NOV 18 1987

Docket No. 50-289

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

Subject: Issuance of Amendment (TAC # 65891)

The Commission has issued the enclosed Amendment No. 134 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, (TMI-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated July 24, 1987.

The amendment revises the TSs to incorporate updated reactor coolant system heatup and cooldown limits for operation to 10 effective full power years.

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

Sincerely,

ORIGINAL SIGNED BY

Gordon E. Edison, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 134 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

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* Previously concurred

Mr. Henry D. Hukill
GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

cc:
Mr. G. Broughton
O&M Director, TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Richard J. McGoey
Manager, PWR Licensing
GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, New Jersey 70754

Mr. C. W. Smyth
TMI-1 Licensing Manager
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Ernest L. Blake, Jr., Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, D.C. 20037

Mr. Larry Hochendoner
Dauphin County Commissioner
Dauphin County Courthouse
Front and Market Streets
Harrisburg, Pennsylvania 17120

Mr. David D. Maxwell, Chairman
Board of Supervisors
Londonderry Township
RFD#1 - Geyers Church Road
Middletown, Pennsylvania 17057

Mr. Richard Conte
Senior Resident Inspector (TMI-1)
U.S.N.R.C.
P.O. Box 311
Middletown, Pennsylvania 17057

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Governor's Office of State Planning
and Development
ATTN: Coordinator, Pennsylvania
State Clearinghouse
P. O. Box 1323
Harrisburg, Pennsylvania 17120

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Resources
P. O. Box 2063
Harrisburg, Pennsylvania 17120

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND 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. 134
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 July 24, 1987, 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.

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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:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 134 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: NOV 18 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 134

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
vii	vii
3-3	3-3
3-4	3-4
3-5	3-5
Figure 3.1-1	Figure 3.1-1
Figure 3.1-2	Figure 3.1-2

LIST OF FIGURES

Figure	Title
2.1-1	TMI-1 Core Protection Safety Limit
2.1-2	TMI-1 Core Protection Safety Limits
2.1-3	TMI-1 Core Protection Safety Bases
2.3-1	TMI-1 Protection System Maximum Allowable Set Points
2.3-2	Protection System Maximum Allowable Set Points for Reactor Power Imbalance, TMI-1
3.1-1	Reactor Coolant System Heatup/Cooldown Limitations (Applicable to 10 EFPY)
3.1-2	Reactor Coolant System, Inservice Leak and Hydrostatic Test Limitations (Applicable to 10 EFPY)
3.1-3	Limiting Pressure vs. Temperature Curve for 100 STD cc/Liter H_2O
3.5-2A	Rod Position Limits for 4 Pump Operation from 0 to 30+10/-0 EFPD, TMI-1
3.5-2B	Rod Position Limits for 4 Pump Operation from 30+10/-0 to 250±10 EFPD, TMI-1
3.5-2C	Rod Position Limits for 4 Pump Operation after 250±10 EFPD, TMI-1
3.5-2D	Rod Position Limits for 3 Pump Operation from 0 to 30+10/-0 EFPD, TMI-1
3.5-2E	Rod Position Limits for 3 Pump Operation from 30+10/-0 to 250±10 EFPD, TMI-1
3.5-2F	Rod Position Limits for 3 Pump Operation after 250±10 EFPD, TMI-1
3.5-2G	Rod Position Limits for 2 Pump Operation from 0 to 30+10/-0 EFPD, TMI-1
3.5-2H	Rod Position Limits for 2 Pump Operation from 30+10/-0 to 250±10 EFPD, TMI-1
3.5-2I	Rod Position Limits for 2 Pump Operation after 250±10 EFPD, TMI-1
3.5-2J	Power Imbalance Envelope for Operation from 0 to 30+10/-0 EFPD, TMI-1

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 ten 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 ten 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.[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 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[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 NDTT.[3]

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 by adding the predicted radiation-induced RT NDT and the unirradiated RT NDT for each of the reactor coolant beltline materials.

The predicted RT NDT was calculated using the respective neutron fluence after ten effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2. The analysis of the reactor vessel material contained in the second Three Mile Island Nuclear Station Unit 1 surveillance capsule confirmed that the current techniques used for predicting the change in impact properties due to irradiation are conservative.[4]

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 (Reference 4). Extrapolation of reactor vessel fluxes, and corresponding fluence accumulations, based on predicted fuel cycle design conditions during 10 effective full power years of operation are described in Reference 4.

Based on the predicted RT NDT after ten 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.[5] 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 assumed 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) FSAR, Section 4.1.2.4

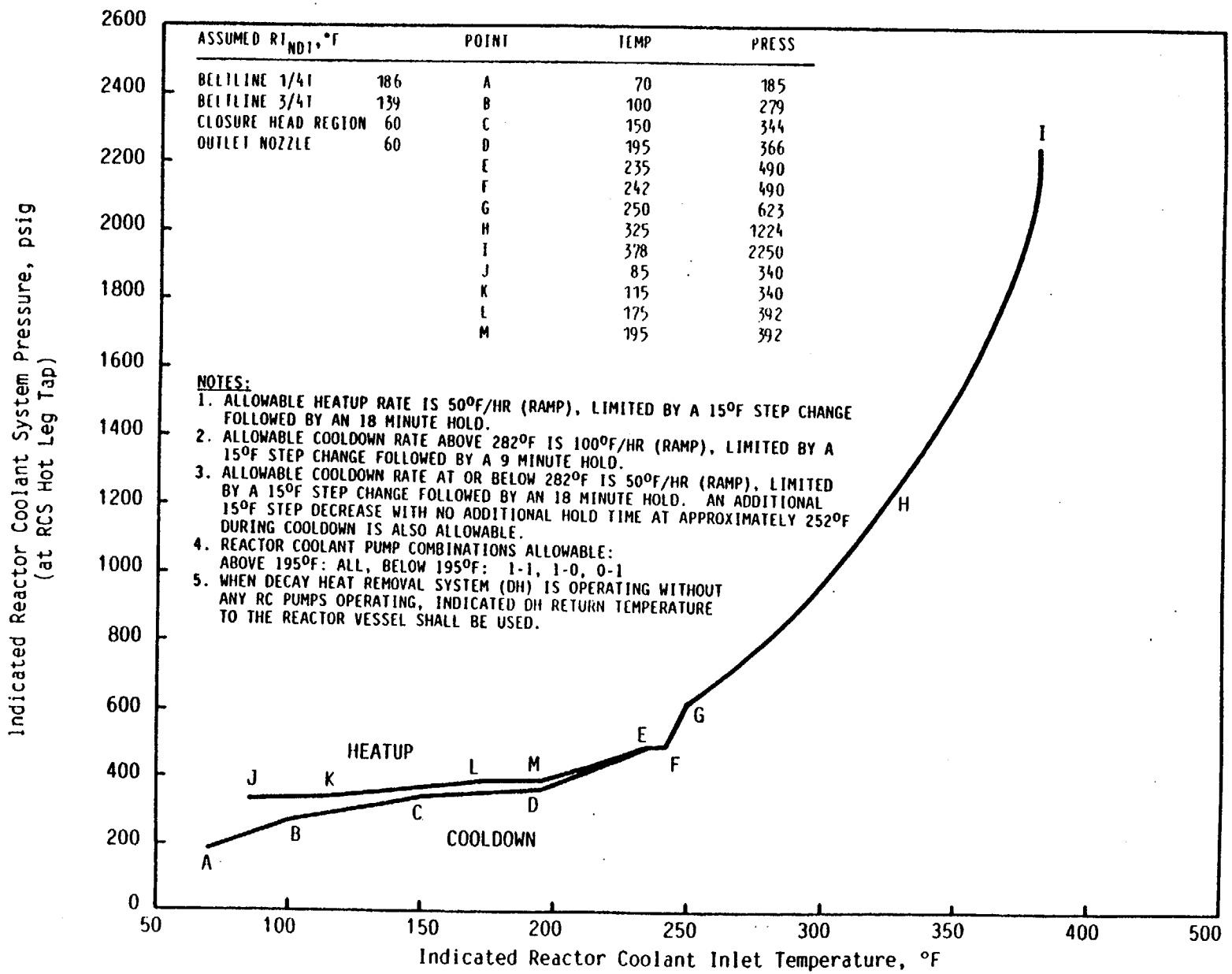
(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

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

(5) 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.

Figure 3.1-1 Reactor Coolant System Heatup/Cooldown Limitations
(Applicable thru 10EFPY)

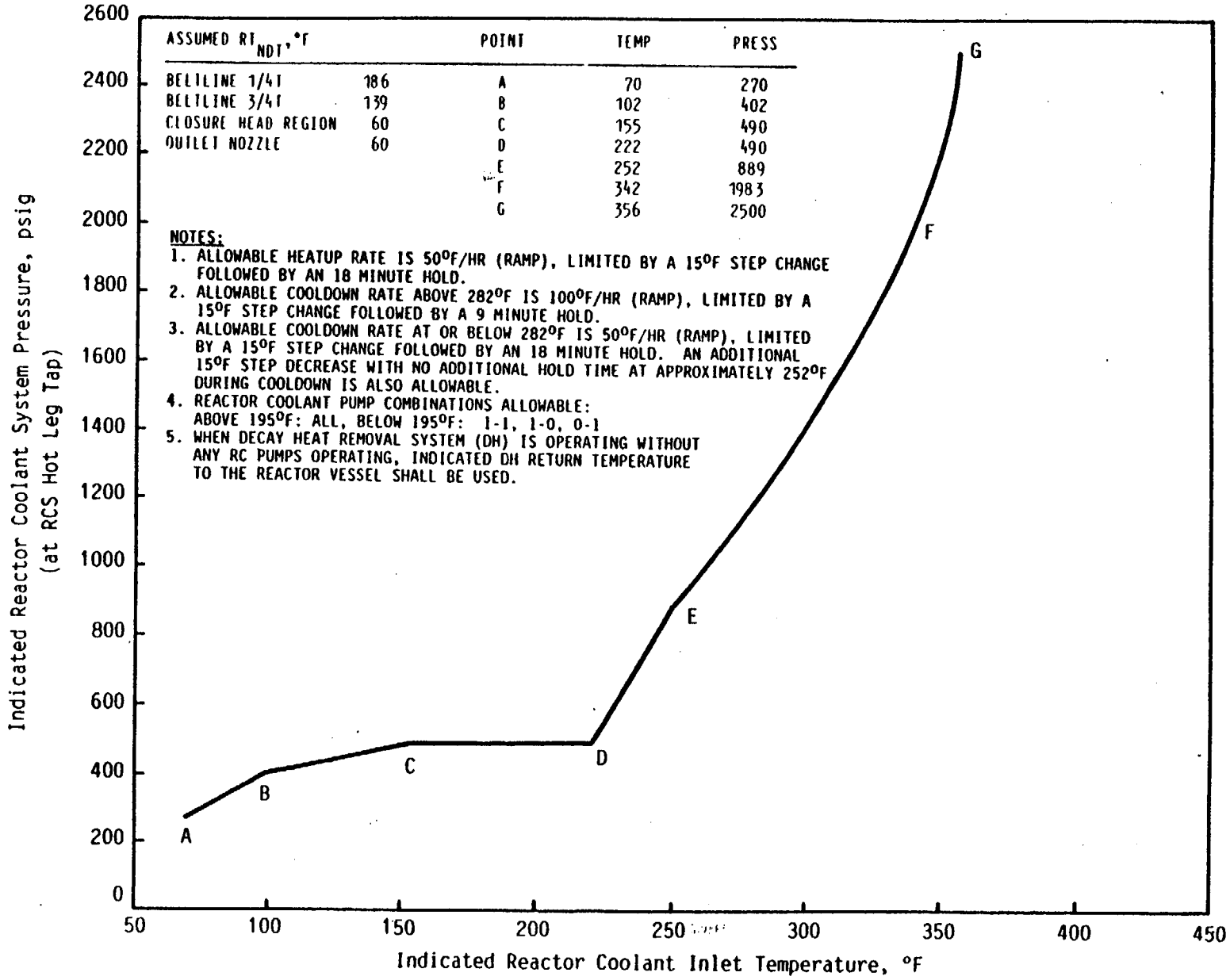


ASSUMED RT_{NDT}, °F		POINT	TEMP	PRESS
BELTLINE 1/4T	186	A	70	185
BELTLINE 3/4T	139	B	100	279
CLOSURE HEAD REGION	60	C	150	344
OUTLET NOZZLE	60	D	195	366
		E	235	490
		F	242	490
		G	250	623
		H	325	1224
		I	378	2250
		J	85	340
		K	115	340
		L	175	392
		M	195	392

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 thru 10EFPY)



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.
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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-50

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

THREE MILE ISLAND NUCLEAR STATION UNIT 1

DOCKET NO.: 50-289

INTRODUCTION

By letter dated July 24, 1987, General Public Utilities Nuclear Corporation (GPUN/licensee) proposed to revise the pressure-temperature limits in the Three Mile Island Nuclear Station Unit 1 (TMI-1) Technical Specifications. The proposed limits will incorporate the updated pressure vs. temperature curves through 10 effective full power years (EFPY) to account for irradiation effects on the reactor vessel material. The proposed limits were based on the Babcock and Wilcox (B&W) analysis of irradiation data taken from several TMI-1 surveillance specimens (Ref. 1). The limits provide maximum permissible pressure at temperature for three reactor operations--system hydrostatic and leakage tests, heatup or cooldown, and core critical condition. The pressure limits at temperature have to satisfy the requirements in Appendix G of 10 CFR 50.

DISCUSSION

Part of the NRC's effort to ensure integrity of the reactor vessel is to periodically evaluate the reduction in fracture toughness of the vessel material due to neutron irradiation damage. The effort consists of three steps. First, the licensee is required to establish a surveillance program in accordance with Appendix H of 10 CFR 50, which requires periodic withdrawal of surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and they should contain test specimens that were made from the plate, weld, and heat affected zone materials of the reactor beltline. Secondly, the licensee is required to perform Charpy impact tests, tensile tests, and neutron fluence measurements of the specimens. These tests provide data for the actual neutron irradiation damage to the reactor vessel in terms of the reference temperature, RT_{NDT}, and the upper shelf energy (USE). The neutron damage is indicated by the decrease in USE and temperature shift in RT_{NDT}. The USE is the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. The USE decreases as a function of neutron fluence and copper content in the irradiated material. The shift of the adjusted reference temperature is the temperature shift in the Charpy curve for the irradiated material relative to that for the unirradiated material.

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According to Appendix G of 10 CFR 50, the USE must not be less than 50 ft-lb and the adjusted RT_{NDT} not more than 200°F. Thirdly, the licensee is required to construct three minimum pressure-temperature curves for the three operating conditions mentioned above and incorporate the curves in the Technical Specifications. To construct the curves, the licensee may use the guidelines delineated in Reg. Guide 1.99, Revision 2, and Appendix G, Section III of the ASME Code.

EVALUATION

The proposed TMI-1 pressure-temperature limits were derived from the irradiation data taken from surveillance specimens in withdrawal capsule TMI-1C. This capsule is a part of the Integrated Reactor Vessel Materials Surveillance Program sponsored by the B&W owners group. The capsule was irradiated in cycle 1 of the TMI-1 reactor and in cycles 3 through 5 of the Crystal River Unit 3 reactor. The staff had reviewed and found the TMI-1 surveillance program, including the capsule withdrawal schedule, acceptable. (Ref. 2 & 3).

The TMI-1C capsule was removed during the fifth refueling outage of Crystal River Unit 3 for examination. The capsule experienced a cumulative neutron fluence of 8.66×10^{18} n/cm^2 (E 1MeV). It contained base metal, correlation metal, heat-affected-zone metal, weld metal, and dosimeter wires. The controlling material was included in the capsule, which was the middle circumferential weld, heat number WF-25. This particular weld metal was fabricated by B&W using a submerged-arc weld process with Linde 80 flux.

The WF-25 weld received a neutron fluence of 3.4×10^{18} n/cm^2 at the inside surface of the vessel. The copper and nickel content of the weld were 0.35% and 0.68% respectively. The adjusted RT_{NDT} through the 10 EFPY at T/4 vessel wall location was calculated to be 191°F using Regulatory Guide 1.99, Rev. 2. The USE was estimated to be above the required 50 ft-lb through 10 EFPY. Both adjusted RT_{NDT} and USE satisfy the 10 CFR 50 Appendix G requirements.

Presently, TMI-1 is in fuel cycle 6 which is equivalent to 4.83 EFPY. The staff believes that the proposed pressure-temperature limits are valid and that fracture toughness of the TMI-1 reactor vessel will be within the safety limits through 10 EFPY.

The staff has reviewed the proposed TMI-1 pressure-temperature limits and changes in the Technical Specifications. The staff has determined that the licensee has appropriately applied the methodology in 10 CFR 50, Appendix G and Regulatory Guide 1.99, Rev. 2. The staff concludes that the pressure-temperature limits are valid through 10 EFPY and may be incorporated into the TMI-1 Technical Specifications.

ENVIRONMENTAL CONSIDERATION

This 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. We have 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Lowe, Jr. A.L. et al, "Analysis of Capsule TMI 1-C GPU Nuclear Three Mile Island Nuclear Station - Unit 1." Babcock and Wilcox, BAW-1901, and Supplement 1, March 1986.

2. Memorandum from D. Crutchfield to J. Thomas, Subject: GPU Nuclear Corporation, TMI-1 (Tac No. #59813), March 24, 1986

3. Memorandum from W. Johnson to G. Lainas, Subject: GPU Corporation TMI Unit 1, May 16, 1983

Dated: November 18, 1987

Principal Contributor:
J. Tsao