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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Request No. 286, Transmittal Of Camera-Ready Technical Specification Pages

Three Mile Island, Unit 1 (TMI Unit 1)
Operating License No. DPR-50
NRC Docket No. 50-289

This letter transmits the camera-ready Technical Specification pages to support NRC issuance of an amendment approving TMI Unit 1 License Amendment Request No. 286. The markup revisions to Technical Specification 4.9.1.1, submitted in Attachment 2 to AmerGen letter to the NRC dated December 19, 2001 (5928-01-20298) incorrectly identified the equipment tag number for the turbine-driven emergency feedwater (EFW) pump as "EF-P-1A." As shown in Figure 1 of the above referenced letter, the correct designation for the turbine-driven EFW pump is "EF-P-1." This editorial correction has been incorporated on the attached camera-ready page 4-52. AmerGen Energy Company, LLC (AmerGen) requests a 60-day implementation for this amendment due to the number of plant procedure changes involved.

No new regulatory commitments are established by this submittal. Please contact David J. Distel at (610) 765-5517, if you have any questions regarding this submittal.

I declare under penalty of perjury that the foregoing is true and correct.

Very truly yours,

11-14-02
Executed On

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid Atlantic Regional Operating Group

Enclosure: TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 286

cc: H. J. Miller, USNRC Regional Administrator, Region I
J. D. Orr, USNRC TMI Unit 1 Senior Resident Inspector
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File No. 99064

A001

ENCLOSURE

TMI Unit 1 Technical Specification Revised Pages for

License Amendment Request No. 286

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3.4 DECAY HEAT REMOVAL (DHR) 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 DHR.*

Specification

3.4.1 Reactor Coolant System (RCS) temperature greater than 250 degrees F.

3.4.1.1 Three independent Emergency Feedwater (EFW) Pumps and two redundant flowpaths to each Once Through Steam Generator (OTSG) 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 two OPERABLE main steam supply paths.

(1) With one main steam supply path inoperable, restore the inoperable steam supply path to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 12 hours.

(2) With one EFW Pump or any EFW flowpath inoperable, restore the inoperable pump or flowpath to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours.

(3) With one main steam supply path to the turbine-driven EFW Pump and one motor-driven EFW Pump inoperable, restore the steam supply or the motor-driven EFW Pump to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 12 hours.

(4) With more than one EFW Pump or both flowpaths to either OTSG inoperable, initiate action immediately to restore at least two EFW Pumps and one flowpath to each OTSG:

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

** HSPS operability is specified in Specification 3.5.1. When HSPS is not required to be OPERABLE, EFW is OPERABLE by manual control of pumps and valves from the Control Room.

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY (Continued)

Notes:

1. Specification 3.0.1 and all other actions requiring shutdown or changes in REACTOR OPERATING CONDITIONS are suspended until at least two EFW Pumps and one EFW flowpath to each OTSG are restored to OPERABLE status.

2. While performing surveillance testing, more than one EFW Pump or both flowpaths to a single OTSG may be inoperable for up to 8 hours provided that:

   (a) At least one motor-driven EFW Pump shall remain OPERABLE, and

   (b) With the reactor in STARTUP, HOT STANDBY, or POWER OPERATION, a designated qualified individual who is in communication with the control room shall be continuously stationed in the immediate vicinity of the affected EFW local manual valves. On instruction from the Control Room, the individual shall realign the valves from the test mode to their operational alignment.

b. Four of six Turbine Bypass Valves (TBVs) OPERABLE. With more than two TBVs inoperable, restore operability of at least four TBVs within 72 hours.

c. The Condensate Storage Tanks (CSTs) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST.

   (1) With a CST inoperable, restore the CST to operability within 72 hours or be in HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours.

   (2) With more than one CST inoperable, restore at least one CST to OPERABLE status or be subcritical within 1 hour, in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

3.4.1.2.1 With the Reactor between 250 degrees F and HOT SHUTDOWN, and having been subcritical for at least one (1) hour, two (2) Main Steam Safety Valves (MSSVs) per OTSG shall be OPERABLE. With less than two (2) MSSVs per OTSG OPERABLE, restore at least two (2) MSSVs to OPERABLE status for each OTSG within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

3.4.1.2.2 With the Reactor between HOT SHUTDOWN and 5% power, and having been subcritical for at least one (1) hour, two (2) MSSVs per OTSG shall be OPERABLE provided the overpower trip setpoint in the RPS is set to less than 5% full power. With less than two (2) MSSVs per OTSG OPERABLE, restore at least two (2) MSSVs to OPERABLE status for each OTSG within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY (Continued)

3.4.1.2.3 Except as provided in Specification 3.4.1.2.2 above, when the Reactor is above HOT SHUTDOWN, all eighteen (18) MSSVs shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of MSSVs Disabled on Any OTSG	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than three (3) MSSVs inoperable, restore at least fifteen (15) MSSVs to OPERABLE status within 4 hours or be in HOT SHUTDOWN within the next 6 hours.

3.4.2 RCS temperature less than or equal to 250 degrees F.

3.4.2.1 At least two of the following means for maintaining DHR capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. DHR String (Loop "A").

b. DHR String (Loop "B").

c. RCS Loop "A" and its associated OTSG with an EFW Pump and a flowpath.

d. RCS Loop "B" and its associated OTSG with an EFW Pump and a flowpath.

With less than the above required means for maintaining DHR capability OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

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

3.4.2.3 The number of means for DHR required to be OPERABLE per Specification 3.4.2.1 may be reduced to one provided that the Reactor is in a REFUELING SHUTDOWN condition with the Fuel Transfer Canal water level greater than or equal to 23 feet above the Reactor Vessel flange.

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

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

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

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal DHR is by the OTSGs with the steam dump to the condenser when RCS temperature is above 250 degrees F and by the DHR System below 250 degrees 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 OTSG is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the OTSGs is provided by the main feedwater system.

The Emergency Feedwater (EFW) System supplies adequate feedwater to the OTSGs at accident pressures, removing heat from the Reactor Coolant System (RCS) to support safe shutdown of the reactor when the normal feedwater supply is unavailable. EFW is not required for normal plant startup and shutdown.

The turbine-driven EFW Pump and two motor-driven EFW Pumps take suction from the Condensate Storage Tanks (CSTs) and deliver flow to a common discharge header. Flowpath redundancy is provided for those portions of the EFW flowpath containing active components between the pumps and each of the OTSGs. Each EFW line to an OTSG includes two redundant flowpaths, each equipped with an automatic control valve (EF-V-30A/B/C/D) and a manual isolation valve (EF-V-52A/B/C/D). Each redundant flowpath is capable of providing adequate flow to the associated OTSG. Heat removed from the OTSGs returns to the Main Condenser through the Turbine Bypass Valves (TBVs) or discharges to the atmosphere through the Main Steam Safety Valves (MSSVs) and/or the Atmospheric Dump Valves (ADVs). An unlimited supply of river water to the EFW Pumps is available using either of the two Reactor Building Emergency Cooling Water (Reactor River Water) Pumps (RR-P-1A/B).

Redundant main steam supply paths are provided to the turbine-driven EFW Pump for certain events involving loss of one steam supply (e.g., main steam and feedwater line breaks). An operable Main Steam supply path delivers steam to the turbine-driven EFW Pump upon HSPS actuation or by operator action from the control room when HSPS is not required. During low pressure conditions, additional steam supply paths from Main Steam (MS-V-10A/B) or Auxiliary Steam can be made available to the turbine-driven EFW Pump as necessary.

During design basis events the EFW System can withstand any single active failure and still perform its function. The limiting design basis accident for the EFW System is a loss of feedwater event with off-site power available. In the event of a loss of all AC power, which assumes multiple single failures, the turbine-driven EFW Pump alone delivers the necessary EFW flow. Consideration of additional failures in the EFW System or Heat Sink Protection System (HSPS) is not required for this event. Additionally, the EFW System capabilities are sufficient to deliver the required flow in licensing basis events (e.g., ATWS failure to trip events, Generic Letter 81-14 seismic events, and the Station Blackout event).

The most limiting EFW flow requirement is met when at least two EFW Pumps are operable and at least one EFW flowpath to each OTSG is operable. When three pumps and two flowpaths to each OTSG are operable, the EFW System can withstand any single active failure. Examples of single active failures include: failure of any one EFW Pump to actuate, failure of one HSPS train to actuate, or failure of one redundant flowpath to either OTSG. Initially after a shutdown, any two EFW Pumps are required to remove RCS heat with one pump eventually sufficing as the decay heat production rate diminishes.

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY (Continued)

Bases (Continued)

If EFW were required during surveillance testing, minor operator action (e.g., opening a local isolation valve or manipulating a control switch from the control room) may be needed to restore operability of the required pumps or flowpaths. An exception to permit more than one EFW Pump or both EFW flowpaths to a single OTSG to be inoperable for up to 8 hours during surveillance testing requires 1) at least one motor-driven EFW Pump operable, and 2) an individual involved in the task of testing the EFW System must be in communication with the control room and stationed in the immediate vicinity of the affected EFW flowpath valves. Thus the individual is permitted to be involved in the test activities by taking test data and his movement is restricted to the area of the EFW Pump and valve rooms where the testing is being conducted.

The allowed action times are reasonable, based on operating experience, to reach the required plant operating conditions from full power in an orderly manner and without challenging plant systems. Without at least two EFW Pumps and one EFW flowpath to each OTSG operable, the required action is to immediately restore EFW components to operable status, and all actions requiring shutdown or changes in Reactor Operating Condition are suspended. With less than two EFW pumps or no flowpath to either OTSG operable, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, which might result in a trip. The seriousness of this condition requires that action be started immediately to restore EFW components to operable status. TS 3.0.1 is not applicable, as it could force the unit into a less safe condition.

The EFW system actuates on: 1) loss of all four Reactor Coolant Pumps, 2) loss of both Main Feedwater Pumps, 3) low OTSG water level, or 4) high Reactor Building pressure. A single active failure in the HSPS will neither inadvertently initiate the EFW system nor isolate the Main Feedwater system. OTSG water level is controlled automatically by the HSPS system or can be controlled manually, if necessary.

The MSSVs will be able to relieve to atmosphere the total steam flow if necessary. Below 5% power, only a minimum number of MSSVs need to be operable as stated in Specifications 3.4.1.2.1 and 3.4.1.2.2. This is to provide OTSG overpressure protection during hot functional testing and low power physics testing. Additionally, when the Reactor is between hot shutdown and 5% full power operation, the overpower trip setpoint in the RPS shall be set to less than 5% as is specified in Specification 3.4.1.2.2. The minimum number of MSSVs required to be operable allows margin for testing without jeopardizing plant safety. Plant specific analysis shows that one MSSV is sufficient to relieve reactor coolant pump heat and stored energy when the reactor has been subcritical by 1% delta K/K for at least one hour. Other plant analyses show that two (2) MSSVs on either OTSG are more than sufficient to relieve reactor coolant pump heat and stored energy when the reactor is below 5% full power operation but had been subcritical by 1% delta K/K for at least one hour subsequent to power operation above 5% full power. According to Specification 3.1.1.2a, both OTSGs shall be operable whenever the reactor coolant average temperature is above 250 degrees F. This assures that all four (4) MSSVs are available for redundancy. During power operations at 5% full power or above, if MSSVs are inoperable, the power level must be reduced, as stated in Specification 3.4.1.2.3 such that the remaining MSSVs can prevent overpressure on a turbine trip.

Bases (Continued)

The minimum amount of water in the CSTs required by Specification 3.4.1.1.c, provides at least 12 hours of DHR with steam being discharged to the atmosphere. This provides adequate time to align alternate water sources for RCS cooldown. After cooling to 250 degrees F, the DHR System is used to achieve further cooling.

When the RCS temperature is below 250 degrees F, a single DHR String (Loop), or single OTSG with an EFW Pump and a flowpath capable of supporting natural circulation is sufficient to provide removal of decay heat at all times following the cooldown to 250 degrees F. The DHR String (Loop) redundancy required by Specification 3.4.2.1 is achieved with independent active components capable of maintaining the RCS subcooled. A single DHR flowpath with redundant active components is sufficient to meet the requirements of Specifications 3.4.2.1.a and 3.4.2.1.b. The requirement to maintain two operable means of DHR ensures that a single active failure does not result in a complete loss of DHR capability. The requirement to keep a DHR Loop in operation as necessary to maintain the RCS subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

With the Reactor Vessel head removed and 23 feet of water above the Reactor Vessel flange, a large heat sink is available for core cooling. In this condition, only one DHR Loop is required to be operable because the volume of water above the Reactor Vessel flange provides a large heat sink which would allow sufficient time to recover active DHR means.

Following extensive outages or major core off-loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists (Reference 4).

REFERENCES

(1) UFSAR, Table 6.1-4 - ECCS "Single Failure Analysis"

(2) UFSAR, Section 9.5 - "Decay Heat Removal System"

(3) UFSAR, Section 10.6 - "Emergency Feedwater System"

(4) TMI Unit 1 Calculation C-3320-85-001, "RCS Decay Heat Removal-Ambient Losses," Revision 0, February 28, 1985

3.5.5 ACCIDENT MONITORING INSTRUMENTATION (Continued)

The Emergency Feedwater System (EFW) is provided with two channels of flow instrumentation on each of the two discharge lines. Local flow indication is also available for the EFW System.

Although the pressurizer has multiple level indications, the separate indications are selectable via a switch for display on a single display. Pressurizer level, however, can also be determined via the patch panel and the computer log. In addition, a second channel of pressurizer level indication is available independent of the NNI.

Although the instruments identified in Table 3.5-2 are significant in diagnosing situations which could lead to inadequate core cooling, loss of any one of the instruments in Table 3.5-2 would not prevent continued, safe, reactor operation. Therefore, operation is justified for up to 7 days (48 hours for pressurizer level). Alternate indications are available for Saturation Margin Monitors using hand calculations, the PORV/Safety Valve position monitors using discharge line thermocouple and Reactor Coolant Drain Tank indications, and for EFW flow using Steam Generator level and EFW Pump discharge pressure. Pressurizer level has two channels, one channel from NNI (2 D/P instrument strings through a single indicator) and one channel independent of the NNI. Operation with the above pressurizer level channels out of service is permitted for up to 48 hours. Alternate indication would be available through the plant computer.

The operability of design basis accident monitoring instrumentation as identified in Table 3.5-3, ensures that sufficient information is available on selected plant parameters to monitor and assess the variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Rev. 3, May 1983.) These instruments will be maintained for that purpose.

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TABLE 3.5-2

ACCIDENT MONITORING INSTRUMENTS

FUNCTION	INSTRUMENTS	NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS
1	Saturation Margin Monitor	2	1
2	Safety Valve Differential Pressure Monitor	1 per discharge line	1 per discharge line
3	PORV Position Monitor	2	1*
4	Emergency Feedwater Flow	2 per OTSG	1 per OTSG
5	Pressurizer Level	2	1
6	Backup Incore Thermocouple Display Channel	4 thermocouples/core quadrant	2 thermocouples/core quadrant

* With the PORV Block Valve closed in accordance with Specification 3.1.12.4.a, the minimum number of channels is zero.

4.9 DECAY HEAT REMOVAL (DHR) 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 DHR are capable of performing their design function.

Specification

4.9.1 Reactor Coolant System (RCS) Temperature greater than 250 degrees F.

4.9.1.1 Verify each Emergency Feedwater (EFW) Pump is tested in accordance with the requirements and acceptance criteria of the ASME Section XI Inservice Test Program.

Note: This surveillance is not required to be performed for the turbine-driven EFW Pump (EF-P-1) until 24 hours after exceeding 750 psig.

4.9.1.2 DELETED

4.9.1.3 At least once per 31 days, each EFW System flowpath valve from both Condensate Storage Tanks (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 STARTUP, 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 CSTs to the Steam Generators.

4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY-PERIODIC TESTING (Continued)

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 RCS Temperature less than or equal to 250 degrees F.*

4.9.2.1 On a daily basis, verify operability of the means for DHR required by Specification 3.4.2 by observation of console status indication.

* These requirements supplement the requirements of Specifications 4.5.2.2 and 4.5.4.

Bases

ASME Section XI specifies requirements and acceptance standards for the testing of nuclear safety related pumps. The quarterly EFW Pump test frequency specified by the ASME Section XI Code will be sufficient to verify that the turbine-driven and both motor-driven EFW Pumps are operable. 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.

Deferral of the requirement to perform IST on the turbine-driven EFW Pump is necessary to assure sufficient OTSG pressure to perform the test using Main Steam.

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