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Docket No. 50-289

Mr. Henry D. Hukill
Vice President
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

October 28, 1983

DMB016

Dear Mr. Hukill:

SUBJECT: AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued Amendment No. 90 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 application dated June 20, 1983.

This amendment revises the Technical Specifications to offset a potential non-conservatism in the prediction of peak cladding temperature during a loss of coolant accident. It revises the centerline fuel melt limit in the Technical Specifications for Cycle 5 operation from 19.6 kw/ft to 20.15 kw/ft. The amendment also reduces the reactor protection system flux to pump trip setpoint for two pump operation from 91 percent (%) to 55 percent (%) of rated power, and revises the quadrant tilt instrumentation requirements with respect to the preferred order of use of the three detector systems.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Monthly Federal Register Notice.

Sincerely,

Original signed by

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

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PDR ADOCK 05000289
P PDR

Enclosures:
1. Amendment No. 90 to DPR-50
2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 28, 1983

Docket No. 50-289

Mr. Henry D. Hukill
Vice President
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

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SUBJECT: AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-50

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Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 90 to DPR-50
2. Safety Evaluation

cc w/enclosures:
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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. 90
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 licensees), dated June 20, 1983, 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. 90 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 28, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

Remove Pages	Insert Pages
2-2	2-2
2-9	2-9
3-34	3-34
Fig. 3.5-2G	Fig. 3.5-2G

a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112 percent) when the reactor coolant flow is $139.8 \times 10^{+6}$ lbs/h, which is less than the actual flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors (2) with potential fuel densification and fuel rod bowing effects;

$$F_q^N = 2.57, \; F_{\Delta H}^N = 1.71; \; F_z^N = 1.50$$

The 1.5 axial peaking factor associated with the cosine flux shape provides a lesser margin to a DNBR of 1.3 than the 1.7 axial peaking factor associated with a lower core flux distribution. For this reason the cosine flux shape and the associated $F_z^N = 1.50$ is more limiting and thus the more conservative assumption.

The 1.50 cosine axial flux shape in conjunction with $F\Delta H = 1.71$ define the reference design peaking condition in the core for operation at the maximum overpower. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing;

a. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.57$ of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than 1.3 DNBR.

b. The combination of radial and axial peak that prevents central fuel melting at the hot spot. The limit is 20.15 kW/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

		Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1.	Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(3)
2.	Nuclear power based on flow (2) and imbalance max. of rated power	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear power based (5) on pump monitors, Max. % of rated power	NA	NA	55%	Bypassed
4.	High reactor coolant system pressure, psig max.	2300	2300	2300	1720(4)
5.	Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6.	Variable low reactor coolant system pressure psig, min.	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	Bypassed
7.	Reactor coolant temp. F., Max.	619	619	619	619
8.	High Reactor Building pressure, psig. max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only during reactor shutdown.
(4) Automatically set when other segments of the RPS (as specified) are bypassed.
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.
(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable connectors.

f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2., operation may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

g. If the inoperable rod in Paragraph "e" above is in groups 5, 6, 7, or 8, the other rods in the group may be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality is limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant tilt:

a. Except for physics tests the quadrant tilt shall not exceed +3.52% as determined using the full incore detector system.

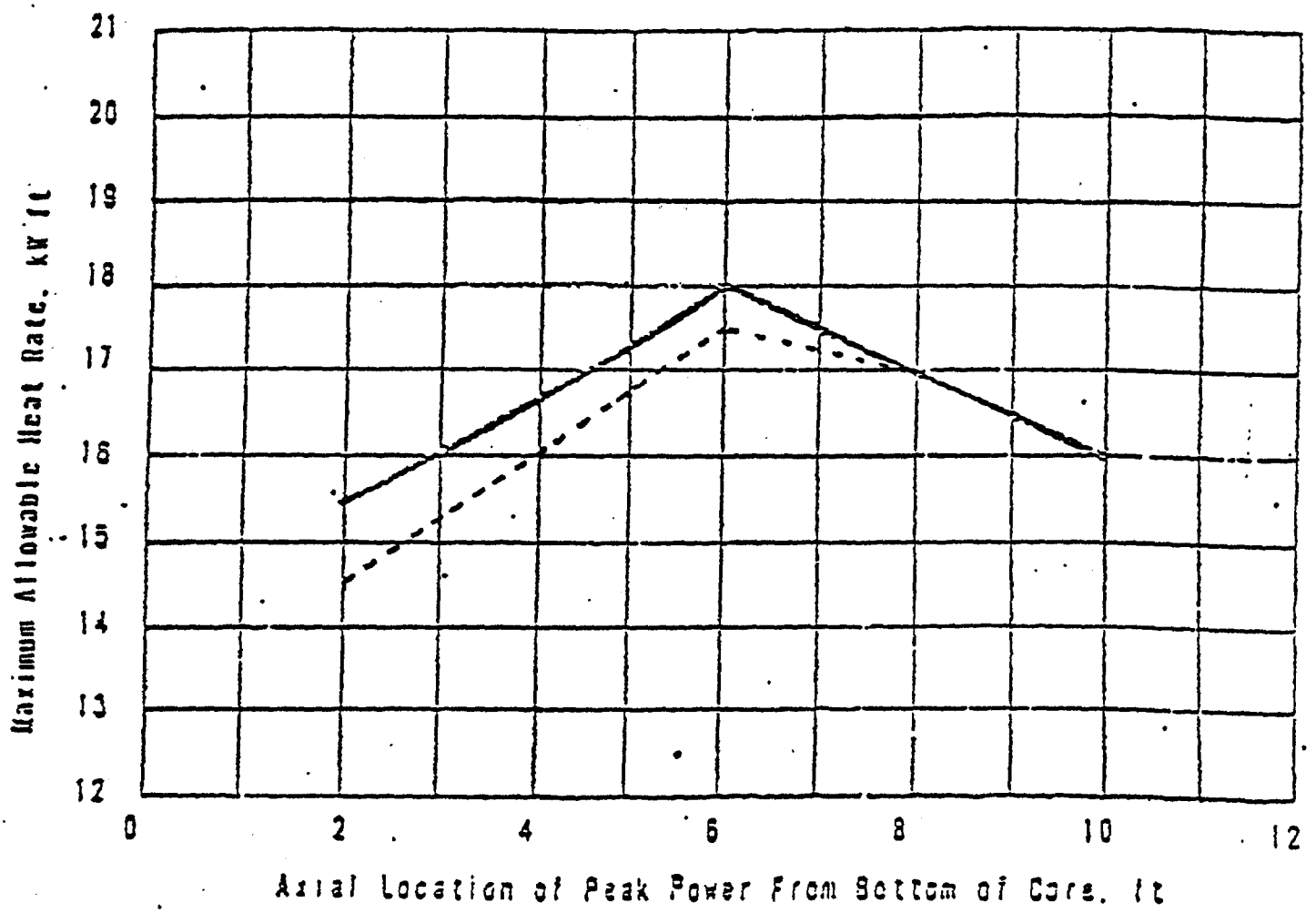
b. When the full incore detector system is not available and except for physics tests quadrant tilt shall not exceed +1.96% as determined using the power range channels displayed on the console each quadrant (out of core detection system).

c. When neither detector system above is available and, except for physics tests, quadrant tilt shall not exceed +1.90% as determined using the minimum incore detector system.

d. Except for physics tests if quadrant tilt exceeds the tilt limit power shall be reduced immediately to below the power level cutoff (see Figures 3.5-2A, and 3.5-2B). Moreover, the power level cutoff value shall be reduced 2 percent for each 1 percent tilt in excess of the tilt limit. For less than four pump operation, thermal power shall be reduced 2 percent of the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of the tilt limit.

e. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following adjustments in setpoints and limits shall be made:

1. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt.



LOCA LIMITED MAXIMUM ALLOWABLE

LINEAR HEAT RATE - TMI-1

Balance of Cycle

First 50 EFPD of Cycle 5

Figure 3.5-2G

Amendment No. 17, 29, 39, 30, 90

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 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 NO. 1

DOCKET NO. 50-289

1.0 Introduction

In response to a request (Ref. 1) from the TMI-1 Restart Issues Task Force, we have reexamined the Three Mile Island Unit 1 Cycle 5 Reload Application (Refs. 2-3) and the original (March 16, 1979) NRC staff evaluation (Ref.4) of that application to determine their continued validity. The reevaluation was conducted in a manner similar to all other operating plants. That is, current, rather than 1979, NRC requirements were applied in the review.

As a result of this reexamination, we identifed two issues that were not addressed in the original reload application, but are now being addressed by all other operating B&W reactors. The first issue concerns the so-called TAFY/TACO penalty proposed by B&W (Ref. 5) and accepted by the NRC staff (Ref. 6) to account for a previously undetected nonconservatism in the Loss of Coolant Accident (LOCA) initial conditions. The second issue concerns nonconservative cladding swelling and rupture models as discussed in our July 13, 1982 letter (Ref. 7) to the licensee, GPU Nuclear Corporation.

In response to these two issues, the GPU Nuclear Corporation submitted on June 20, 1983, Technical Specification Change Request No. 127 requesting amendment to Facility Operating License No. DPR-50. This change request also included several other minor changes to the Technical Specifications and are discussed in the following sections.

2.0 TAFY/TACO Penalty

The Three Mile Island Unit 1 (TMI-1) Cycle 5 Reload Application makes use of two B&W fuel thermal performance codes, TAFY-3 (Ref. 8) and TACO-1 (Ref. 9). Although both of these codes have been approved for use in safety analysis, we believe (Ref. 10) that only the newer TACO-1 code is capable of correctly calculating fission gas release (and therefore rod pressure) at high burnups. Babcock & Wilcox has responded (Ref. 11) to this concern with an analytical comparison between the TAFY-3 and TACO-1 codes. In this response, they have stated that the fuel rod internal pressure predicted by TACO-1 is lower than that predicted by TAFY-3 for fuel rod exposures of up to 42,000 MWd/MtU. Although we have not examined this comparison, we note that the analyses exceed the maximum expected exposure (32,387 MWd/MtU peak assembly) for all fuel in the TMI-1 Cycle 5 core.

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For the LOCA analysis (Section 7.2 of the Reload Report), the average fuel temperature as a function of linear heat rate and the lifetime pin pressure data were calculated with the older TAFY-3 code. The licensee has stated that corresponding parameters used in the generic LOCA analysis are conservative compared with those calculated for Cycle 5 at Three Mile Island Unit 1. However, information obtained by the NRC staff (Ref. 12) indicates that these TAFY-3 predictions do not produce higher calculated peak cladding temperatures in the generic LOCA analysis than the newer TACO-1 code. The issue involves excessive fuel densification and lowered fuel rod internal gas pressures at beginning of life. Babcock & Wilcox has proposed a method of resolving this issue which has been adopted by GPU Nuclear (Ref. 13). The method relies on reduced peak linear heat rate (PLHR) limits at low core elevations for the first 50 effective full power days (EFPD) of operation and is based upon a comparison of TAFY-3 and TACO-1 calculated LOCA initial conditions.

Two sets of bounding values for allowable LOCA PLHRs are given as a function of core height. The first set applies to the first 50 EFPD and the second set to the balance of Cycle 5. We have reviewed the comparison of Cycle 5 limits to limiting core protection safety limits given in Table 1 of the original reload report (Ref. 2). We find that sufficient margin is available in the Cycle 5 Technical Specifications on rod index, axial power shaping rod position, and axial power imbalance so as to bound the interim LOCA PLHR limits from 0 to 50 EFPD. We, therefore, conclude that the new limits are satisfactorily incorporated into the Technical Specifications for Cycle 5.

3.0 Cladding Swelling and Rupture Models

In addition to the issue of initial fuel temperatures and rod internal pressures used in the LOCA analysis, a second issue involving cladding swelling and rupture models has affected the proposed Cycle 5 operating limits for TMI-1. In late 1979, the NRC staff reviewed Emergency Core Cooling System (ECCS) fuel cladding models in light of new data. Adequacy of the models then in use was questioned and new models, developed as Appendix K acceptance criteria, were presented in NUREG-0630 (Ref. 14). Each fuel vendor was then asked to show how, in light of the new models, the plants analyzed with their analytical methods continued to meet the applicable LOCA limits. The B&W response (Ref. 15) concluded that the impact of the NRC models was small and did not result in analytical results in excess of the LOCA limits.

A more recent B&W calculation (Ref. 16), however, found that the cladding swelling and rupture models presented by the staff have a non-trivial effect on LOCA peak cladding temperatures in B&W 177 fuel assembly plants. Because this calculation was applicable to all B&W plants, the licensee was requested (Ref. 17) to provide supplemental calculations for TMI-1 similar to those provided in Reference 16. The licensee's responses (Refs. 18, 19 and 20) culminated in a supplemental ECCS calculation (Ref. 21) for TMI-1. This calculation not only considers cladding swelling and rupture effects, but also considers the fuel densification effects with a more recent B&W fuel performance code called TACO-2 (Ref. 22). The combined analysis results in low core elevation PLHR limits which are more restrictive than those which consider only fuel densification (with TACO-1).

In general, the supplemental calculation utilizes previously approved methods except for the substitution of the NRC cladding models. However, there are segments of the analysis (e.g. THETA1-B-Ref. 23) that are currently undergoing NRC review. The licensee has also presented results from a calculation using

a new FLECSET heat transfer correlation (Refs. 24 and 25). This correlation appears to offset the NUREG-0630 penalties. However, the benchmarking and other final evaluations of FLECSET have not been completed and provided to the NRC for review. Because the FLECSET correlation has not yet been approved, the licensee has committed (Ref. 22) to administratively implement the more restrictive operating limits during early in Cycle 5 operation (i.e., less than 31 EFPD and greater than 80% power). Because of the planned Cycle 5 power ascension schedule, it is expected that these limits will not affect actual operation.

Considering the above, we conclude that the licensee's proposed administrative limits on operation are both appropriate and necessary. Since these operating limits are more restrictive than those previously proposed (Ref. 13) for TMI-1, since they are only needed for a brief time period, and since potential but unused compensating benefits may exist, we therefore conclude that the administrative limits imposed on an interim basis are acceptable for incorporating the NUREG-0630 penalties until our final evaluation of FLECSET is completed.

4.0 Other Modifications to the Technical Specifications

The licensee has also requested (Ref. 13) several other minor modifications to the plant Technical Specifications. These are discussed below.

The bases for the Technical Specifications incorrectly retained a centerline fuel melt limit based on a fuel assembly design no longer used at TMI-1. The revised fuel melt limit is based on the fuel design (Mark B-4) used in the Cycle 5 core. We find this change acceptable.

The licensee proposes to change the nuclear power setpoint based on the reactor coolant pump monitors (Item 3 of Table 2.3.1 - RPS Trip Setting Limit) from 91% to 55% of rated power for 1/1 (one pump in each loop) RC pump operation. The pump monitor's power setting is intended to prevent the core minimum Departure from Nucleate Boiling Ratio (DNBR) from decreasing below the 1.3 limit by tripping the reactor due to the loss of RC pumps. This is a redundant trip function complemental to the flux/flow trip setting where the power level trip setpoint is based on a power to flow ratio adjusted for the power imbalance. The existing 91% setpoint was established by the original Final Safety Analysis Report. The licensee has indicated that an evaluation accounting for changes in DNBR-related design factors since Cycle 1 has shown the 91% limit to remain conservative for Cycle 5 operation. The decision to reduce the setpoint from 91% to 55% of rated power is based on B&W recommendations for the 177 fuel assembly plants with redundant pump monitors to provide a consistent basis for any future B&W analyses. This change will not significantly impact anticipated operations of TMI-1. Since the proposed change is in a more conservative direction, we conclude that it is acceptable.

The Technical Specifications have also been rewritten so that the quadrant tilt is always determined by the most accurate detector system available. This does not represent a change in the existing allowable tilt limits for any of the detector systems. We find this administrative change acceptable.

5.0 Summary

From our reexamination of the Cycle 5 Reload Application, and from our review of additional information submitted by the licensee, we conclude that this core reload will not adversely affect the Three Mile Island Nuclear Station's ability to operate safely during Cycle 5 of Unit 1.

6.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5 (d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

7.0 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.

Dated: October 28, 1983

The following NRC personnel have contributed to this Safety Evaluation: J. Voglewede, Y. Hsii, L. Kopp.

REFERENCES

1. R. Starostecki and D. G. Eisenhut (NRC) memorandum to R. Jacobs et al. (NRC) on "TMI-1 Restart Issues Task Force" dated April 15, 1983.

2. J. G. Herbein (MetEd) letter to R. W. Reid (NRC) on "Technical Specification Change Number 86" dated December 28, 1978.

3. J. G. Herbein (MetEd) letter to R. W. Reid (NRC) on "Cycle 5 Reload - Additional Information" dated March 1, 1979.

4. R. W. Reid (NRC) letter to J. G. Herbein (MetEd) on "Amendment 50 to Facility Operating License No. DPR-50" dated March 16, 1979.

5. J. H. Taylor (B&W) letter to L. S. Rubenstein (NRC) dated September 5, 1980.

6. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) dated October 28, 1980.

7. T. M. Novak (NRC) letter to H. D. Hukill (GPU) dated July 13, 1982.

8. C. D. Morgan and H. S. Kao, "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," Babcock and Wilcox Company Report BAW-10044, May 1972.

9. R. H. Stoudt et al., "TACO: Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10087P-A, Rev. 2, August 1977.

10. D. F. Ross, Jr. (NRC) letter to J. H. Taylor (B&W) dated January 18, 1973.

11. J. H. Taylor (B&W) letter to L. S. Rubenstein (NRC) dated September 5, 1980.

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