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Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
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Dear Mr. Hukill:

On April 20, 1981, the Commission issued an Order for Modification of License concerning primary coolant system pressure isolation valves. This Order revised the Technical Specifications (TSs) for Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1. Subsequent to the issuance of this Order, on August 3, 1981, the Commission issued Amendment No. 71 to the license. However, TS page 4-12 of Amendment No. 71, inadvertently did not incorporate TS 4.2.6 which had been issued by the April 20, 1981, Order. An updated TS page 4-12 is enclosed.

Likewise, when Amendment No. 100 was issued on October 1, 1984, TS page 4-1 inadvertently omitted a change which had been made by Amendment No. 99 issued on August 8, 1984. An updated TS page 4-1 is also enclosed.

Please incorporate updated TS pages 4-12 and 4-1 into your TSs.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
TS pages 4-12 and 4-1

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL
RIngram	JVan Vliet;cf	JStolz
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4. SURVEILLANCE STANDARDS

During Reactor Operational Conditions for which a Limiting Condition for Operation does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/component operable, the associated surveillance requirement must be current. The above applicability requirements assure the operability of systems/components for all Reactor Operating Conditions when required by the Limiting Conditions for Operation.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system and engineered safety feature protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Each post accident monitoring intrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies shown in Table 4.1-4.

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annuciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Amendment No. 46, 99, 100
(Corrected February 11, 1985)

4.2.5 The licensee shall submit a report or application for license amendment to the NRC within 90 days after any time that Crystal River Unit No. 3 fails to maintain a cumulative reactor utilization factor of at least 65%.

The report shall provide justification for continued operation of TMI-1 with the reactor vessel surveillance program conducted at Crystal River Unit No. 3, or the application for license amendment shall propose an alternate program for conduct of the TMI-1 reactor vessel surveillance program.

For the purpose of this technical specification, the definition of commercial operation is that given in Regulatory Guide 1.16, Revision 4. The definition of cumulative reactor utilization factor is:

Cumulative reactor utilization factor - (Cumulative megawatt hours (thermal) since attainment of commercial operation at 100% power x (100)) divided by (licensed power (MWt) x (Cumulative hours since attainment of commercial operation at 100% power)).

4.2.6 In addition to the reports required by Specification 4.2.4, a report shall be submitted to the NRC prior to September 1, 1982, which summarizes the first five years of operating experience with the TMI-1 integrated surveillance program performed at a host reactor. If, at the time of submission of this report, it is desired to continue the surveillance program at a host reactor, such continuation shall be justified on the basis of the attained operating experience.

4.2.7 A surveillance program for the pressure isolation valves between the primary coolant system and the low pressure injection system shall be as follows:

1. Periodic leakage testing[(a)] at test differential pressure greater than 150 psid shall be accomplished for the valves listed in Table 3.1.6.1 for the following conditions:
   (a) prior to achieving hot shutdown after returning the valve to service following maintenance repair or replacement work, and

   (b) prior to achieving hot shutdown following a cold shutdown of greater than 72 hours duration unless testing has been performed within the previous 9 months.
2. Whenever integrity of a pressure isolation valve listed in Table 3.1.6.1 cannot be demonstrated, the integrity of the other remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Bases

a. Specifications 4.2.1 and 2 ensure that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a(g). Relief from any of the above requirements has been provided in writing by the NRC and is not a part of these technical specifications.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.