weekly, and correct administrative errors.

The following refers to page numbers associated with the location of text on the revised pages and describes the changes, provides a safety evaluation justifying the change, and includes a markup of the existing specification:

A. Technical Specifications Page 1-3:

Description

The Upgraded Final Safety Analysis Report was revised to eliminate Figure 7.1-1. The recommended changes in Technical Specifications sections 1.4.2, 1.4.3, and 1.4.4 replace the references to this non-existent figure and refers to the appropriate UFSAR section.

Safety Evaluation

These changes are administrative in nature and serve only to correct an omission from a previous Technical Specification amendment. Also the referenced sections of the UFSAR give a more complete description of the Reactor Protection System including the controlled drawing which constituted the previous figure 7.1-1.

Markup

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is ~~shown~~ described in ~~Figure 7.1-1~~ Section 7.1 of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as ~~shown~~ described in ~~Figure 7.1-1~~ Section 7.1 of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as ~~shown~~ described in ~~Figure 7.1-1~~ Section 7.1 of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

B. DELETED

C. Technical Specifications Pages 3-21 and 22:

Description

The present specifications 3.3.1.2.b, 3.3.1.2.d, 3.3.1.3.b, & 3.3.1.3.c do not have associated action statements. The revised specification includes action statements consistent with NUREG 1430, B&W Standard Technical Specifications. STS 3.5.1 specifies a maximum of 72 hours with CFT boron concentration not within limits. The TMI specification 3.3.1.2.b is the equivalent specification to standard technical specification 3.5.1.A. RSTS 3.6.7 specifies a maximum of 72 hours with an inoperable spray additive system. The TMI specifications 3.3.1.3.b & c are the equivalent specifications to RSTS 3.6.7

The present specification 3.3.1.2.d conflicts with the action statement 3.3.2. Specification 3.3.1.2.d requires one CFT pressure instrument channel to be operable. The maintenance specification explicitly states its applicability to CFT pressure instruments but then allows for 72 hours only as long as it will not remove more than one train from service. The same conflict applies to the level instrument channel. Specification 3.3.2 apparently intended to allow 72 hours when specification 3.3.1.2.d was not met, but the present statement does not clearly allow this. The revised wording would make this clear.

Safety Evaluation

These specifications do not have associated action statements. The revised specification includes action statements consistent with NUREG 1430, B&W Standard Technical Specifications, and provide assurance of proper actions in the event these specifications are not met.

There is no STS requirement for CFT pressure or level instrumentation operability. The CFT pressure and level instruments do not have an accident mitigation function. This issue was reviewed and the NRC agreed with this conclusion when accepting the

RG 1.97 categorization of these instruments. These instruments are RG 1.97 Category 3.

Markup

3.3.1.2 Core Flooding System

a. Two core flooding tanks each containing 940 $\pm$ 30 ft^3 of borated water at 600 $\pm$ 25 psig shall be available. Specification 3.0.1 applies.

b. Core flooding tank boron concentration shall not be less than 2,270-ppm boron. **Specification 3.3.2.1 applies.**

c. The electrically operated discharge valves from the core flood tank will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. Specification 3.0.1 applies.

d. One core flood tank pressure instrumentation channel and one core flood tank level instrumentation channel per tank shall be operable. **Specification 3.3.2.1 applies.**

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System
The following components must be OPERABLE:

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

b. The sodium hydroxide (NaOH) tank shall be maintained at 8 ft. ±6 inches lower than the BWST level as measured by the BWST/NaOH tank differential pressure indicator. The NaOH tank concentration shall be 10.0 $\pm$5 weight percent (%). ~~If the NaOH concentration is not within limits, restore to OPERABLE within 72 hours. If the BWST/NaOH tank level differential is not within limits, restore to OPERABLE within 72 hours.~~ **Specification 3.3.2.1 applies**.

c. All manual valves in the discharge lines of the sodium hydroxide tank shall be locked open. **Specification 3.3.2.1 applies.**

3.3.1.4 Cooling Water Systems – Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.
b. Two nuclear service river water pumps must be OPERABLE.
c. Two decay heat closed cycle cooling water pumps must be OPERABLE.
d. Two decay heat river water pumps must be OPERABLE.
e. Two reactor building emergency cooling river water pumps must be OPERABLE.

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

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

3.3.2.1 If the CFT boron concentration is outside of limits, or NaOH tank is outside of limits, restore the system to operable status within 72 hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

D. Technical Specifications Page 4-4 (Table 4.1-1, Item 14):

Description

Tech Spec 4.1-1.14 (item 14 on Table 4.1-1) was originally (i.e. Amendment 47 issued with FSAR) "High Pressure Injection Logic Channel." It revised TS 14.1-1.14 to read "High Reactor Building Logic Channel" by errors in reproduction when other changes were made to this page. There is no Technical Specification change request or Technical Specification Amendment that supports the present wording. The original wording was correct. There are four items on Table 4.1-1 for quarterly testing of ESAS logic channels (items 14,16,18 & 20). These correspond to the four major sections of the original ESAS logic. This change would correct an editorial error that revises TS 4.1-1.14 to read "High Pressure Injection Logic Channel."

Safety Evaluation

This administrative change restores the Technical Specifications to the approved condition. The present version of this specification was not intentionally changed. This change corrects an editorial error made in the past.

Markup

1. TABLE 4.1-1 (Continued)

2.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	F	
11. (Deleted)	----	----	----	
12. Pump Flux Comparator	S	M	R	
13. High Reactor Building Pressure Channel	S	M	F	
14. High ~~Reactor Building~~ **Pressure Injection** Logic Channels	NA	Q	NA	

E. Technical Specifications Page 4-6 (Table 4.1-1 items 31 and 32):

Description

License Amendment No. 196 (dated 9/19/95) revised the Technical Specifications to remove the Makeup, Purification, and Chemical Addition systems from Section 3.2 of the

Technical Specifications and the pertinent design information was relocated to the UFSAR. This proposed change serves to conform the Technical Specifications to Amendment No. 196 due to an administrative oversight in the corresponding TSCR No. 252. Therefore, the change is administrative in nature.

This proposed change removes instrument surveillance requirements for the Boric Acid Mix Tank temperature instrument and the Reclaimed Boric Acid Storage Tank temperature instrument and is consistent with the Standard Technical Specifications for Babcock and Wilcox Plants, NUREG-1430, July 1992. Also this change meets the intent of the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors issued July 1992 and codified in 10 CFR 50.36.

Safety Evaluation

These items were changed to correct an administrative error. License Amendment No. 196 (dated 9/19/95) revised the Technical Specifications to remove the Makeup, Purification, and Chemical Addition systems from Section 3.2 of the Technical Specifications and the pertinent design information was relocated to the UFSAR. This proposed change serves to conform the Technical Specifications to Amendment No. 196 due to an administrative oversight in the corresponding TSCR No. 252. Therefore, the change is administrative in nature.

Markup

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
30.	Borated Water Storage Tank Level Indicator	W	NA	F	
31.	**DELETED** ~~Boric Acid Mix Tank~~				
	~~a. Level Channel~~	~~NA~~	~~NA~~	~~F~~	
	~~b. Temperature Channel~~	~~M~~	~~NA~~	~~F~~	
32.	**DELETED** ~~Reclaimed Boric Acid Storage Tank~~				
	~~a. Level Channel~~	~~NA~~	~~NA~~	~~F~~	
	~~b. Temperature Channel~~	~~M~~	~~NA~~	~~F~~	

F. Technical Specifications Page 4-10:

Description

Item 4 to Table 4.1-3 was revised to change sampling the spent fuel pool for boron concentration from after each makeup and monthly to weekly in accordance with the B&W Standard Technical Specifications (NUREG-1430) 3.7.15.1.

Item 6 to Table 4.1-3 was deleted to conform to Technical Specification amendment No. 196, which deleted Section 3.2 and relocated the design information to the UFSAR based upon GPUN's TSCR no. 252, dated August 11, 1995 that failed to request deletion of the corresponding surveillances in Section 4.1, which must also be deleted. Hence this TSCR serves to conform the Technical Specifications to amendment No. 196.

Safety Evaluation

Item 4 to Table 4.1-3 was revised to weekly for the requirement to sample for boron concentration in the spent fuel pool in accordance with the B&W Standard Technical Specifications (NUREG-1430) 3.7.15.1. The criteria that govern the storage rack locations of fuel assemblies in the Spent Fuel Pool were developed without taking credit for boron. The analyses show that a boron concentration of 600 ppmb will meet the NRC maximum allowable reactivity value under the postulated fuel handling accident condition; if not moving fuel, no minimum boron concentration is required. The spent fuel pool boron is normally about 2700 ppmb in order to match the refueling boron requirement in Technical Specification 3.8.4 in the reactor coolant system and the fuel transfer canal during refueling operations.

If water were added to the spent fuel pool, based upon the current high and low level alarm set points, it would not drop below 343' 6" and would not be increased above 345' – about a two foot change. In this pool with a maximum volume of about 435,000 gallons, the two-foot drop represents about 32,000 gallons, which would lower the pool to 403,000 gallons. Procedure N1800.2 (Chemistry Specifications) has an administrative limit of 2,650 to 5,000 ppm boron. Using the lower limit of 2,650 ppm, the boron would drop to 2,455 ppm, which is well above the Technical Specification minimum value of 600 ppm. Even if the spent fuel pool boron dropped to as low as 650 ppm boron, the addition of 32,000 gallons of water would lower it to 602 ppm boron that is still above the minimum Technical Specification value. Therefore, routine makeups of the spent fuel pool with non-borated demineralized water would not reasonably be expected to exceed the minimum boron concentration limits.

Movements of fuel assemblies in the spent fuel pool storage racks are independently verified at the time of each movement to assure that the storage requirements of technical specification 5.4 are met. Makeup to the spent fuel pool is typically required due to evaporation of water from the surface of the pool, which leaves the pool at a slightly higher born concentration. Makeup to the pool simply restores the pool level and boron concentration. These margins and the large volume of the spent fuel pool provide assurance that boron concentration in the pool will be maintained within the limit by a weekly boron concentration sampling frequency.

Therefore, sampling weekly is justified and is consistent with the RSTS. The administrative changes have no affect upon safety.

Markup

TABLE 4.1-3 Cont'd

MINIMUM SAMPLING FREQUENCY

	Item	Check	Frequency
4.	Spent Fuel Pool Water Sample	Boron concentration greater than or equal to 600 ppmb	**Weekly** ~~Monthly and after each makeup~~
5.	Secondary Coolant System Activity	Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	At least once per 72 hours when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F.

~~6. Boric Acid Mix Tank or Reclaimed Boric Acid Tank~~ ~~Boron Concentration~~ ~~Twice weekly***~~

~~*** The surveillance of either the Boric Acid Mix Tank or the Reclaimed Boric Acid Tank is not necessary when that respective tank is empty.~~

G. Technical Specifications Page 4-41:

Description

Tech Spec 4.5.2.1 requires that a high-pressure injection system performance test be performed each refueling interval. The current specification requires that testing must be done by an operator starting the pumps and a test signal must be used to open the valves. The requirement to use a "test signal" to open the valves adds complexity to the testing without adding value. Other Technical Specifications provide requirements to ensure that the high-pressure injection valve actuation logic is tested (TS 4.1-1 item 14) and the valve operation is tested (TS 4.2.2 & 4.5.2.4.a). These tests are performed quarterly using the ES logic test signal. During the refueling interval, the high-pressure injection performance test, adds complexity. The operators must carefully manipulate the ES test features and verify that only the desired components are actuated, at a time when their attention must also consider low-temperature over protection issues and verification of proper high-pressure injection component and system performance. Removing the ES test signal requirement would allow the valves to be opened as part of the test setup and the risks associated with inadvertent ESAS actuation would be minimized. This revision to 4.5.2.1 removes the requirement to use the test signal to open the valves and uses the measured flow to confirm system performance.

Safety Evaluation

The scope of testing of the Engineered Safeguards Actuation System (ESAS) and High-Pressure Injection (HPI) systems is not affected by this change. The change eliminates unnecessary overlap between requirements. The frequency of testing of the valve actuation logic and valve operation is not affected. The required system performance is verified by system flow. System flow is the most accurate and appropriate means to verify the capability of the system to meet accident analysis assumptions.

The change in test method does not degrade the quality or scope of the system performance evaluation. The present test method requirement evaluates system performance based on the flow indication after a "test signal" has been used to open the valves (MU-V-16 A/B or C/D). The test acceptance criteria are unchanged. The HPI system configuration when HPI flow is measured is unchanged.

RSTS includes three separate requirements [3.5.2.4, 3.5.2.5, 3.5.2.6] to (1) test pump performance IAW IST requirements (2) verify auto actuation of the HPI valves every 18 months and (3) verify the auto actuation of the pump every 18 months. The revised TMI Technical Specifications would include similar requirements except where the TMI requirements exceed the RSTS. The component operation on an ES actuation is tested quarterly vs. every 18 months per the Technical Specifications.

Markup

Specification:

4.5.2.1 High Pressure Injection

a. During each refueling interval and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

~~The makeup pump and its required supporting auxiliaries will be started manually by the operator and a test signal will be applied to the high pressure injection (HPI) valves MU-V-16A/B/C/D to demonstrate actuation of the high pressure injection system for emergency core cooling operation.~~

b. The test will be considered satisfactory if the valves **(MU-V-14A/B & 16A/B/C/D)** have completed their travel and the make-up pumps are running as evidenced by **system flow** ~~the control board component operating lights~~. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

H. Technical Specifications page 4-42:

Description

Technical Specification 4.5.2.3 requires that a core flood system test be performed each refueling interval. The test verifies that the check valves between the core flood tanks and the reactor vessel will open as designed. The present requirement specifies that the test be performed "while depressurizing the reactor coolant system." Literal compliance with this phrase would not allow the test to be done when at cold shutdown or during plant heat up. The inservice test requirements (TS 4.2.2) for full flow testing of these same check valves is met by testing while the reactor coolant system is depressurized. The present version of 4.5.2.3 does not allow the inservice test (which verifies that the check valve opens and can pass the accident required flow rate) to be credited as a system test. The revision requires that the system test be completed, without specifying plant conditions during the test. A single test can then be performed to satisfy the inservice test requirement and the system test requirement.

Safety Evaluation

The core flood valves can be tested (open verification) with the RCS depressurized without impacting the quality of the test result. The test per specification 4.5.2.3 requires that operation of the check valves be verified by an indication that the core flood tank level has decreased (as described in the bases). This test, "Will water flow from the core flood tanks to the RV?" can be performed with the RCS depressurized by opening the core flood tank isolation valves when core flood tank pressure is greater than RCS pressure.

Markup

4.5.2.3 Core Flooding

a. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. ~~During depressurization of the Reactor Coolant System,~~ Verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

I. Technical Specifications page 4-43:

Description

Technical Specification 4.5.3.1 (similar to Item G above) requires that a system reactor building emergency cooling system performance test be performed on a refueling interval. This specification also includes the same unnecessary complexity as found in 4.5.2.1. The specification requires that the valves be opened by a "test signal." This additional requirement does not add value to the test. The valve actuation logic (TS 4.1-1.18) and valve operation (TS 4.2.2 & TS 4.5.3.2.a) is tested as required quarterly. This revision to 4.5.3.1 removes the requirement to use the test signal to open the valves and uses the measured flow to confirm system performance.

Safety Evaluation

The scope of testing of the engineered safeguards actuation system and the reactor building emergency cooling system is not affected by this change. The change eliminates unnecessary overlap between requirements. The frequency of testing of the valve actuation logic and valve operation is not affected. The required system performance is verified by system flow. System flow is the most accurate and appropriate means to verify the capability of the system to meet accident analysis assumptions.

The change in test method does not degrade the quality or scope of the system performance evaluation. The present test method requirement evaluates system performance based on the component indications after a "test signal" has been used to open the valves. The revised test acceptance criteria uses flow to evaluate system performance. The RBEC system configuration when flow is measured is unchanged.

RSTS includes two related requirements [3.6.6.3, 3.6.6.7] to (1) verify RBEC system flow capacity exceeds design every 31 days and (2) verify the auto actuation of the RBEC system every 18 months. The revised TMI Tech Specs would include similar requirements except that the component operation on an ES actuation is tested quarterly vs. every 18 months and the system flow would be verified on a refueling interval versus every 31 days.

Markup

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At each refueling interval and simultaneously with the test of the emergency loading sequence, a reactor building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.
   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during the component test of the R. B. cooling and isolation system.

   The test will be considered satisfactory if the spray pumps have been successfully started as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

2. Compressed air will be introduced into the spray headers to verify each spray nozzle is unobstructed at least every ten years.

b. <u>Reactor Building Cooling and Isolation Systems</u>

1. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. ~~A test signal will actuate the Reactor Building Emergency Cooling system valves to demonstrate operability of the coolers.~~

2. The test will be considered satisfactory if ~~the valves have completed their expected travel as evidenced by the control board component operating lights and a second means of verification, such as: the station computer, local verification, verification of pressure/flow, or control board component operating lights initiate by separate limit switch contacts.~~ **measured system flow is greater than accident design flow rate.**

III <u>NO SIGNIFICANT HAZARDS CONSIDERATIONS</u>

GPU Nuclear has determined that the requested Technical Specification Change poses no significant hazard consideration as defined by10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The proposed amendment makes administrative corrections, adds conditions to limiting conditions for operation, revises selected time clocks and surveillance requirements consistent with NUREG 1430, and adds a time clock to a unique LCO. These changes have no affect upon the plant design or operation. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes. Therefore, operation in accordance with the proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated, because no new accident initiators would be created.

3. Operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety because no changes to plant operating limits or limiting safety system settings are proposed.

IV <u>IMPLEMENTATION</u>

GPU Nuclear requests that the amendment authorizing these changes become effective within 60 days of issuance.

Enclosure 2
TMI-1 Technical Specification Revised Pages

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is **described in Section 7.1** of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as **described in Section 7.1** of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as **described in Section 7.1** of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 INSTRUMENTATION SURVEILLANCE

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

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

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

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

Specification

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

3.3.1.1 Injection Systems

a. The borated water storage tank shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500-ppm boron at a temperature not less than 40°F. Specification 3.0.1 applies.

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

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

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

e. Two BWST level instrument channels are operable.

f. The two reactor building sump isolation valves (DH-V6A/6B) shall be remote-manually operable. Specification 3.0.1 applies.

3.3.1.2 Core Flooding System

a. Two core flooding tanks each containing 940 ± 30 ft^3 of borated water at 600 ± 25 psig shall be available. Specification 3.0.1 applies.

b. Core flooding tank boron concentration shall not be less than 2,270-ppm boron. **Specification 3.3.2.1 applies.**

c. The electrically operated discharge valves from the core flood tank will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. Specification 3.0.1 applies.

d. One core flood tank pressure instrumentation channel and one core flood tank level instrumentation channel per tank shall be operable. **Specification 3.3.2.1 applies.**

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

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

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

b. The sodium hydroxide (NaOH) tank shall be maintained at 8 ft. ±6 inches lower than the BWST level as measured by the BWST/NaOH tank differential pressure indicator. The NaOH tank concentration shall be 10.0 ±.5 weight percent (%). **Specification 3.3.2.1 applies.**

c. All manual valves in the discharge lines of the sodium hydroxide tank shall be locked open. **Specification 3.3.2.1 applies.**

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.
b. Two nuclear service river water pumps must be OPERABLE.
c. Two decay heat closed cycle cooling water pumps must be OPERABLE.
d. Two decay heat river water pumps must be OPERABLE.
e. Two reactor building emergency cooling river water pumps must be OPERABLE.

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

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

3.3.2.1 If the CFT boron concentration is outside of limits, or the CFT pressure instrumentation, CFT level instrumentation, or NaOH tank is outside of limits, restore the system to operable status within 72 hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	F	
11. (Deleted)		--	--	
12. Pump Flux Comparator	S	M	R	
13. High Reactor Building Pressure Channel	S	M	F	
14. High **Pressure Injection** Logic Channels	NA	Q	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F.
16. Low Pressure Injection Logic Channel	NA	Q	NA	
17. Lower Pressure Injection Analog Channels			0	
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
30. Borated Water Storage Tank Level Indicator	W	NA	F	
31. **DELETED**				
32. **DELETED**				
33. Containment Temperature	NA	NA	F	
34. Incore Neutron Detectors	M(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout when reactor power is greater than 15%
35. Emergency Plant Radiation Instruments	M (1)	NA	F	(1) Battery check.
36. Strong Motion Accelerometer	Q(1)	NA	Q	(1) Battery check.
37. Reactor Building Sump Level	NA	NA	R	

TABLE 4.1-3 Cont'd

	Item	Check	Frequency
4.	Spent Fuel Pool Water Sample	Boron Concentration greater than or equal to 600 ppmb	**Weekly**
5.	Secondary Coolant	Isotopic analysis for DOSE EQUIVALENT I-131 concentration	At least once per 72 hours when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F.
6.	**Deleted**		
7.	Deleted		
8.	Deleted		
9.	Deleted		
10.	Sodium Hydroxide Tank	Concentration	Semi-Annually and after each makeup.
11.	Deleted		
12.	Deleted		

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** Deleted

*** Deleted

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification

4.5.2.1 High Pressure Injection

a. During each refueling interval and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

b. The test will be considered satisfactory if the valves **(MU-V-14A/B & 16A/B/C/D)** have completed their travel and the make-up pumps are running as evidenced by **system flow**. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A/B/C/D shall be conducted only under either of the following conditions:

1) Tavg shall be greater than 332°F.
2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. During each refueling period and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps listed in 4.5.1.1b have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.

c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance, which affects the position indicator.

4.5.2.3 Core Flooding

a. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. Verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flooding tank level verifies that all valves have opened.

4.5.2.4 Component Tests

a. At intervals not to exceed 3 months, the components required for emergency core cooling will be tested.

b. The test will be considered satisfactory if the pumps and fans have been successfully started and the valves have completed their travel as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, verification of pressure/flow, or control board indicating lights initiated by separate limit switch contacts.

Bases

The emergency core cooling systems (Reference 1) are the principal reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

The minimum acceptable HPI/LPI flow assures proper flow and flow split between injection legs.

With the reactor shutdown, the valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

Reference

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

4.5.3 REACTOR BUILDING COOLING AND ISOLATION SYSTEM

Applicability

Applies to testing of the reactor building cooling and isolation systems.

Objective

To verify that the reactor building cooling systems are operable Specification

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At each refueling interval and simultaneously with the test of the emergency loading sequence, a Reactor Building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.

   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during thecomponent test of the Reactor Building Cooling and Isolation System.The test will be considered satisfactory if the spray pumps have been successfully started as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

2. Compressed air will be introduced into the spray headers to verify each spray nozzle is unobstructed at least every ten years.

b. Reactor Building Cooling and Isolation Systems

1. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system.

2. The test will be considered satisfactory **if measured system flow is greater than accident design flow rate**.

Enclosure 3
Certificate of Service for
TMI-1 Technical Specification Change Request No. 262

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF
AMERGEN ENERGY, LLC

DOCKET NO. 50-289
LICENSE NO. DPR-50

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 262 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Darryl LeHew, Chairman
Board of Supervisors of
Londonderry Township of Dauphin County
R. D. #1, Geyers Church Road
Middletown, PA 17057

Ms. Sally S. Klein, Chairman
Board of County Commissioners
Dauphin County Courthouse
Front & Market Streets
Harrisburg, PA 17101

Director, Bureau of Radiation Protection
PA Dept. of Environmental Resources
Rachael Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469
Attn: Mr. Stan Maingi

AMERGEN ENERGY, LLC

BY: John B Cotton
Vice President TMI, Unit 1

DATE: 06/14/00