August 16, 1993

Docket No. 50-289

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Mr. T. Gary Broughton, Vice President
and Director - TMI-1
GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

Dear Mr. Broughton:

SUBJECT: ISSUANCE OF AMENDMENT NO.176 FOR TSCR NO. 207 (TAC NO. M86085)

The Commission has issued the enclosed Amendment No.176 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1) in response to your letter dated March 19, 1993, as supplemented on July 23, 1993. The amendment is enclosed.

The amendment extends the effectiveness of the Technical Specification (TS) pressure vs. temperature limits until plant operation has reached 15.2 effective full power years (EFPY). The original TS Change Request proposed extending these limits (TS Figures 3.1-1 and 3.1-2) to 32 EFPY corresponding to the end of the license period. However, in a series of conference calls with your Headquarters staff at Parsippany, the NRC technical staff raised a number of questions regarding the methodology and data to be used in calculating the adjusted reference temperature. The methodology proposed by GPU Nuclear is somewhat different than that prescribed in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." There are ongoing evaluations of reactor vessel integrity in conjunction with Generic Letter 92-01 that will not be completed until 1994. Therefore, the NRC staff is not yet in a position to accept this modified approach. However, the staff does find that the present limits are acceptable to at least 15.2 EFPY.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

OFFICE	LA:PDI-4	PM:PDI-4	PD:PDI-4	OGC NLO	
NAME	SNorris	RHernan	JStolz	M ZOBLGN	
DATE	8/13/93	8/13/93	8/16/93	8/10/93	/ /

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one minor addition

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

August 16, 1993

Docket No. 50-289

Mr. T. Gary Broughton, Vice President
and Director - TMI-1
GPU Nuclear Corporation
Post Office Box 480
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Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

Mr. T. Gary Broughton
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee) dated March 19, 1993, as supplemented on July 23, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 176 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 16, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3-3	3-3
3-4	3-4
3-5	3-5
3-5a	3-5a
3-5b	3-5b

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objective

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

Specification

3.1.2.1 For operations until 15.2 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding 15.2 effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G, Section V.C.

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional 15°F step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 252°F.

The unirradiated reference nil ductility temperature (RT_{NDT}) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated as described in reference No. 6.

The predicted RT_{NDT} was calculated using the respective predicted neutron fluence at 15.2 effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2, Section C.1.1 for the plate metals and the provisions of Section C.2.1 for the weld metals. The analysis of the reactor vessel material contained in the second Three Mile Island Nuclear Station Unit 1 surveillance capsule as well as B&WOG surveillance capsules confirmed that the current technique, as described in reference No. 6, for predicting the change in impact properties due to irradiation are conservative.

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 4. Extrapolation of reactor vessel fluxes (average of cycles 8 & 9), and corresponding fluence accumulations, based on predicted future fuel cycle design conditions during 15.2 effective full power years of operation are described in References 5 and 6.

Based on the predicted RT_{NDT} after 15.2 effective full power years of operation, the pressure/temperature limits of Figure 3.1-1 and 3.1-2 have been established in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figures 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in either loop.

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

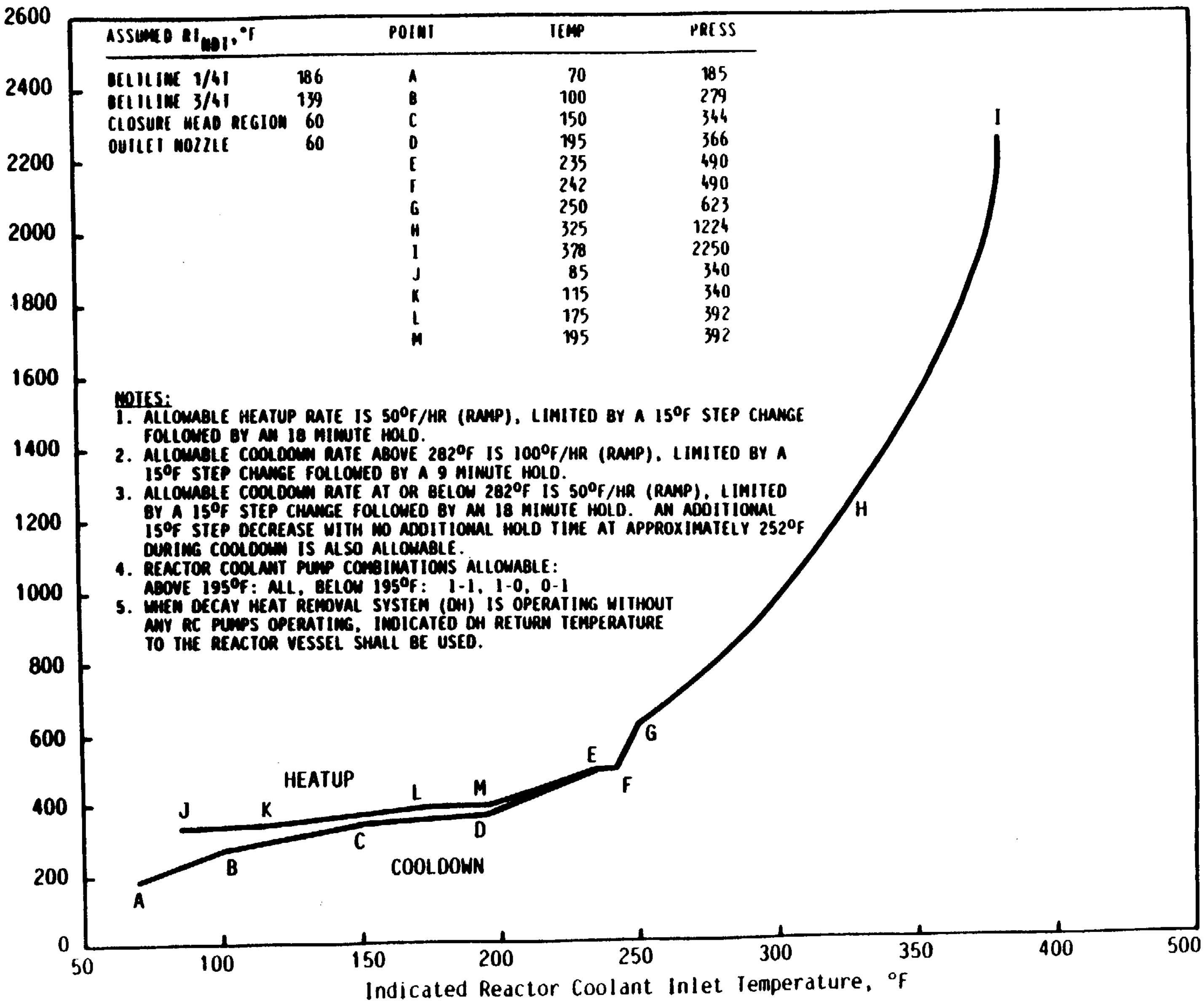
(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits

(5) BAW-2108, Rev. 1, B&WOG Materials Committee Report "Fluence Tracking System"

(6) GPUN Topical Report 095, Rev. 0, "Summary Evaluation of the TMI-1 Reactor Vessel Embrittlement for Operating Pressure/Temperature Limits"

Figure 3.1-1 Reactor Coolant System Heatup/Cooldown Limitations

(Applicable through 15.2 EFPY)



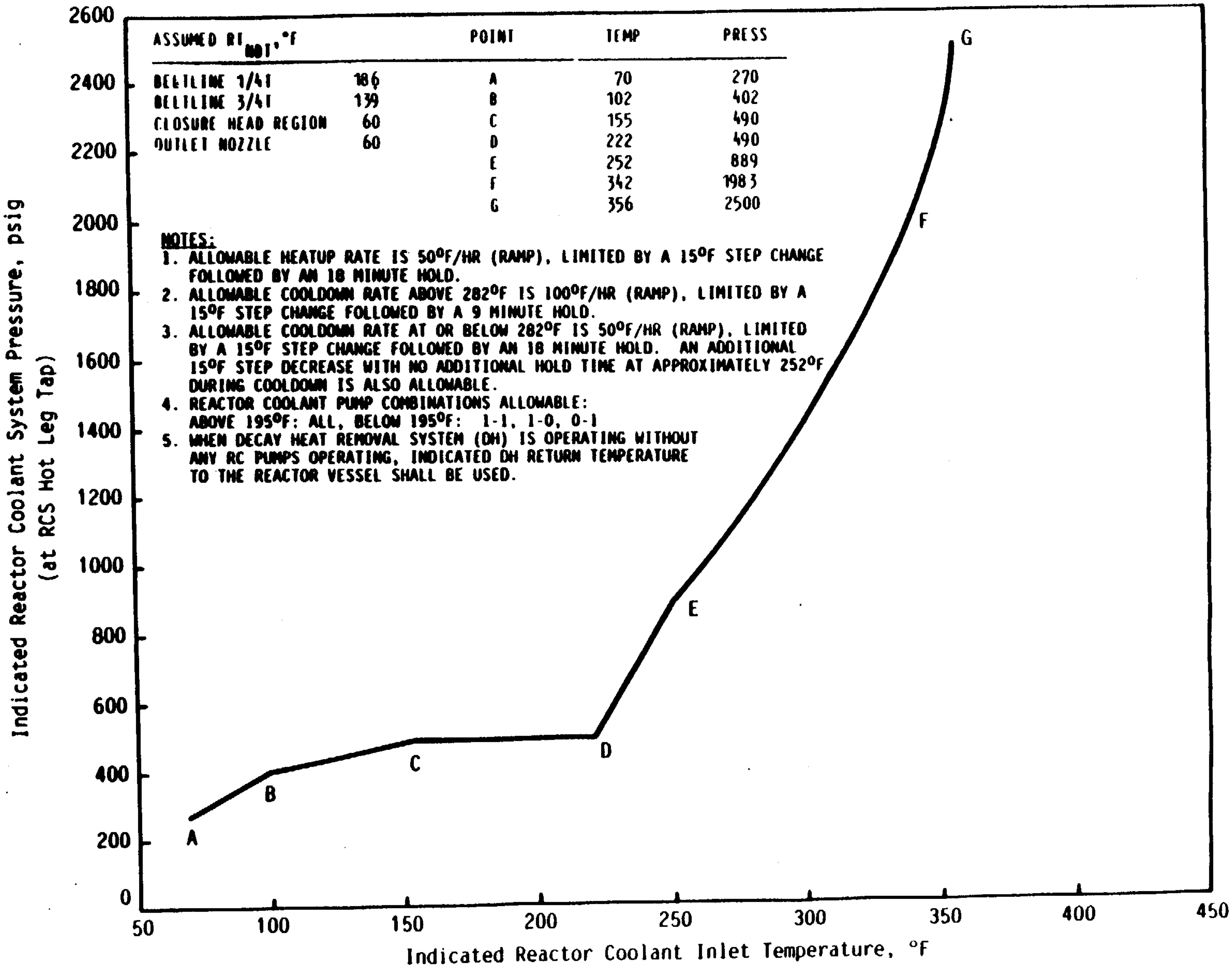
ASSUMED RT_{NDT}, °F		POINT	TEMP	PRESS
BELTLINE 1/4T	186	A	70	185
BELTLINE 3/4T	139	B	100	279
CLOSURE HEAD REGION	60	C	150	344
OUTLET NOZZLE	60	D	195	366
		E	235	490
		F	242	490
		G	250	623
		H	325	1224
		I	378	2250
		J	85	340
		K	115	340
		L	175	392
		M	195	392

NOTES:

1. ALLOWABLE HEATUP RATE IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD.
2. ALLOWABLE COOLDOWN RATE ABOVE 282°F IS 100°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY A 9 MINUTE HOLD.
3. ALLOWABLE COOLDOWN RATE AT OR BELOW 282°F IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD. AN ADDITIONAL 15°F STEP DECREASE WITH NO ADDITIONAL HOLD TIME AT APPROXIMATELY 252°F DURING COOLDOWN IS ALSO ALLOWABLE.
4. REACTOR COOLANT PUMP COMBINATIONS ALLOWABLE: ABOVE 195°F: ALL, BELOW 195°F: 1-1, 1-0, 0-1
5. WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.

Figure 3.1-2 Reactor Coolant Inservice Leak and Hydrostatic Test

(Applicable through 15.2 EFPY)



ASSUMED RT_{NDT}, °F		POINT	TEMP	PRESS
BELTLINE 1/4T	186	A	70	270
BELTLINE 3/4T	139	B	102	402
CLOSURE HEAD REGION	60	C	155	490
OUTLET NOZZLE	60	D	222	490
		E	252	889
		F	342	1983
		G	356	2500

NOTES:

1. ALLOWABLE HEATUP RATE IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD.
2. ALLOWABLE COOLDOWN RATE ABOVE 282°F IS 100°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY A 9 MINUTE HOLD.
3. ALLOWABLE COOLDOWN RATE AT OR BELOW 282°F IS 50°F/HR (RAMP), LIMITED BY A 15°F STEP CHANGE FOLLOWED BY AN 18 MINUTE HOLD. AN ADDITIONAL 15°F STEP DECREASE WITH NO ADDITIONAL HOLD TIME AT APPROXIMATELY 252°F DURING COOLDOWN IS ALSO ALLOWABLE.
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5. WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By letters dated March 19 and July 23, 1993, GPU Nuclear Corporation (the licensee) requested to extend the applicability of pressure-temperature (P-T) limits in section 3.1.2 of the TMI Unit 1 Technical Specifications (TS). The licensee proposed to extend the applicability period of the current P-T limits from 10 effective full power years (EFPY) to 15.2 EFPY with the current limit curves unchanged. The July 23, 1993, submittal provided clarifying information within the scope of the initial Federal Register notice.

The following NRC regulations and guidance are applicable in the evaluation of the P-T limits: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials;" Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations."

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide technical specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the technical specifications. The P-T limits are among the limiting conditions of operation in the technical specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P-T limits. An acceptable method for constructing the P-T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature (RT_{NDT}). Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of initial RT_{NDT}, the increase in RT_{NDT} resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone materials of the reactor beltline.

2.0 EVALUATION

The licensee requested to extend the applicability of the current P-T limits from 10 EFPY to 15.2 EFPY with the curves unchanged because the licensee calculated that the curves are acceptable for 15.2 EFPY.

To evaluate the P-T limits, the NRC staff performed calculations to determine the EFPY that corresponds to the limiting ART of 186°F in the current P-T limits using RG 1.99, Rev. 2.

Based on the highest ART of all beltline materials, the staff has determined that the limiting material is the circumferential weld, WF-25, which is located between the upper shell and lower shell. This weld was fabricated using a submerged arc process with Linde-80 flux which also was used in other reactor vessels. The WF-25 weld and the lower shell longitudinal weld, SA-1526, were fabricated using the same weld wire heat number, 299L44. The WF-25 weld receives more neutron fluence than the SA-1526 weld and, therefore, is the limiting material.

These two welds are similar in material properties; therefore, their initial RT_{NDT} data should be treated as a group to obtain a representative initial RT_{NDT}. The staff calculated the mean value of initial RT_{NDT} and its standard deviation to be -7°F and 20.6°F, respectively (Table 1).

Based on RG 1.99, the staff calculated the chemistry factor for weld WF-25 to be 219.4°F using surveillance data from TMI Unit 1 and other reactor vessels that have the same weld metal (Table 2). The staff used a margin of 50°F which was determined based on a sigma I of 20.6°F and sigma delta of 14°F. Based on the above parameters (chemistry factor, initial RT_{NDT}, and margin), the staff determined that an ART of 186°F corresponds to 15.2 EFPY of TMI-1 operation. Substituting the ART of 186°F into SRP 5.3.2., the staff determined that the P-T limits satisfy Sections IV.A.2 & IV.A.3 of Appendix G to 10 CFR 50.

Paragraph IV.B of Appendix G requires that the predicted Charpy upper shelf energy at end of life be above 50 ft-lb. The conformance of upper shelf energy to paragraph IV.B of Appendix G will be determined pending the staff evaluation of licensee's response to Generic Letter 92-01, "Reactor Vessel Structural Integrity."

The licensee has removed two surveillance capsules from TMI-1 reactor vessel. Surveillance results from capsules 1E and 1C were published in Babcock & Wilcox reports, BAW-1439 and BAW-1901, respectively. The conformance of the surveillance program to Appendix H to 10 CFR 50 also will be determined under the staff evaluation of licensee's response to Generic Letter 92-01.

The proposed P-T limits do not contain limits for core criticality because Technical Specification 3.1.3.1 contains a core criticality limit (525°F) that is more restrictive than the limits required by Section IV.A.3 in Appendix G of 10 CFR 50.

The staff has determined that the current P-T limits for heatup, cooldown, and inservice leak and hydrostatic test are valid for 15.2 EFPY. The staff's evaluation included performing independent calculations using RG 1.99, Rev. 2 and SRP 5.3.2. The staff concludes that the proposed changes to the current P-T limits may be incorporated in the TMI Unit 1 Technical Specifications.

The staff is evaluating the issues concerning the upper shelf energy and reactor vessel material surveillance program under Generic Letter 92-01. If the staff evaluation of the licensee's responses to Generic Letter 92-01 result in more conservative P-T limits than the one proposed, the proposed P-T limits may need to be revised.

The staff has reviewed this request and its justification and concludes they are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 34080). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Tsao

Date: August 16, 1993

TABLE 1

PLANT	WELD	IRT,°F	REFERENCES AND COMMENTS
TMI-1	WF-25	-19	BAW-1803, Rev.1, page A-20
TMI-1	WF-25	-13	BAW-1803, Rev.1, page 3-3, WF-25(5)
SURRY	SA-1526	-10	BAW-1803, Rev.1, page A-72
CRYSTAL RIVER	WF-25	-27	BAW-1803, Rev.1, Page A-35
CRYSTAL RIVER	SA-1526	-20	BAW-1803, Rev.1, page 3-3
OCONEE 3	WF-25	7	BAW-1803, Rev.1, page 3-3
TMI-2	WF-25	33	BAW-1803, Rev.1, page 3-3

Mean value of IRT = -7°F
Standard Deviation = 20.6°F

WF-25 and SA-1526 0.35% copper and 0.68% nickel

IRT = initial reference temperature

BAW-1803, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," Babcock & Wilcox, May 1991

TABLE 2

1	2	3	4	5	6	7	8	9	10	11
TMI-1	E	WF-25	124	1.07	.431	.185	53	95	29	299L44
TMI-1	C	WF-25	203	8.66	.959	.919	195	212	-9	299L44
Surry	T	SA1526	167	2.86	.658	.433	110	146	21	299L44
Surry	V	SA1526	240	19.4	1.18	1.39	283	261	-21	299L44
Crystal River	C1	WF-25	214	7.79	.93	.865	199	206	8	299L44

1. plant name
2. surveillance capsules identification
3. surveillance weld identification
4. measured RTndt shift taken from capsule reports
5. neutron fluence of the capsule taken from capsule reports
6. fluence factor per RG 1.99
7. square of the fluence factors in column 6
8. measured RTndt shift x fluence factor (column 4 x column 6) per RG 1.99
9. predicted RTndt shift (chemistry factor x column 6) per RG 1.99
10. measured RTndt shift minus predicted RTndt shift (col. 4 - col. 9)
11. weld wire heat number

Sum of column 8 = 840

Sum of column 7 = 3.792

Based on Position 2.1 of Regulatory guide 1.99, Rev. 2,

Chemistry Factor = 840/3.792 = 221.5