

February 11, 1993

Docket No. 50-289

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

Dear Mr. Broughton:

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SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 82020 AND M84842) - TSCR 208/220

The Commission has issued the enclosed Amendment No.168 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, in response to your letters dated August 9, 1991 and October 29, 1992.

The amendment revises the Technical Specifications Bases addressing the minimum borated water storage volumes to ensure adequate shutdown margin exist with respect to the loss of coolant accident (LOCA) linear heat rate allowable limits. Technical Specifications Figure 3.5-2M, "LOCA Limited Maximum Allowable Linear Heat Rate," is revised to reflect the Babcock & Wilcox re-evaluations of generic linear heat rate limits. In accordance with the intent of NRC Generic Letter 88-16, this figure is removed from the TMI-1 Technical Specifications and incorporated into the TMI-1 Core Operating Limits Report.

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:

1. Amendment No. 168 to DPR-50
2. Safety Evaluation

cc w/enclosures:

See next page

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DATE	1/12/92	1/11/93	1/11/92	01/12/93	2/1/93

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GPU Nuclear Corporation

Three Mile Island Nuclear Station,
Unit No. 1

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

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. 168
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The applications for amendment by GPU Nuclear Corporation, et al. (the licensee), dated August 9, 1991, and October 29, 1992, comply 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. 168, 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, Director
Project Directorate 1-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 11, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 168

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

viii
3-20
3-35a
3-36b (Figure 3.5-2M)
6-19a
6-20

Insert

viii
3-20
3-35a
3-36b ("DELETED" Figure 3.5-2M)
6-19a
6-20

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>
3.5-2K thru 3.5-2L	DELETED
3.5-2M	DELETED
3.5-1	Incore Instrumentation Specification Axial Imbalance Indication, TMI-1
3.5-2	Incore Instrumentation Specification Radial Flux Tilt Indication, TMI-1
3.5-3	Incore Instrumentation Specification
3.11-1	Transfer Path to and from Cask Loading Pit
4.17-1	Snubber Functional Test - Sample Plan 2
5-1	Extended Plot Plan TMI
5-2	Site Topography 5 Mile Radius
5-3	Locations of Gaseous Effluent Release Points and Liquid Effluent Outfalls

The quantity of boric acid in storage from either of the three above mentioned sources is sufficient to borate the reactor coolant system to a one percent subcritical margin in the cold condition at the worst time in core life with a stuck control rod assembly. Minimum volumes of 906 ft³ of 8700 ppm boron as concentrated boric acid solution in the boric acid mix tank or in a reclaimed boric acid storage tank or 60,000 gallons of 2270 ppm boron as boric acid solution in the borated water storage tank will each satisfy this requirement. Technical Specification 3.3 assures that at least two of these supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The primary method of adding boron to the reactor coolant system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using either the 10 gpm boric acid pumps or the 30 gpm reclaimed boric acid pumps. Using only one of the two 10 gpm boric acid pumps, the required volume can be injected in less than 13 hours. The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup and purification pumps. The 60,000 gallons of boric acid can be injected in less than four hours using only one of the makeup and purification pumps.

Concentration of boron in the boric acid mix tank or a reclaimed boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, the boric acid mix tank is provided with an immersion electric heating element and the reclaimed boric acid tanks are provided with low pressure steam heating jackets to maintain the temperature of their contents well above (10°F or more) the crystallization temperature of the boric acid solution contained in them. Both types of heaters are controlled by temperature sensors immersed in the solution contained in the tanks. Further, all piping, pumps and valves associated with the boric acid mix tank and the reclaimed boric acid storage tanks to transport boric acid solution from them to the makeup and purification system are provided with redundant electrical heat tracing to ensure that the boric acid solution will be maintained 10°F or more above its crystallization temperature. The electrical heat tracing is controlled by the temperature of the external surfaces of the piping systems. Once in the makeup and purification system, the boric acid solution is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

References

- (1) UFSAR, Section 9.1 - "Makeup and Purification System"
- (2) UFSAR, Section 9.2 - "Chemical Addition and Sampling Systems"
- (3) UFSAR, Figures 6.0-1, 6.0-2 - Simplified ECCS Diagrams

- e. If an acceptable axial power imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.
 - f. Axial power imbalance shall be monitored on a minimum frequency of once every two hours during power operation above 40 percent of rated power.
- 3.5.2.8 A power map shall be taken at intervals not to exceed 30 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in the CORE OPERATING LIMITS REPORT.

Bases

The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. The effects of the APSRs are included in the limit development. Additional conservatism included in the limit development is introduced by application of:

- a. Nuclear uncertainty factors
- b. Thermal calibration uncertainty
- c. Fuel densification effects
- d. Hot rod manufacturing tolerance factors
- e. Postulated fuel rod bow effects
- f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

The incore instrumentation system uncertainties used to develop the axial power imbalance and quadrant tilt limits accounted for various combinations of invalid Self Powered Neutron Detector (SPND) signals. If the number of valid SPND signals falls below that used in the uncertainty analysis, then another system shall be used for monitoring axial power imbalance and/or quadrant tilt.

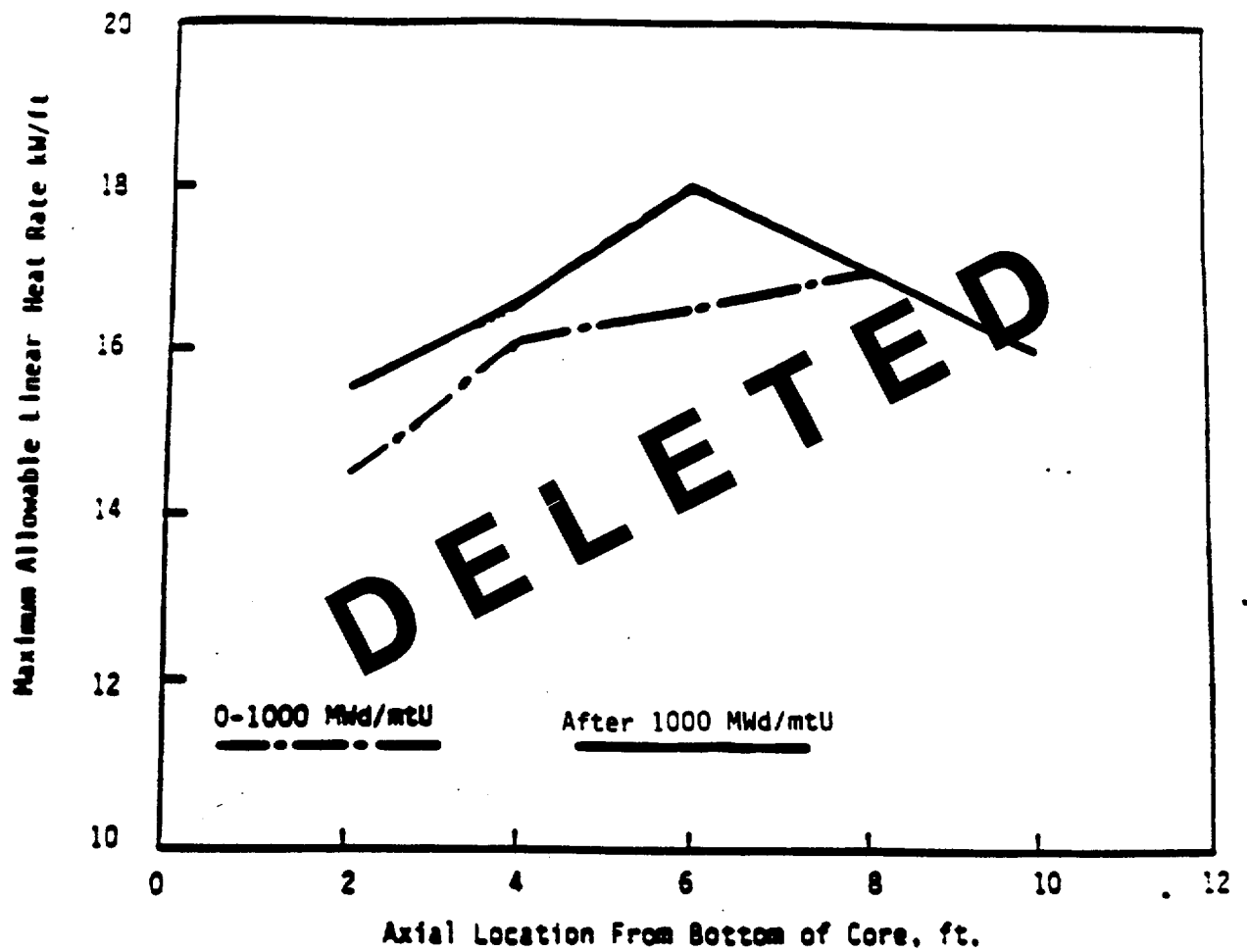


FIGURE DELETED AND INCORPORATED
 INTO THE CORE OPERATING LIMITS REPORT.

LOCA LIMITED MAXIMUM
 ALLOWABLE LINEAR HEAT RATE

TMI-1

Amendment No. 1/2 152,168

Figure 3.5-2M

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-10122A Rev. 1, "Normal Operating Controls," May 1984.
- (2) BAW-10116-A, "Assembly Calculations and Fitted Nuclear Data," May 1977.
- (3) BAW-10117P-A, "Babcock & Wilcox Version of PDQ User's Manual," January 1977.
- (4) BAW-10118A, "Core Calculational Techniques and Procedures," December 1979.
- (5) BAW-10124A, "FLAME 3 - A Three-Dimensional Nodal Code for Calculating Core Reactivity and Power Distributions," August 1976.
- (6) BAW-10125A, "Verification of Three-Dimensional FLAME Code," August 1976.
- (7) BAW-10152A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985.
- (8) BAW-10119, "Power Peaking Nuclear Reliability Factors," June 1977.
- (9) BAW-10103, Rev. 3, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," July 1977.
- (10) BAW-1915P-A, "Bounding Analytical Assessment of NUREG-0630 Models on LOCA kw/ft Limits With Use of FLECSET," May 1986.
- (11) BAW-10104P-A, Rev. 5, B&W ECCS Evaluation Model," November 1988.
- (12) BAW-10162P-A, "TACO-3 Fuel Pin Thermal Analysis Computer Code," November 1989.

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.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

- a. Records of normal station operation including power levels and periods of operation at each power level.
- b. Records of principal maintenance activities, including inspection, repairs, substitution, or replacement of principal items of equipment related to nuclear safety.
- c. All REPORTABLE EVENTS.
- d. Records of periodic checks, tests and calibrations.
- e. Records of reactor physics tests and other special tests related to nuclear safety.

- f. Changes to procedures required by Specification 6.8.1.
- g. Records of solid radioactive shipments.
- h. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.
- i. Results of annual physical inventory verifying accountability of licensed sources on record.
- j. Control Room Log Book.
- k. Shift Foreman Log Book.

6.10.2 The following records shall be retained for the duration of Operating License DPR-50 unless otherwise specified in 6.10.1 above.

- a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.
- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Routine unit radiation surveys and monitoring records.
- d. Records of radiation exposure history and radiation exposure status of personnel, including all contractors and unit visitors who enter radioactive material areas.
- e. Records of radioactive liquid and gaseous wastes released to the environment, and records of environmental monitoring surveys.
- f. Records of transient or operational cycles for those facility components which affect nuclear safety for a limited number of transients or cycles as defined in the Final Safety Analysis Report.
- g. Records of training and qualification for current members of the unit staff.
- h. Records of in-service inspections performed pursuant to these Technical Specifications.
- i. Records of Quality Assurance activities required by the Operational Quality Assurance Plan.
- j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- k. Records of reviews by the Independent Onsite Safety Review Group.
- l. Records of analyses required by the radiological environmental monitoring program.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 168 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 letter dated August 9, 1991 and October 29, 1992, the GPU Nuclear Corporation (the licensee) submitted two requests for changes to the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1) Technical Specifications (TS). The requested changes would revise the Technical Specifications Bases addressing the minimum borated water storage volumes with respect to the loss of coolant accident (LOCA) linear heat rate (LHR) allowable limits. Figure 3.5-2M, "LOCA Limited Maximum Allowable Linear Heat Rate," is revised to reflect the Babcock & Wilcox (B&W) re-evaluations of generic linear heat rate limits. The second requested change would remove Figure 3.5-2M from the TS and incorporate it into the TMI-1 Core Operating Limits Report (COLR).

2.0 EVALUATION

By letter dated August 9, 1991, the licensee requested a change that would revise TMI-1 TS to reflect the B&W re-evaluation of the generic LOCA linear heat rate (LHR) allowable limits and to administratively update the Bases discussion associated with the minimum borated water storage volumes to ensure adequate shutdown margin. To assure preservation of 10 CFR 50.46 acceptance criteria over entire core life, the licensee developed new LOCA LHR limits using NRC approved B&W emergency core cooling system (ECCS) evaluation model and fuel performance computer codes (TS section 6.9.5.2 new references 9 through 12). From the analysis, LOCA LHR are required to be reduced at the 6 foot core evaluation from 16.5 to 16.1 Kw/ft at the beginning of core life (0 to 1000 MWD/MTU*) and from 18.0 to 16.1 Kw/ft for above 1000 MWD/MTU. To assure that internal fuel rod pressure does not exceed reactor coolant system (RCS) pressure, the LHR limits are reduced at all core levels as a function of maximum rod average fuel burnup from 36,375 to 60,000 MWD/MTU. Re-evaluation of the peak clad temperatures at various fuel elevations using NRC approved analytical methods determined that peak centerline temperature (PCT) values meet the acceptance criteria using the new LHR limits. These re-evaluations for the revised generic allowable LOCA LHR for B&W 177A lowered-loop plants confirm the preservation of the acceptance criteria. These changes are reflected in the revised Figure 3.5-2M.

* Megawatt-days per metric ton of uranium

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Application of B&W Topical Report BAW-1016P-A "TACO-3 Fuel Pin Thermal Analysis Computer Code" causes a new restriction on the end-of-life LOCA LHR limits. Accordingly, in order to maintain fuel rod internal pressure below RCS pressure for higher burnups, the allowable end-of-life LOCA LHR limits for all fuel assembly elevations are reduced based on the burnup of the limiting fuel rods. Figure 3.5-2M incorporates a diagonally decreasing limit as a function of maximum rod average burnup from 36,375 to 60,000 MWD/MTU.

TS Section 3.2 Bases is revised to change the boric acid solution volume in the Borated Water Storage Tank (BWST) for emergency shutdown requirements from 40,000 gallons to a bounding value of 60,000 gallons. The required volume varies slightly as the core reactivity parameters vary from cycle to cycle. The value is provided in the TS Bases for information only and is not related to any existing TS requirements. Existing TS Section 3:3.1.1 bounds this value by specifying that the BWST shall contain a minimum volume of 350,000 gallons of water having a minimum concentration of 2270 ppm boron. Therefore, the proposed change is administrative.

By letter dated October 29, 1992, the licensee requested a change that would remove TMI-1 Figure 3.5-2M from the TS and incorporate it into the COLR. The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16, which allows the removal of cycle-specific parameter limits from TS. All changes in cycle-specific parameter limits are to be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted to the NRC prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee has provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR 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 amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. 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 (57 FR 34583 and 57 FR 58246). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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: Francis Young

Date: February 11, 1993