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March 23, 2001
5928-01-20085

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

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
LICENSE AMENDMENT REQUEST NO. 301, TRANSMITTAL OF
CAMERA-READY TECHNICAL SPECIFICATION PAGES

This letter transmits the camera-ready Technical Specification pages to support NRC issuance of an amendment approving TMI Unit 1 License Amendment Request No. 301.

Please contact David J. Distel at (610) 765-5517, if you have any questions regarding this submittal.

Very truly yours,



G. B. Rombold
Manager, TMI Unit 1 Licensing

GBR/djd

Enclosure: TMI Unit 1 Technical Specification Revised Pages for License
Amendment Request No. 301

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. 00097

ADD 1

ENCLOSURE 1

**TMI Unit 1 Technical Specification Revised Pages for
License Amendment Request No. 301**

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, axial power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

- 2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.
- 2.1.2 The combination of reactor thermal power and axial power imbalance (power in the top half of core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the protective limit as defined by the locus of points (solid line) for the specified flow set forth in the Axial Power Imbalance Protective Limits given in the Core Operating Limits Report (COLR). If the actual-reactor-thermal-power/axial-power-imbalance point is above the line for the specified flow, the protective limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed, departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in excessive cladding temperature and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Reference 1) and BWC (Reference 2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel with inconel intermediate spacer grids and the BWC correlation applies to Mark-B fuel with zircaloy or M5 intermediate spacer grids (non-mixing vane). The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal

5.3 REACTOR

Applicability

Applies to the design features of the reactor core and reactor coolant system.

Objective

To define the significant design features of the reactor core and reactor coolant system.

Specification

5.3.1 REACTOR CORE

- 5.3.1.1 A fuel assembly normally contains 208 fuel rods arranged in a 15 by 15 lattice. The reactor shall contain 177 fuel assemblies. Fuel rods shall be clad with zircaloy, ZIRLO, or zirconium-based M5 alloy materials and contain an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. The details of the fuel assembly design are described in TMI-1 UFSAR Chapter 3.
- 5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches. The active fuel height is defined in TMI-1 UFSAR Chapter 3.
- 5.3.1.3 The core average and individual batch enrichments for the present cycle are described in TMI-1 UFSAR Chapter 3.
- 5.3.1.4 The control rod assemblies (CRA) and axial power shaping rod assemblies (APSRA) are distributed in the reactor core as shown in TMI-1 FSAR Chapter 3. The CRA and APSRA design data are also described in the UFSAR.
- 5.3.1.5 The TMI-1 core may contain burnable poison rod assemblies (BPRA) and gadolinia-urania integral burnable poison fuel pellets as described in TMI-1 UFSAR Chapter 3.
- 5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation data described in the UFSAR. Enrichment shall not exceed a nominal 5.0 weight percent of U_{235} .

5.3.2 REACTOR COOLANT SYSTEM

- 5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements. (Refer to UFSAR Chapter 4 for details of design and operation.)

6.9.5 CORE OPERATING LIMITS REPORT

- 6.9.5.1 The core operating limits addressed by the individual Technical Specifications shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle.
- 6.9.5.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at TMI-1, specifically:
- (1) BAW-10179 P-A, "Safety and Methodology for Acceptable Cycle Reload Analyses." The current revision level shall be specified in the COLR.
 - (2) TR-078-A, "TMI-1 Transient Analyses Using the RETRAN Computer Code", Revision 0. NRC SER dated 2/10/97.
 - (3) TR-087-A, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code", Revision 0. NRC SER dated 12/19/96.
 - (4) TR-091-A, "Steady State Reactor Physics Methodology for TMI-1", Revision 0. NRC SER dated 2/21/96.
 - (5) TR-092P-A, "TMI-1 Reload Design and Setpoint Methodology", Revision 0. NRC SER dated 4/22/97.
 - (6) BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", NRC SER dated February 4, 2000.
- 6.9.5.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits; nuclear limits such as shutdown margin, and transient/accident analysis limits) of the safety analysis are met.
- 6.9.5.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.