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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 142 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 April 5, 1988 (Ref. 1), GPU Nuclear Corporation (GPUN) submitted an application to reload Unit No. 1 of the Three Mile Island (TMI) Nuclear Generating Station and operate it for a seventh cycle. To support the application, GPUN submitted report BAW-2015 (Ref. 2) entitled "Three Mile Island Unit 1 Cycle 7 Reload Report" and proposed changes to the Unit 1 Technical Specifications.

The Cycle 7 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rods, and one incore instrument guide tube. Cycle 7 is to have an operating length of approximately 445 effective full power days (EFPD). Cycle 7 will be operated in a rods out, feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length silver-indium-cadmium (Ag-In-Cd) control rods and 68 burnable poison rod assemblies (BPRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution.

Although the licensed core full power level is 2535 megawatts-thermal (MWt), the Cycle 7 analyses were performed at a core power level of 2568 MWt. By letter dated April 18, 1988 (Ref. 16), GPUN submitted a request for an increase in the licensed rated power from 2535 MWt to 2568 MWt for TMI-1. This is also evaluated in part, herein, and will be the subject of a separate amendment and safety evaluation.

2.0 EVALUATION

2.1 EVALUATION OF FUEL SYSTEM DESIGN

Cycle 7 will contain 36 fresh (unirradiated) Mark B4 fuel assemblies with a U-235 enrichment of 2.85 weight percent (Batch 9A), four fresh Mark B4 assemblies with a 2.95 weight percent U-235 enrichment (Batch 9B) and 36 fresh Mark B4Z fuel assemblies with a 3.63 weight percent U-235 enrichment (Batch 9C). The remainder of the core will contain 12 Mark B4 once-burned Batch 8A assemblies, 64 once-burned Batch 8B assemblies and 25 twice-burned Batch 7 assemblies. All of these fuel assemblies are mechanically interchangeable. The Batch 9C Mark BZ assembly design is similar to the Mark B4 fuel assembly except that the six intermediate Inconel spacer grids have been replaced with zircaloy grids.

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Although the Mark BZ fuel design (Ref. 3) has been reviewed and approved by the NRC (Ref. 4), the NRC safety evaluation states that a licensee incorporating this design is required to submit a plant-specific analysis of combined seismic and loss of coolant accident (LOCA) loads according to Appendix A of Standard Review Plan 4.2 (Ref. 5). The licensee has verified that the analysis that was presented in the Rancho Seco Cycle 7 reload report (Ref. 3) envelopes the TMI-1 plant design requirements and, therefore, the margin of safety reported for the Mark BZ fuel is applicable to TMI-1. Therefore, the staff concludes that the Mark BZ assemblies satisfy the above mentioned NRC requirement for Cycle 7.

The pin prepressure in some of the Batch 9 fuel assemblies has been lowered by 50 psi in order to provide a higher burnup limit for pin pressure but may be limiting in terms of cladding collapse. The licensee has stated that the cladding collapse time for the most limiting Cycle 7 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 7 assembly. The methods and procedures used for the analyses (Ref. 6) have been previously reviewed and approved by the staff. The staff concludes that cladding collapse has been appropriately considered and will not occur for Cycle 7 operation.

All other fuel rod thermal and mechanical analyses were also performed with previously approved methodology and the results were within the design criteria, including carability to centerline melt and internal pin pressure.

Based on the fact that approved methods have been used and fuel design criteria are all met, the staff finds the fuel design for Cycle 7 acceptable.

2.2 EVALUATION OF NUCLEAR DESIGN

The nuclear design parameters characterizing the TMI-1 Cycle 7 core have been computed by methods previously used and approved for Babcock and Wilcox (B&W) reactors (Ref. 7). Comparisons have been made between the parameters for Cycle 6 and Cycle 7. Core design changes including a core power level increase to 2568 MWt, an increase in cycle length to 445 ± 15 EFPD, as well as U-235 enrichment and shuffle pattern differences between cycles account for the differences in control rod worths, critical boron concentrations, Doppler coefficients, and moderator temperature coefficients (MTCs). The low neutron leakage Cycle 7 design is consistent with the GPUN reactor vessel fluence reduction efforts for TMI-1 as described in their response on the Pressurized Thermal Shock Rule 10 CFR 50.61 (Ref. 8).

The fresh Batch 9C Mark BZ fuel will have an initial enrichment of 3.63 weight percent U-235. The staff finds this acceptable since the TMI-1 spent fuel pool has been designed to store fuel with a maximum enrichment of 4.3 weight percent U-235.

Shutdown margin calculations for Cycle 7 include the effects of poison material depletion, a 10% calculational uncertainty, allowance for rod bite, the power deficit in going from hot full power (HFP) to hot zero power (HZP), and neutron flux redistribution as well as a maximum worth stuck rod. Beginning of cycle (BOC) and end of cycle (EOC) shutdown margins show adequate reactivity worth exists above the total required worth during the cycle. Shutdown margins at BOC and EOC are 4.2% delta k/k and 3.0% delta k/k, respectively, compared to the minimum required value of 1.0% delta k/k.

Based on its review, the staff concludes that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of TMI-1 Cycle 7 is acceptable.

2.3 EVALUATION OF THERMAL HYDRAULIC DESIGN

Although a full Mark BZ core and a full Mark B core provide practically the same departure from nucleate boiling (DNB) margin for both steady-state and transient conditions (Ref. 4), incompatibility in the hydraulic characteristics has an effect on thermal margin during transitional mixed core cycles when both Mark BZ and Mark B fuel assemblies co-exist in the core. Since the Mark BZ assemblies have a higher hydraulic resistance due to the BPRA retainers and the zircaloy intermediate spacer grids, some of the coolant flow is diverted from the Mark BZ fuel to the lower-powered Mark B fuel. The fact that the Mark BZ assemblies have less flow in a mixed core results in lower maximum allowable power peaking and a lower enthalpy rise factor required in order to maintain the same DNBR limit compared to a whole core of Mark BZ fuel. The licensee, therefore, performed a bounding thermal-hydraulic design analysis in which a full Mark BZ core and a core bypass flow of 8.8% were assumed. The DNB results were compared to an analysis using the actual mixed core configuration and bypass flow (7.6%) and found to be bounding. Therefore, a transition core penalty due to the introduction of Mark BZ assemblies is not required for Cycle 7.

For Cycle 7, the BWC critical heat flux correlation (Ref. 9) was used for analysis of the Mark BZ fuel assembly instead of the B&W-2 correlation used in Cycle 6. The BWC correlation has been reviewed and approved by the staff and has been found to be applicable to the Mark BZ design.

Based on the fact that the licensee's thermal-hydraulic analyses were performed using approved analytical methods and correlations and resulted in acceptable performance, the staff finds the thermal-hydraulic design of Cycle 7 acceptable.

2.4 ACCIDENT AND TRANSIENT ANALYSIS

The important physics, thermal-hydraulic, and kinetics parameters for Cycle 7 have been compared to the values used in the FSAR (Ref. 10), fuel densification report (Ref. 11), reference cycle and/or the generic LOCA analyses (Refs. 12, 13, & 14). Although some Cycle 7 values are not bounded by those previously used, the licensee has determined that the initial conditions defined by these parameters would produce less severe transients than the initial conditions assumed in the reference analyses and, therefore, no reanalysis was necessary.

The consequences of certain transients and accidents are not affected by physics, thermal-hydraulic, or kinetics parameters but rather by radiological considerations due to core isotopic inventory changes. Although the radionuclide inventory generated at a bounding power level of 2568 MWt was found to be only slightly greater than that obtained at a power level of 2535 MWt (Cycle 6), the licensee conservatively assumed a 10% increase in the Cycle 7 core fission product inventory in reevaluating the most adversely affected events. All of the resulting Cycle 7 accident doses were well below the dose acceptance criteria based on 10 CFR 100.

The important cycle specific parameters for Cycle 7 have also been compared to the limiting values used in the generic LOCA analyses and have been found to be bounded. Therefore, adherence to the linear heat rate (LHR) limits for Cycle 7 given in Table 7-2 of the Reload Report assures that the emergency core cooling system (ECCS) Final Acceptance Criteria will be met.

Based on the safety analysis review, the staff finds that the consequences of transients and accidents during Cycle 7 meet all safety criteria and are acceptable.

2.5 TECHNICAL SPECIFICATION CHANGES

The TMI-1 Cycle 7 Technical Specifications have been modified to support a longer fuel cycle length (445 EFPD) as well as various operational and design changes. These include changes in power peaking and control rod worths and the removal of the variable low pressure trip as well as incorporation of a low leakage fuel design, mixed Mark B/Mark BZ fuel, and a power level upgrade from 2535 MWt to 2568 MWt.

Changes were made to the following Technical Specification items:

(a) core protection safety limit pressure/temperature curves;
(b) core protection safety limit axial power imbalance limits;
(c) protection system maximum allowable setpoints;
(d) power level dependent quadrant tilt setpoints;
(e) overpower trip setpoint at 50% power or less;
(f) rod position setpoints;
(g) axial power imbalance envelope for operation;
(h) LOCA limited maximum allowable LHR;
(i) maximum allowable enrichment of Cycle 7 and future reload fuel;
(j) BWC correlation with DNBR limit of 1.18 for Mark BZ fuel.

In addition, various administrative and editorial changes were made.

The staff has reviewed the proposed changes (Ref. 15) and finds them acceptable because they have been derived from analyses performed using approved methods and have been appropriately considered in the Cycle 7 safety analyses.

2.6 RATED POWER UPGRADE

As shown above, the staff has found the proposed Cycle 7 reload and the associated modified Technical Specifications acceptable. The Cycle 7 core characteristics and Technical Specification limits were developed for a full power level of 2568 MWt or higher and, therefore, the proposed power upgrade does not change the original design conditions. In addition, the staff concludes that the power upgrade effect on reactor vessel accumulated fluence is acceptable.

The staff has reviewed the high pressure injection (HPI) flow split of 64% to the core and 36% out a cold leg discharge break which was justified in the TMI-1 Restart Report based on a rated power of 2535 MWt. Although the B&W generic small break LOCA analysis, which was performed at a rated power of 2772 MWt, used an HPI flow split of 70% - 30%, the 64% - 36% flow split was reevaluated for the requested increased rated power of 2568 MWt. Based on this reevaluation, which demonstrated that the TMI-1 HPI system will deliver as much water to the core as the generic LOCA analysis assumed during the time period of concern, the staff concludes that TMI-1 has sufficient HPI capacity at a rated power of 2568 MWt.

TMI-1 has an estimated natural circulation cooldown time of 22 hours (at 10°F/hr). Since the condensate-grade feedwater supply has sufficient inventory to support a cooldown time in excess of 100 hours, the staff concludes that this large margin assures that a natural circulation cooldown will not be affected by the proposed small increase in rated power.

The design basis safety analyses of flooding from plant sources assumed a flow rate greater than that expected to support operation at 2568 MWt. Since the flood level is limited by the amount of water available to be pumped into the building, and the upgraded power level will not change the available water inventory, the staff concludes that the maximum FSAR predicted flood level will not change due to the proposed power uprate.

The proposed upgraded power level will not cause a change in either the primary system or secondary system available water inventory. Since the flood level is limited by the amount of primary/secondary water available to be pumped into the building, the staff concludes that the maximum predicted flood levels from either a primary or secondary break will not change due to the upgraded power level.

Based on the Cycle 7 reload evaluation and the design basis safety analyses evaluations discussed above, the staff concludes that the proposed power uprate does not change the original design conditions and that all existing reactor design and safety criteria are preserved at the upgraded power level of 2568 MWt. Further evaluation of this power uprate will be contained in a separate safety evaluation to be issued in support of an amendment approving the power upgrade.

2.7 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, thermal-hydraulic, and accident information presented in the TMI-1 Cycle 7 reload report and finds the proposed reload and the associated modified Technical Specifications acceptable. Based on this evaluation and the separate safety evaluation supporting the amendment approving the power upgrade, the staff also finds that Cycle 7 can be operated at a rated core power of either 2568 MWt or at the existing rated power level of 2535 MWt without exceeding the established safety criteria.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact relating to the proposed license amendment was published in the Federal Register on July 18, 1988 (53 FR 27092).

Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 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: July 18, 1988

Principal Contributor: Lawrence I. Kopp

REFERENCES

1. Letter from H. D. Hukill (GPUN) to USNRC, C311-88-2033, Technical Specification Change Request No. 182, Cycle 7 Reload, April 5, 1988.

2. "Three Mile Island Unit 1, Cycle 7 Reload Report," BAW-2015, March 1988.

3. "Rancho Seco Cycle 7 Reload Report, Volume 1, Mark BZ Fuel Assembly Design Report," BAW-1781P, April 1983.

4. Letter from J. F. Stolz (NRC) to Sacramento Municipal Utility District (SMUD), "Rancho Seco Nuclear Generating Station, Evaluation of Mark BZ Fuel Assembly Design," November 16, 1984.

5. "Standard Review Plan," NUREG-0800, Revision 2, July 1981.

6. "Program to Determine In-Reactor Performance of B&W Fuels, Cladding Creep Collapse," BAW-10084A, Revision 2, October 1978.

7. "NOODLE-A Multi-Dimensional Two-Group Reactor Simulator," BAW-10152-A, June 1985.

8. Letter from GPUN to USNRC, 5211-86-2007, January 23, 1986.

9. "BWC Correlation of Critical Heat Flux," BAW-10143P-A, April 1985.

10. "Three Mile Island Nuclear Station, Unit 1, Final Safety Analysis Report," USNRC Docket No. 50-289.

11. "Three Mile Island Unit 1 Fuel Densification Report," BAW-1389, June 1973.

12. "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103-A, Rev. 3, July 1977.

13. "TACO2 Loss-of-Coolant Accident Limit Analysis for 177-FA Lowered Loop Plants," BAW-1775, Rev. 0, February 1983.

14. "Bounding Analytical Assessment of NUREG-0630 Models on LOCA kW/ft Limits With Use of FLECSET," BAW-1915P, May 1986.

15. "Technical Specification Change Request No. 182, Cycle 7 Reload," Attachment to Reference 1.

16. Letter from H. D. Hukill (GPUN) to USNRC, C311-88-2036, Technical Specification Change Request No. 184, Rated Power Upgrade, April 18, 1988.