METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 207

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY: MJ Ross for
Vice President and Director, TMI-1

Sworn and Subscribed to before me
this 19th day of March, 1993.

Erin M. Flowers
Notary Public

Notarial Seal
Erin M. Flowers, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Sept. 11, 1993

Member, Pennsylvania Association of Notaries

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF
GPU NUCLEAR CORPORATION

DOCKET NO. 50-289
LICENSE NO. DPR-50

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 207 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Daryl LeHew, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, PA 17057

Mr. Russell L. Sheaffer, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

Director, Bureau of Radiaton Protection
PA Dept. of Environmental Resources
P.O. Box 2063
Harrisburg, PA 17120
Attn: Mr. Richard Janati

GPU NUCLEAR CORPORATION

BY: MJ Ross for
Vice President and Director, TMI-I

DATE: March 19, 1993

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) No. 207

GPUN requests that the following changed replacement pages be inserted into the existing Technical Specification:

Revised pages: 3-3, 3-4, 3-5, Figures 3.1-1 and 3.1-2

These pages are attached to this change request.

II. REASON FOR CHANGE

This change is submitted in compliance with T.S. Section Nos. 3.1.2.4 and 3.1.2.5, which require an update of the operating P/T limits (Figures 3.1-1 and 3.1-2) in accordance with 10 CFR 50, Appendix G, Section V.B prior to exceeding 10 EFPY of operation.

TMI-1 expects to reach 10 EFPY of operation coincident with its Cycle 10 refueling shutdown towards the end of 1993.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The proposed Technical Specification revises the period of applicability of the present provisions, including Figures No. 3.1-1 and 3.1-2, regarding the reactor vessel pressurization heatup and cooldown limitations and inservice leak and hydrostatic testing, contained in Section 3.1.2, and LTOP provisions, contained in Sections 3.1.12 and 4.5.2.1.c.1. These pressure limits are established to provide protection for the reactor vessel from potential brittle fracture during normal plant operating modes.

The embrittlement trend of the reactor vessel materials is established in accordance with the requirements of 10 CFR 50, App. G. The pressure/temperature limits are calculated, on the basis of the projected state of embrittlement (RT_{ndt}) for the period of applicability, in accordance with the requirements of ASME Code, Section III, App. G.

The basis for this change is that sufficient surveillance capsule data is available in order to establish a better prediction of a bounding embrittlement trend of the TMI-1 reactor vessel beltline materials (welds), in accordance with the provisions of NRC Regulatory Guide No. 1.99 Rev. 2, paragraph C.2.1, as clarified below, versus the more conservative results obtained by application of paragraph C.1.1.

The evaluation (Technical Specification Section 3.1.2 Reference 6) of the B&W Reactor Vessel Owners Group (B&WOG) surveillance data in accordance with the provisions of R.G. 1.99 Rev. 2, paragraph C.2.1, utilized a method which is described and justified in the evaluation, and is labeled as Position No. 1 Alternate method to yield upper bound trends for the RT_{NDT} of the TMI-1 beltline weld materials. This Position No. 1 Alternate method yields virtually the same result as strict adherence with the method described in R.G. 1.99 Rev. 2, paragraph C.2.1 for weld wire 299L44. Additionally, the Position No. 1 Alternate method provides a better bounding trend of weld wire 72105 and can also be applied to Linde-

80 weld wires for which no surveillance data exits. Application of the resulting trend to the TMI-1 reactor vessel as a function of predicted fluence, for the current fuel loading plan, at the critical weld locations, results in a conservative prediction of the time at which the RT_{NDT} at the controlling weld is equal to the RT_{NDT} value which was utilized as the basis for the existing provisions of the Technical Specification. The RT_{NDT} trend of the beltline plate materials were conservatively calculated on the basis of the Regulatory Guide 1.99 Rev. 2, paragraph C.1.1, and the results indicated that the plate material trend is bounded by the predicted RT_{NDT} trend of the vessel weld metals. The TMI-1 reactor vessel limiting material remains the SA-1526 longitudinal beltline weld for which the 1/4T RT_{NDT} value at 32 EFPY is equal to the 186°F which had been previously predicted, on the basis of NRC R.G. 1.99 Rev. 2, paragraph C.1.1, to occur at 10 EFPY.

The RT_{NDT} is an index by which the material fracture toughness property is obtained for input to the fracture mechanics analysis. Having a predictive trend of the RT_{NDT} as a function of applied fluence which reduces excessive conservatism yields correspondingly higher allowable fluence levels at which the same RT_{NDT} values occur. Thus, this Technical Specification change removes excessive conservatism in predicting the RT_{NDT} of the TMI-1 reactor vessel welds in accordance with the provisions of NRC R.G. 1.99 Rev. 2, paragraph C.2.1 and Position 1 Alternate method described and justified in Technical Specification Section 3.1.2 Reference 6. Since the value of RT_{NDT} which is utilized as the input for the material toughness property has not been increased by this change, nor is there a change in any other input to the fracture mechanics analysis, there is no impact on the resulting pressure/temperature (P/T) limits. Such P/T limits had previously been calculated to establish the 10 EFPY plant heatup and cooldown and hydrostatic test limit curves, ie: Figures No. 3.1-1 and 3.1-2, as well as to establish the provisions for low temperature overpressurization protection.

<u>Upper Shelf Energy (C_v-USE):</u>

Although not required for this Technical Specification Change Request, a discussion concerning the Upper Shelf Energy is provided in this section so that the Staff will have complete information regarding the TMI-1 reactor vessel embrittlement issue.

The upper-shelf energy criteria in 10 CFR 50, Appendix G, Section IV.A.1, requires that the vessel beltline materials must maintain an upper shelf energy of no less than 50 Ft-Lbs throughout the life of the vessel, unless acceptable margin of safety is demonstrated if the upper-shelf energy of vessel materials are expected to fall below 50 ft-lbs. In response to NRC Generic Letter No. 92-01, as well as other documentation, it has been identified that some of the Linde-80 welds in the TMI-1 reactor vessel will fall below this screening criteria. If the 50 Ft-Lb criteria can not be met, then the reactor vessel may continue to be operated provided the requirements of Section IV.C are satisfied. TMI-1 has already performed its 10 year ISI of the reactor vessel and found no unacceptable flaws in beltline materials which do not meet the upper shelf energy criteria. Thus, the examination requirements of App. G. Section V.C.1 have been

satisfied. A plant specific analysis to document demonstration of acceptable margin of safety has not been submitted, however, an assessment of the adequacy of the margin of safety has been provided by comparison of the TMI-1 vessel to analysis results which had been submitted to the NRC by the B&WOG (BAW-2148P). The B&WOG also evaluated these analyses and determined that they bound the expected results of plant specific analysis, including the TMI-1 vessel. As stated in our response to NRC Generic Letter No. 92-01, preparation and submittal of the required analysis, to specifically address each of the B&WOG plants, will be accomplished in 1993 by the B&WOG in accordance with commitments already made to the NRC. Submittal of the plant specific analysis will satisfy the requirements of 10 CFR 50, Appendix G, Sections V.C.2 and V.C.3.

PTS Rule (RT_{PTS}), 10 CFR 50.61:

In order to provide the NRC with complete information regarding the TMI-1 reactor vessel embrittlement status at the end of plant license, 10 CFR 50.61, known as the PTS Rule, and NRC SER for TMI-1 require an update of the RT_{PTS} value whenever a TSCR is submitted for the operating pressure/temperature curves. Review of this issue for TMI-1 indicates that there is basically no change in the RT_{PTS} value which is predicted in accordance with the calculative procedure given in paragraph (b)(2) of the Rule, however, in accordance with paragraph (b)(3) of the Rule, the predicted value of RT_{PTS} for the TMI-1 reactor vessel welds is significantly lower than the screening criteria. The limiting RT_{PTS} for the TMI-1 vessel had been previously calculated at $\leq$ 270°F screening criteria for longitudinal weld SA-1526 made from weld wire No. 299L44. The present estimate (Technical Specification Section 3.1.2 Reference No. 6) at 32 EFPY (beyond the present license) for this controlling longitudinal SA-1526 weld, per paragraph (b)(3) of the Rule, is approximately 217 deg.F and the limiting circumferential WF-25 weld has an RT_{PTS} of 225 deg.F.

GPU Nuclear will maintain continued vigilance and proactiveness in monitoring and assessing the TMI-1 reactor vessel embrittlement and, as necessary, continue to actively participate in the B&W Reactor Vessel Working Group.

IV. NO SIGNIFICANT HAZARDS CONSIDERATIONS

GPU Nuclear has determined that the Technical Specification Change Request poses no significant hazards as defined by NRC in 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The design basis event related to this change is nonductile failure of the reactor coolant pressure boundary. The updated pressure/temperature limits have been established in accordance with the requirements of 10 CFR 50, Appendix G. Extending the curves for applicability to thirty-two (32) EFPY is based on maintaining the design margin assumed in the original curves. Operation of the facility in accordance with the proposed amendment provides assurance

of protection against nonductile failure of the reactor coolant pressure boundary for operation of thirty-two (32) EFPY. Therefore, operation in accordance with the proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The design basis event related to the change is nonductile failure of the reactor coolant pressure boundary. The proposed amendment provides assurance of protection against nonductile failure of the reactor coolant pressure boundary for operation of 32 EFPY and is unrelated to the possibility of creating a new or different kind of accident.

3. Operation of the facility in accordance with the proposed amendment would not involve any reduction in a margin of safety since the design margin assumed in the original curves is still maintained.

V. IMPLEMENTATION

The existing TMI-1 Technical Specification heatup and cooldown limits are applicable for plant operation to ten (10) EFPY. GPU Nuclear request issuance of this amendment by August 1, 1993 to allow sufficient time for implementation prior to the plant exceeding 10 EFPY.