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

Subject: License Amendment Request Nos. 249 & 311, Transmittal of Camera - Ready Technical Specification Pages

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

This letter transmits the camera-ready Technical Specification pages to support NRC issuance of amendments approving TMI Unit 1 License Amendment Request Nos. 249 and 311. Enclosure 1 provides the camera-ready Technical Specification pages for License Amendment Request No. 249. Enclosure 2 provides the camera-ready Technical Specification pages for License Amendment Request No. 311.

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.

Very truly yours,

9-25-01
Executed On

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

Enclosure 1: TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 249

Enclosure 2: TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 311

cc: H. J. Miller, USNRC Regional Administrator, Region I
T. G. Colburn, USNRC Senior Project Manager, TMI Unit 1
J. D. Orr, USNRC Senior Resident Inspector, TMI Unit 1
File No. 00109

A001

ENCLOSURE 1

TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 249

(Page 3-44, 3-45, 3-45a, and 4-29)

3.8 FUEL LOADING AND REFUELING

Applicability: Applies to fuel loading and refueling operations.

Objective: To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the Reactor Building refueling area shall be monitored by RM-G6 and RM-G7. Radiation levels in the spent fuel storage area shall be monitored by RM-G9. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one decay heat removal pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the Reactor Building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the Reactor Building at least one door in each of the personnel and emergency air locks shall be capable of being closed.* The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or capable of being closed,* or

2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

*Administrative controls shall ensure that appropriate personnel are aware that air lock doors and/or other penetrations are open, a specific individual(s) is designated and available to close the air lock doors and other penetrations as part of a required evacuation of containment. Any obstruction(s) (e.g., cable and hoses) that could prevent closure of an air lock door or other penetration will be capable of being quickly removed.

3.8.8 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.9 The reactor building purge system, including the radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than one week prior to refueling operations.

3.8.10 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

3.8.11 During the handling of irradiated fuel in the Reactor Building at least 23 feet of water shall be maintained above the level of the reactor pressure vessel flange. If the water level is less than 23 feet above the reactor pressure vessel flange, place the fuel assembly(s) being handled into a safe position, then cease fuel handling until the water level has been restored to 23 feet or greater above the reactor pressure vessel flange.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the UFSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration. The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core (Reference 1). The boron concentration will be sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, however only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

Per Specification 3.8.6 and 3.8.7, the personnel and emergency air lock doors, and penetrations may be open during movement of irradiated fuel in the containment provided a minimum of one door in each of the air locks, and penetrations are capable of being closed in the event of a fuel handling accident, and the plant is in REFUELING SHUTDOWN or REFUELING OPERATION with at least 23 feet of water above the fuel seated within the reactor pressure vessel. The minimum water level specified is the basis for the accident analysis assumption of a decontamination factor of 200 for the release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel. Should a fuel handling accident occur inside containment, a minimum of one door in each personnel and emergency air lock, and the open penetrations will be closed following an evacuation of containment. Administrative controls will be in place to assure closure of at least one door in each air lock, as well as other open containment penetrations, following a containment evacuation.

Provisions for equivalent isolation methods in Technical Specification 3.8.7 include use of a material (e.g. temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for other containment penetrations during fuel movements.

The specification requiring testing Reactor Building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.10 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours (Reference 2).

<u>REFERENCES</u>

(1) UFSAR, Section 14.2.2.1- "Fuel Handling Accident"

(2) UFSAR, Section 14.2.2.1(2)- "FHA Inside Containment"

4.4 REACTOR BUILDING

4.4.1 CONTAINMENT LEAKAGE TESTS

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Testing (ILRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program at test frequencies established in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2 Local Leakage Rate Testing (LLRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program. LLRT shall be performed at a pressure not less than peak accident pressure P_{ac} with the exception that the airlock door seal tests shall normally be performed at 10 psig and the periodic containment airlock tests shall be performed at a pressure not less than P_{ac}. LLRT frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Operability of the personnel and emergency air lock door interlocks and the associated control room annunciator circuits shall be determined at least once per six months. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable, except as provided in Technical Specification Section 3.8.6.

Bases (1)

The Reactor Building is designed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design internal pressure of 55 psig with a coincident temperature of 281°F at accident conditions. The peak calculated Reactor Building pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig. The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}. Containment Isolation Valves are addressed in the UFSAR (Reference 2).

ENCLOSURE 2
TMI Unit 1 Technical Specification Revised Pages for License Amendment Request No. 311

(Pages 2-3, 2-4a, and 2-4c)

The specified flow rates for curves 1, 2, and 3 of the Axial Power Imbalance Protective Limits given in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent, (BAW-2), or 26 percent (BWC) whichever condition is more restrictive. The curves of Figures 2.1-1 and 2.1-3 were developed assuming a reactor coolant design flow rate of 102% of 352,000 gpm. However, a higher minimum flow rate (105.5% of 352,000 gpm) is specified in order to offset transition core effects due to the introduction of the Mark-B12 fuel design with fine mesh debris filter.

The maximum thermal power for each reactor coolant pump operating condition (four pump, three pump, and one pump in each loop) given in the COLR is due to a power level trip produced by the flux-flow ratio multiplied by the minimum flow rate for the given pump combination plus the maximum calibration and instrumentation error.

Using a local quality limit of 22 percent (BAW-2), or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or BWC correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

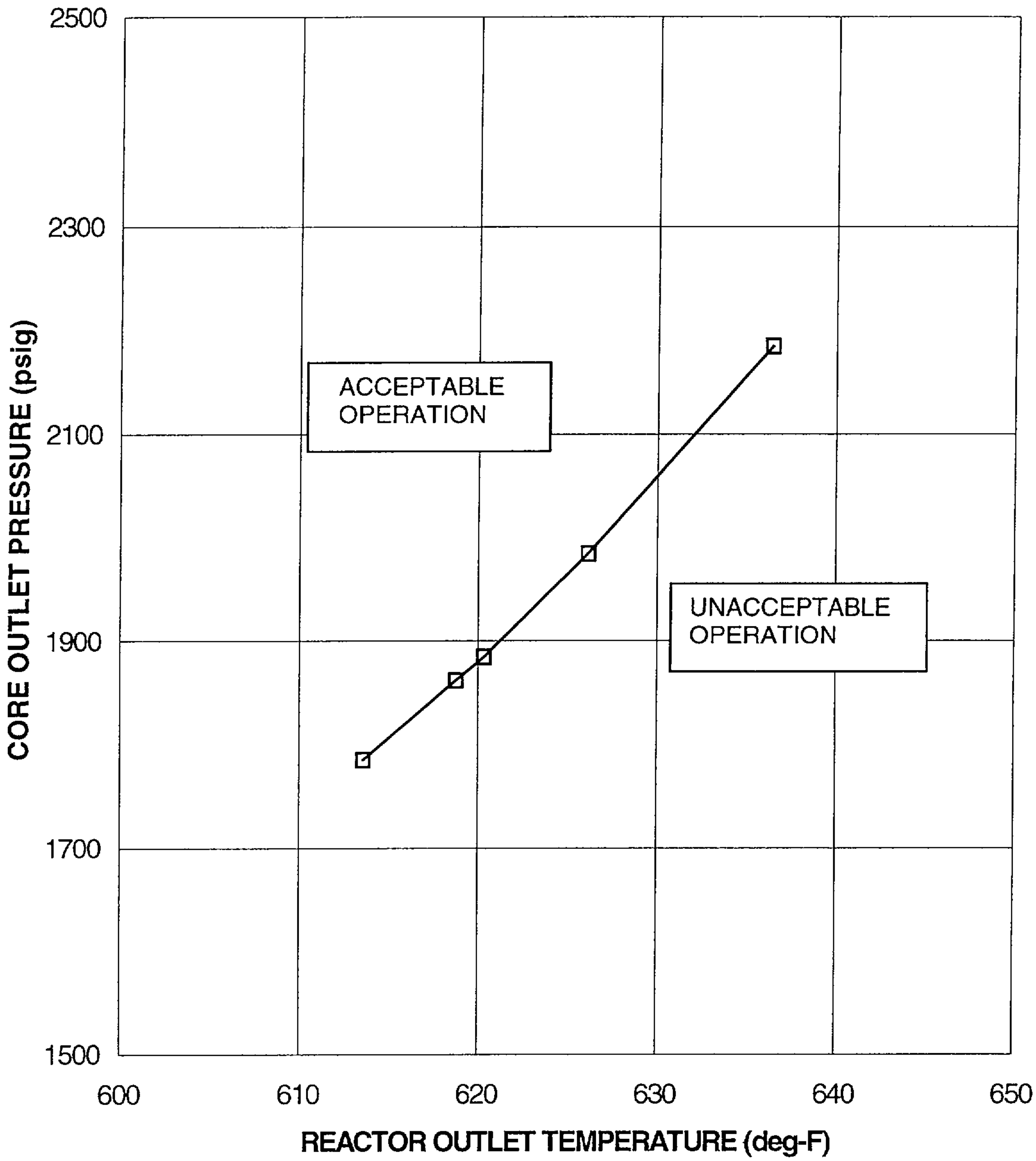
For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2), or 26 percent (BWC) for the particular reactor coolant pump situation. Curve 1 is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

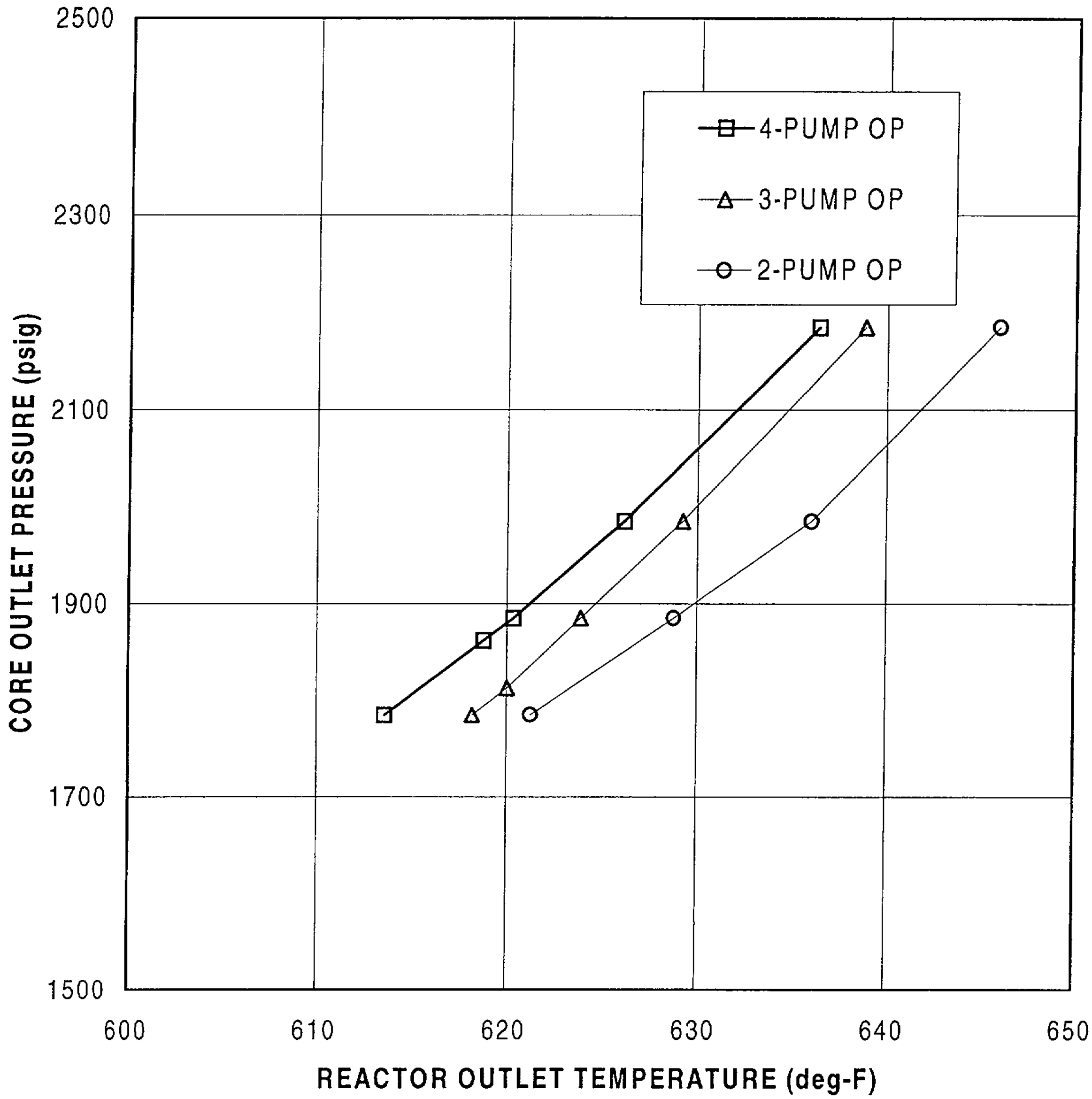
(1) UFSAR, Section 3.2.3.1.1 - "Fuel Assembly Heat Transfer Design"

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, Babcock & Wilcox, Lynchburg, Virginia, April 1985

(3) UFSAR, Section 3.2.3.1.1.3 - "Nuclear Power Factors"



CORE PROTECTION SAFETY LIMIT
TMI-1
FIGURE 2.1-1



RC Pumps	Reactor Coolant Flow (lbs/hr)	Power	Pumps Operating (Type of Limit)
4	$138.52X10^6$	112%	Four Pumps (DNBR Limit)
3	See COLR	See COLR	Three Pumps (DNBR Limit)
2	See COLR	See COLR	One Pump in Each Loop (Quality Limit)

CORE PROTECTION SAFETY BASES
TMI-1
FIGURE 2.1-3