METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request No. 182

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY: H D Hukill
Vice President & Director, TMI-1

Sworn and Subscribed
to before me this 5th
day of April, 1988.

Sharon P. Brown
Notary Public

SHARON P. BROWN, NOTARY PUBLIC
MIDDLETOWN BORO, DAUPHIN COUNTY
MY COMMISSION EXPIRES JUNE 22, 1989
Member, Pennsylvania Association of Notaries

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF
GPU NUCLEAR CORPORATION

DOCKET NO. 50-289
LICENSE NO. DPR-50

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 182 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposit in the United States mail, addressed as follows:

Mr. Kenneth E. Witmer, Chairman
Board of Supervisors of
Londonderry Township
25 Roslyn Road
Elizabethtown, PA 17022

Ms. Sally S. Klein, Chairman
Board of County Commissioners
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Mr. Thomas Gerusky, Director
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GPU NUCLEAR CORPORATION

BY: H. D. Hukill
Vice President & Director, TMI-1

DATE: April 5, 1988

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 182

GPUN requests that the following changed replacement pages be inserted into the existing Technical Specifications:

Revised pages: i, vi, vii, viii, 1-5, 2-1, 2-2, 2-3, 2-5, 2-6, 2-7, 2-8, 2-9, 2-10, 3-34, 3-34a, 3-35, 3-35a, 3-36, 3-36a, 5-4

Revised figures: 2.1-1, 2.1-2, 2.1-3, 2.3-1, 2.3-2, 3.5-2A, 3.5-2B, 3.5-2C, 3.5-2D, 3.5-2E, 3.5-2F, 3.5-2G, 3.5-2H, 3.5-2I, 3.5-2J, 3.5-2K, 3.5-2L, 3.5-2M

These pages are attached to this change request.

II. REASON FOR CHANGE

This change is requested to provide Technical Specifications for operation of TMI-1 Cycle 7.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The attached Reload Report (BAW-2015, March 1988) and the revised Technical Specifications provide the design and administrative bases to support the conclusion 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.

Core Design

Cycle 7 has been designed as a typical B&W lumped burnable poison rod core based on an in-out-in fuel management strategy. The low neutron leakage cycle design is consistent with GPUN reactor vessel fluence reduction efforts for TMI-1 as described in our response on the Pressurized Thermal Shock Rule - 10 CFR 50.61 (GPUN letter 5211-86-2007, January 23, 1986).

The reference cycle for core analyses is Cycle 6 which is scheduled for completion in June 1988 after 425 ±15 EFPD. Operation of Cycle 7 is scheduled to start in August 1988 with a design cycle length of 445 ±15 EFPD.

Core Loading

Cycle 7 loading of Batch 9 includes 36 fresh Mark B4 fuel assemblies at 2.85 weight percent enrichment (Batch 9A), 4 fresh Mark B4 assemblies at 2.95 weight percent enrichment (Batch 9B) and 36 fresh Mark B4Z fuel assemblies at 3.63 weight percent enrichment (Batch 9C). The remainder of the core is comprised of 12 once-burned Batch 8A assemblies with an initial enrichment of 2.95 weight percent and 64 once-burned Batch 8B and 25 twice-burned Batch 7 assemblies with an initial enrichment of 2.85 weight percent.

Reactivity Control

Cycle 7 will be operated in a feed and bleed mode. Reactivity is controlled by 61 full-length Ag-In-Cd control rods, 68 burnable poison rod assemblies and soluble boron control. Control rod group locations and designations remain the same as for Cycle 6.

Also as in Cycle 6, Cycle 7 will utilize advanced design "gray" axial power shaping rods which decrease fuel duty due to local power changes while providing adequate power peaking and imbalance control.

Fuel System Design

The Batch 9 fuel assemblies are of the B&W Mark B 15 X 15 design used in previous cycles. Two changes are implemented in Batch 9: 1) 36 of the new assemblies (9C) are the Mark B4Z design which utilizes Zircaloy intermediate spacer grids; the remaining 40 reload assemblies are of the same Mark B4 Inconel grid design used in Batches 7 and 8. 2) 38 of the new assemblies (9C and 2 FAs in 9B) consist of fuel rods with lower fill-gas prepressure than the Batch 8 rods.

The Mark BZ design was described in BAW-1781P and approved by the NRC for use in B&W lowered loop 177-FA plants (J. F. Stolz to SMUD, November 16, 1984). A condition of NRC approval for use of the Mark BZ design was that a plant-specific analysis of combined seismic and LOCA loads according to Appendix A to Standard Review Plan 4.2 be submitted. It was verified that the generic seismic/LOCA analysis as described in BAW-1781P is applicable to TMI-1 and that the analysis provides safety margin for TMI-1 plant design requirements.

Full reloads of the Mark BZ assembly have been operating in Rancho Seco and Oconee 1, 2 and 3 over several cycles with no anomalous performance.

The reduced prepressure fuel rod design is expected to provide greater burnup and LOCA KW/ft margins for future application. The lower internal rod pressure was accounted for in all mechanical analyses. Otherwise, the Batch 9 rods are essentially identical to designs used in previous TMI-1 fuel batches.

As described in the Reload Report, all fuel rod thermal and mechanical analyses were performed using previously-approved codes. Results are all within the design criteria, including clad creep collapse, capability to centerline melt and internal pin pressure. It was also confirmed that the results of the TMI-1 Densification Report (BAW-1389) remain bounding for all Cycle 7 fuel since those analyses were based on a lower initial pellet density. Resinter testing on all pellet fabrication lots confirm the conservatism of the densification characteristics assumed in the analyses.

Nuclear Design

Core design changes for Cycle 7 include a core power level of 2568 MWt and the increase in cycle lifetime to 445 ±15 EFPD. Nuclear design calculations were performed using approved codes.

The longer cycle length and differences in the shuffle pattern and BPRA loading resulted in some differences in the physics parameters between Cycle 6 and 7 as shown on Table 5-1 of the Reload Report. The Cycle 7 physics parameters are considered to be reasonable and consistent with the core design changes. The BOC and EOC shutdown margin shown in Table 5-2 of the Reload Report have increased from the Cycle 6 value and meet the 1% Δ k/k margin criteria.

Thermal-Hydraulic Design

The Cycle 7 core is hydraulically and geometrically similar to Cycle 6. One difference is the implementation of the 36 Batch 9C Mark-BZ fuel assemblies in Cycle 7. Compatibility of the Mark-BZ assembly with the Mark B design was demonstrated in BAW-1781P, as was the applicability of the approved BWC critical heat flux correlation (BAW-10143P-1). To account for small flow resistance differences between the Mark-BZ and Mark-B assemblies in the mixed transition core a bounding thermal-hydraulic design analysis was performed. The analysis assumed a full Mark BZ core and a conservative core bypass flow of 8.8%. DNB results of this analysis were compared to an analysis using the actual mixed core configuration and bypass flow (7.6%) and determined to be bounding. A transition core penalty due to introduction of the Mark BZ assembly design, therefore, is not required for Cycle 7.

Accidents and Transient Analysis

The changes in Cycle 7 parameters, including analysis at 2568 MWt, have been evaluated to determine effects on the FSAR accident analyses.

For each accident or transient the key core thermal, thermal-hydraulic and kinetic parameters for the cycle were compared to FSAR, TMI-1 Densification Report, reference cycle and/or generic LOCA analyses values to determine whether the event remained bounded by the previous analyses.

Core thermal and thermal-hydraulic parameters are common to all accidents. Cycle-specific parameters in both of these categories were shown to be conservatively bounded by previous analyses in Sections 4 and 6, respectively, of the Reload Report. Key kinetics parameters for Cycle 7 are compared to those assumed in the reference analyses in Table 7-1 of the Reload Report. For all reload-dependent accidents, the initial conditions defined by the cycle-specific parameters will produce less severe transients than the initial conditions assumed in previous analyses. Cycle 7 transients are thus bounded by previously accepted analyses, and no reanalysis was necessary. The accidents examined included:

Startup Accident
Rod Withdrawal Accidents
Moderator Dilution Accident
Cold Water Accident
Loss-of-Coolant Flow
Stuck-out, Stuck-in or Dropped Control Rod
Steam Line Failure
Feed Line Failure
Rod Ejection Accident
Uncompensated Operating Reactivity Changes

Two Cycle 7 kinetics parameters were noted to be in an undesirable direction for certain accident evaluations. In each case these were determined not to be the controlling parameter for the event.

The EOC Doppler coefficient is more negative than the value used in the FSAR. Although this is not in the conservative direction for the dropped rod accident, the transient results are still conservative with respect to the FSAR analysis as a result of the smaller maximum dropped rod worth and the less negative moderator coefficient for this cycle. Also, the moderator effect is more than an order of magnitude greater than the Doppler.

The EOC moderator temperature coefficient is slightly less negative than the FSAR value. Although this is not in a conservative direction for the ejected rod accident, this rapid reactivity transient is controlled by the negative Doppler coefficient which is conservative. Also, the smaller Cycle 7 ejected rod worth assures the event is within the FSAR analysis.

The consequences of certain transients and accidents are only affected by a reload if core isotopic inventory changes, i.e., they are not affected by thermal or reactivity parameters but rather by radiological considerations. Acceptability is based on offsite doses. These events include:

Loss of Electric Power
Steam Generator Tube Failure
Fuel Handling Accident
Maximum Hypothetical Accident
LOCA Analysis
Waste Gas Tank Rupture
Fuel Cask Drop

Since the potential radiological releases are based on Technical Specification limits, core burnup, or core power rating, the consequences of these events normally need not be reevaluated. As discussed in Section 7 of the Reload Report, the Cycle 7 radionuclide inventory generated at a bounding power level of 2568 MWt was compared to the Cycle 6 inventory (2535 MWt) and found to be slightly greater consistent with the 1.3% power difference. This confirmed that increases in Cycle 7 dose values would not be significant.

To bound future cycle design variations the Cycle 7 core fission product inventory was increased by 10% and the FSAR Chapter 14 accidents were reevaluated using the increased source term. The analysis also incorporated more current plant parameters than those used in the FSAR. Results for all accidents were well below the acceptance criteria of 10CFR100.

Small and large break Loss Of Cooling Accident (LOCA) analyses are primarily dependent upon overall system response rather than reload core characteristics. Cycle 7 key core thermal and reactivity parameters have been compared to the limiting values used in the generic analyses done for these accidents and determined to be bounded. Therefore, the ECCS Final Acceptance Criteria will be met by observance of the linear heat rate (LHR) limits for Cycle 7 shown in Table 7-2 of the Reload Report. The effects of NUREG-0630 cladding swelling and rupture models and the FLECSET reflood heat transfer correlation have been included in the calculation of the bounding LHRs.

Modifications to Technical Specifications

The Technical Specifications were revised for Cycle 7 operation to account for changes in power peaking and control rod worths, and the DNB margins provided by use of the LYNXT crossflow thermal-hydraulics model.

The pressure-temperature core safety limits were recalculated for Cycle 7 using the BWC CHF correlation with the approved LYNXT crossflow code (BAW-10156-A). Results demonstrated that the new safety limits (Figures 2.1-1 and 2.1-3 of the Technical Specifications) are outside the high temperature and low pressure limits (Figure 2.3-1) at all points, thus justifying elimination of the DNB-based variable low pressure trip setpoint used in previous cycles.

For Cycle 7 the rod position curves and the axial power imbalance envelopes for operation are presented as error-adjusted curves in the Technical Specifications (Figures 3.5-2A through 3.5-2L).

Analysis was performed for defining power level-dependent quadrant tilt setpoints. The analysis verified the use of an actual core tilt of 7.50% when power is less than or equal to 50% Full Power (FP) and an actual core tilt of 4.92% when power is greater than 50% FP. The error-adjusted tilt values are presented in Table 3.5-1A in the Technical Specification.

Brief explanations of the Cycle 7 Technical Specification changes are given below:

Pages i, vi, vii and viii

Administrative changes to the Table of contents.

Page 1-5

Changes "reactor power imbalance" to "axial power imbalance." This is an editorial change to be consistent with B&W and industry terminology.

Page 2-1

Changes "reactor power imbalance" to "axial power imbalance."

Incorporates the application of the BWC correlation of critical heat flux for the Mark BZ fuel into the bases.

Page 2-2

First paragraph, eliminates reference to specific actual vs. indicated core pressure difference used in the analysis because value given no longer reflects current methods. Current analyses use various pressure differences depending on pump operating condition being analyzed.

The description of the P-T curve on Figure 2.1-1 is changed to eliminate definition of the power level and reactor coolant flow. The bases now state that the most limiting combination of these parameters was used to generate the figure.

Eliminates paragraph comparing 1.65 cosine shape axial peak to a 1.7 lower core peak because the discussion does not pertain to current power peak methodology.

Incorporates the application of the BWC correlation of critical heat flux for the Mark BZ fuel into the bases.

Change "reactor power imbalance" to "axial power imbalance."

Page 2-3

Incorporates the application of the BWC correlation of critical heat flux for the Mark BZ fuel into the bases.

Figure 2.1-3 (Core Protection Safety Limit)

This figure was derived from the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3.

Figure 2.1-2 (Core Protection Safety Limits)

The figure establishes imbalance windows because power peaking is not a directly observable parameter. The limits preserve the center-line fuel melt (CFM) and/or DNBR criteria always applying the more restrictive condition to establish the limit. The curves are derived from Figure 2.3-2 (RPS Setpoints) by removal of instrumentation error and calculational uncertainties in flux/flow determinations.
The Cycle 6 figure had a maximum power limit for the 2, 3, and 4 pump operation curve of 62.6% FP while the Cycle 7 value is 62.0% FP. A review of the Cycle 6 and Cycle 7 analysis determined that the Cycle 6 value should have been 62.0% FP. The 62.6% FP value was a typographical error on the figure, and the safety margin and plant operation were unaffected.

Figure 2.1-3 (Core Protection Safety Bases)

This figure has been generated using the BWC correlation of critical heat flux and the B&W crossflow thermal-hydraulic model LYNXT. This figure was previously generated using a closed-channel thermal-hydraulics model. The new curves represent the conditions at which either the DNBR limit is predicted at the maximum possible

thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent (B&W-2) or 26 percent (BWC), whichever condition is more restrictive. This is the same criteria used to generate previous cycle curves, thus the margin of safety remains unchanged. In the closed channel model results the three pump curve was based on the DNBR limit; while in the crossflow model results the three pump curve is based on the quality limit.

Pages 2-5 and 2-6

Changes "reactor power imbalance" to "axial power imbalance."

Incorporates the application of the BWC correlation of critical heat flux for the Mark BZ fuel into the bases.

Page 2-7

Changes "reactor power imbalance" to "axial power imbalance."

Incorporates the application of the BWC correlation of critical heat flux for the Mark BZ fuel into the bases.

The new safety limits (see Figures 2.1-1 and 2.1-3) have allowed the low pressure and high temperature trip functions to replace the variable low pressure trip function. By implementing this change the margin of safety is increased because the potential number of unnecessary reactor trips will be decreased.

Reference 7 for ECCS analysis was added as a pressure reduction accident.

Page 2-8

Editorially revised only to accommodate previous page changes.

Page 2-9

Reference 4 is changed to correct the FSAR section. Reference 7 is corrected to update the ECCS BAW Topical revision number. Editorially revised to accommodate previous page changes. Table 2.3-1 previously located on page 2-9 has been relocated to page 2-10.

Table 2.3-1, Page 2-10 (Reactor Protection System Trip Setting Limits) The table has been changed to remove the variable low pressure trip setpoint. The footnotes were also renumbered.

Figure 2.3-1 (TMI-1 Protection System Maximum Allowable Setpoints)

This figure reflects the replacement of the variable low pressure trip setpoint with the low pressure and high temperature trip setpoints. This setpoint was able to be removed because the DNB margin provided by the implementation of the LYNXT cross flow thermal-hydraulics model moved the safety limits (see Figures 2.1-1 and 2.1-3) outside of the setpoints shown on this figure.

Figure 2.3-2 (Protection System Maximum Allowable Setpoints for Axial Power Imbalance)

This figure conservatively establishes power and imbalance limits based on Cycle 7 analyses. A flux/flow setpoint of 1.08 has been maintained to provide additional margin to the Cycle 7 pump coastdown analysis which would allow a flux/flow limit of approximately 1.13. The 3 and 2 pump overpower values are ratioed from the 4 pump 1.08 value based on flows of 74.7% and 49.2% respectively, of design flow. The imbalance setpoints conservatively envelope the thermal limits.

Page 3-34

Sections 3.5.2.4.a/b/c reflect the introduction of Table 3.5-1A to indicate the quadrant tilt setpoint limits.

Section 3.5.2.4.d was clarified by adding the word "allowable" prior to power. This clarification better identifies which power must be reduced when the tilt limit is exceeded.

Page 3-34a

Section 3.5.2.4.e.1 has added the specification that allows reduced overpower trip setpoint of 60% full power when thermal power is 50% full power or less with 4 RC pumps operating. This addition maintains similar margin of safety with regard to the RPS power/imbalance envelope trip setpoints. It increases overall safety by eliminating the need to enter the RPS cabinets and reduce setpoints. This will decrease the potential for an unnecessary reactor trip or human error.

Section 3.5.2.4.e.3: Adds reference to a third figure for the burnup-dependent operational imbalance limits.

Table 3.5-1A has been added because of the implementation of power-dependent quadrant tilt limits in Cycle 7. Additional analysis was performed in Cycle 7 to verify that an actual tilt of 7.50% is acceptable when power is less than or equal to 50% full power. Since tilt is an indirect measure of the peaking increase higher tilts can

be allowed at lower powers while maintaining the same margin to the peaking limits. The actual tilt of 4.92% will continue to be used when power is greater than 50% full power. The actual tilt values are then error-adjusted to obtain the values provided in the table. The implementation of the power-dependent tilt limits will increase the margin of safety by reducing the potential for unnecessary reactor trips. If a tilt limit is exceeded the RPS setpoints for imbalance must be adjusted and this increases the potential for a reactor trip or human error.

Page 3-35

Changes "core imbalance" to "axial power imbalance."

Adds reference to the third figure for the burnup dependent operational imbalance limits.

Changes the figure number for the LOCA Allowable LHR limits due to the addition of a third figure for the burnup-dependent operational imbalance limit.

Page 3-35a

Adds the third figure for the burnup-dependent operational imbalance limits.

Changes "power imbalance" to "axial power imbalance."

Identifies that the operational imbalance limit figures provided in the Technical Specifications are now error-adjusted for observability and measurement uncertainties. This change was made for consistency between the Technical Specification limits and the alarm setpoints.

Page 3-36

Removes the words "xenon transient override" with regard to the Group 7 regulating bank function because the core is now operated in a feed-and-bleed mode.

Adds the word "rod" in the description of the Group 8 APSR bank function.

Identifies that the rod position limit figures provided in the Technical Specifications are now error-adjusted for observability and measurement uncertainties. This change was made for consistency between the Technical Specification limits and the alarm setpoints.

Page 3-36a

Implements the power-dependent quadrant tilt limits.

Adds allowance for reduction of the nuclear overpower trip setpoint to 60% full power when thermal power is 50% full power or less.

Figures 3.5-2A to 3.5-2I (Rod Position Setpoints for 4, 3, and 2 Pump Operation)

These figures were established based on Cycle 7 analyses to preserve shutdown margin, ejected rod worth, and LOCA analysis criteria. Rod limits for 2 and 3 pump operation were developed by power-scaling from 4 pump values.

The "Not Allowed" region is defined by the 1% Δ k/k shutdown margin criteria. This region has been set based on actual Cycle 7 analyses. Ejected rod limits are bounded by the shutdown margin limits. The "Restricted" region is defined by the LOCA criteria. In Cycle 6 this region was limited by thermal-hydraulic accident criteria. For Cycle 7 LOCA criteria are limiting. Figures 3.5-2A, 3.5-2D, and 3.5-2G include the beginning-of-cycle effects of recent ECCS model changes.

These figures are now presented as the error-adjusted alarm setpoints. In previous cycles these figures represented the actual rod position and power while the error-adjusted alarm setpoints were placed in the procedures. Placing the error-adjusted alarm setpoints into the Technical Specifications will simplify operator use and the figures become consistent with other error-adjusted alarm setpoints (e.g., tilt, RPS Trips).

Figures 3.5-2J to 3.5-2L (Axial Power Imbalance Envelope for Operation)

These figures were developed in conjunction with the rod position setpoints to provide imbalance limits that preserve the transient analyses. The limits are based on Cycle 7 analyses with the LOCA criteria being the bounding criteria.

These figures are now presented as error-adjusted alarm setpoints as described above for rod position limits.

Figure 3.5-2M (LOCA Limited Maximum Allowable Linear Heat Rate)

This figure reflects the three burnup windows due to the use of the TACO-2 fuel model and the NUREG-0630 cladding damage models. The values in the figure represent the generic B&W LOCA ECCS analysis results. The generic values bound the predicted Cycle 7 power peaking. The two-foot elevation generic limit for 0-1000 MWD/MTU has been increased by 0.5 KW/ft from the Cycle 6 value. This increase reflects the implementation of the FLECSET heat transfer coefficient improvement model. The limits preserve the LOCA Final Acceptance Criteria.

Page 5-4

Section 5.3.1.2 was changed to read "142 inches" to represent the approximate active height of all fuel batches. This is an editorial correction.

Section 5.3.1.3 was changed to reflect the initial enrichments of Cycle 7 fuel.

Section 5.3.1.6 was changed to reflect that the maximum enrichment allowed for a reload fuel assembly is being changed from 3.5 w/% to 4.3 w/%. The reactivity effects of the 4.3 w/% enriched fuel have been discussed in TSCR 180 which was submitted on January 12, 1988. This change is being made to allow the use of higher-enriched fuel to obtain longer cycles.

Startup Program - Physics Testing

The physics testing program is written in keeping with ANS-19.6.1, "Reload Startup Physics Tests for Pressurized Water Reactors", to limit testing to that necessary to determine if the operating characteristics of the reload core are consistent with the design predictions. This methodology provides assurance that the core can be operated as designed while minimizing unnecessary testing which could place the reactor in atypical configurations.

The Cycle 7 program has been changed slightly from the Cycle 6 program to better reflect the testing performed and to incorporate a revised methodology for the measurement of the full power Doppler coefficient. The previous technique, which required decreasing core power for a period of time, can be replaced by a method that correlates control rod worths to Doppler coefficient. Also, testing now includes the symmetry test to check for even core power distributions. This is a required test of ANS-19.6.1 which was performed in Cycle 6, although it was not described in the Reload Report.

CONCLUSION

Based on the discussions given above it is concluded that the design and Technical Specification limits developed for Cycle 7 support full power operation at the rated power level for a cycle length of 0 - 445 + 15 Effective Full Power Days without endangering the health and safety of the public.

IV. NO SIGNIFICANT HAZARDS CONSIDERATIONS

GPUN has determined that this Technical Specification Change Request poses no significant hazards as defined by NRC in 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The transient evaluation of Cycle 7 has determined that the Cycle 7 design is bounded by previously accepted analyses. Cycle 7 characteristics are conservative with respect to all applicable safety limits and criteria and the probability of occurrence of an accident previously evaluated is not affected. Comparison of the calculated Cycle 7 radionuclide source inventory based on 2568 MWt to the Cycle 6 inventory confirmed that there are no increases that would significantly increase doses. The radiological dose consequences of the accidents presented in FSAR Chapter 14 were conservatively re-evaluated for Cycle 7. All Cycle 7 accident doses are well within the limits established by 10 CFR 100. Therefore, Cycle 7 operation in accordance with the proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. Cycle 7 is designed as a standard B&W in-out-in low leakage core using approved vendor models. Review of Cycle 7 core loading design, fuel assembly thermal-hydraulic design, and analytical methods confirms that there are no safety concerns regarding the Cycle 7 core design. The transient evaluation of Cycle 7 has determined that the Cycle 7 design is bounded by previously accepted analyses. Cycle 7 fresh fuel assemblies are hydraulically and geometrically similar to previously irradiated fuel assemblies. Therefore, it is concluded that Cycle 7 operation, in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. Cycle 7 characteristics are conservative with respect to previous accident analyses, and all safety criteria as described in the Technical Specification bases are preserved by the revised limits. Therefore, it is concluded that Cycle 7 operation does not involve a significant reduction in a margin of safety.

The Commission has provided guidelines pertaining to the application of the three standards by listing specific examples in 48 FR 14870. The proposed amendment is considered to be in the same category as example (iii) of amendments that are considered not likely to involve significant hazards consideration in that the proposed change results from a nuclear reactor core reloading, no fuel assemblies significantly different from those found previously acceptable to the NRC for the previous core at TMI-1 are involved, and it has been adequately demonstrated that the acceptance criteria for the Technical Specifications have not been significantly changed, that the analytical methods utilized to demonstrate conformance with the Technical Specifications and regulations are not significantly changed, and that the NRC has previously found such methods acceptable. Thus, operation of the facility in accordance with the proposed amendment involves no significant hazards considerations.

V. IMPLEMENTATION

It is requested that the amendment authorizing this change become effective upon startup following the Cycle 7 refueling outage.

VI. AMENDMENT FEE (10 CFR 170.21)

Pursuant to the provisions of 10 CFR 170.21, attached is a check for $150.00.

ATTACHMENT

Technical Specification Changes