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

SUBJECT: LICENSE AMENDMENT REQUEST (LAR) NO. 292, TRANSMITTAL OF CAMERA-READY TECHNICAL SPECIFICATION PAGES

THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
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. 292, which was submitted to the NRC on September 20, 2000.

If you have any questions or require additional information, please do not hesitate to contact us.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 24, 2002.

Very truly yours,

Michael P. Gallagher

Michael P. Gallagher
Director - Licensing and Regulatory Affairs
Mid-Atlantic Regional Operating Group

Enclosure: TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 292

cc: H. J. Miller, USNRC Regional Administrator, Region I
J. D. Orr, USNRC TMI Unit 1 Resident Inspector
T. G. Colburn, USNRC TMI Unit 1 Senior Project Manager
File No. 00075

A001

ENCLOSURE

TMI Unit 1 Technical Specification Revised Pages for

License Amendment Request No. 292

(Pages 3-40b, 3-40d, 3-41, 3-41c, 4-10a, 4-10b, 4-38, and 4-55c)

The Emergency Feedwater System is provided with two channels of flow instrumentation on each of the two discharge lines. Local flow indication is also available for the emergency feedwater 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 (3 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.

TABLE 3.5-3

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	REQUIRED NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS	ACTION
1.	High Range Noble Gas Effluent			
	a. Condenser Vacuum Pump Exhaust (RM-A5-Hi)	1	1	A
	b. Condenser Vacuum Pump Exhaust (RM-G25)	1	1	A
	c. Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)	1	1	A
	d. Reactor Building Purge Exhaust (RM-A9-Hi)	1	1	A
	e. Reactor Building Purge Exhaust (RM-G24)	1	1	A
	f. Main Steam Lines Radiation (RM-G26/RM-G27)	1 each OTSG	1 each OTSG	A
2.	Containment High Range Radiation (RM-G22/G-23)	2	2	A
3.	Containment Pressure	2	1	B
4.	Containment Water Level			
	a. Containment Flood (LT-806/807)	2	1	B
	b. Containment Sump (LT-804/805)	1	0	C
5.	DELETED			
6.	Wide Range Neutron Flux	2	1	A
7.	Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)	2	1	A
8.	Reactor Coolant System Hot Leg Water Temperature (TE-958, 960; TI-958A, 960A)	2	1	A
9.	Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)	2	1	A
10.	Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)	2/OTSG	1/OTSG	A
11.	Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)	2/Tank	1/Tank	A

3.6 REACTOR BUILDING

Applicability

Applies to the containment integrity of the reactor building as specified below.

Objective

To assure containment integrity.

Specification

3.6.1 Except as provided in 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (Section 1.7) shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

3.6.2 Containment integrity shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% $\Delta k/k$ shall not be made by control rod motion or boron dilution unless containment integrity is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and are conspicuously marked.

3.6.6 While the reactor is critical, if a reactor building isolation valve (other than a purge valve) is determined to be inoperable in a position other than the required position, the other reactor building isolation valve in the line shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, the OPERABLE valve will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within additional 30 hours.

3.6.7 DELETED

Bases

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30° open position.

Allowing purge operations during STARTUP, HOT STANDBY and POWER OPERATION (T.S. 3.6.10) is more beneficial than requiring a cooldown to COLD SHUTDOWN from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

TABLE 4.1-4

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	CHECK	TEST	CALIBRATE	REMARKS
1.	Noble Gas Effluent				
	a. Condenser Vacuum Pump Exhaust (RM-A5-Hi)	W	M	F	(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that this monitor is functioning.
	b. Condenser Vacuum Pump Exhaust (RM-G25)	W(1)	M	F	
	c. Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)	W	M	F	
	d. Reactor Building Purge Exhaust (RM-A9-Hi)	W	M	F	
	e. Reactor Building Purge Exhaust (RM-G24)	W(1)	M	F	
	f. Main Steam Lines Radiation (RM-G26/RM-G27)	W(1)	M	F	
2.	Containment High Range Radiation (RM-G22/G23)	W	M	R	
3.	Containment Pressure	W	N/A	F	
4.	Containment Water Level	W	N/A	R	
5.	DELETED				
6.	Wide Range Neutron Flux	W	N/A	F	

TABLE 4.1-4 (Continued)

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	CHECK	TEST	CALIBRATE	REMARKS
7.	Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)	W	N/A	R	
8.	Reactor Coolant System Hot Leg (TE-958, 960; TI-958A, 960A)	W	N/A	R	
9.	Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)	W	N/A	R	
10.	Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)	W	N/A	R	
11.	Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)	W	N/A	F	

(Page 4-38 deleted)
(Page 4-38a deleted)

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once every refueling interval to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with Halogenated hydrocarbon refrigerant shall be performed in accordance with approved test procedures. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable all adsorbent in the system should be replaced with an adsorbent qualified according to ASTM D3803-1989. Tests of the HEPA filters with DOP aerosol shall also be performed in accordance with approved test procedures. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Guide 1.52, March 1978.

Fans AH-E7A & B performance verification is necessary to ensure adequate flow to perform the filter surveillance of T.S. 4.12.2.1 and 4.12.2.3 and can only be demonstrated by running both fans simultaneously. This can only be accomplished when purge valves are note limited to 30° open (i.e., cold shutdown).

The reactor building purge exhaust system no longer is relied upon to serve an operating accident mitigating (i.e. LOCA) function. The retest requirement of T.S. 4.12.2.2a has therefore been changed to reflect the same retest requirements as the auxiliary and fuel handling building ventilation system which similarly serves no operating accident mitigating function.

If significant painting, steam, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the Vice President-TMI Unit 1.

References

(1) UFSAR, Section 5.6 - "Ventilation and Purge Systems"