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. 244

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF DAUPHIN)

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 part of this request, proposed replacement pages for Appendix A are also included.

GPU NUCLEAR CORPORATION

BY: J G Broughton
Vice President and Director, TMI

Sworn and Subscribed to before me
this 20th day of May, 1994.

Debra S Klick
Notary Public

Notarial Seal
Debra S. Klick, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires July 4, 1998
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. 244 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. Darryl LeHew, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, PA 17057

Mr. Russell L. Sheaffer, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

Director, Bureau of Radiation Protection
PA. Department of Environmental Resources
Fifth Floor, Fulton Building
Third and Locust Streets
P. O. Box 2063
Harrisburg, PA 17120
Attn: Mr. Robert J. Barkanic

GPU NUCLEAR CORPORATION

BY: T G Broughton
Vice President and Director, TMI

DATE: 05/20/94

GPU Nuclear

GPU Nuclear Corporation
Route 441 South
P.O. Box 480
Middletown, Pennsylvania 17057-0480
(717) 944-7621
Writer's Direct Dial Number:

(717) 948-8005

May 20, 1994
C311-94-2069

Director, Bureau of Radiation Protection
PA Dept. of Environmental Resources
Fifth Floor, Fulton Building
Third and Locust Streets
P. O. Box 2063
Harrisburg, PA 17120
Attn: Mr. Robert J. Barkanic

Dear Mr. Barkanic:

Enclosed please find one copy of Technical Specification Change Request No. 244 to the Operating License for Three Mile Island Nuclear Station, Unit 1.

This document was filed with the U. S. Nuclear Regulatory Commission on the above date.

Sincerely,

T G Broughton

T. G. Broughton
Vice President and Director, TMI

plp

Enclosure

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Route 441 South
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May 20, 1994
C311-94-2069

Mr. Darryl LeHew, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, PA 17057

Dear Mr. LeHew:

Enclosed please find one copy of Technical Specification Change Request No. 244 to the Operating License for Three Mile Island Nuclear Station, Unit 1.

This request was filed with the U. S. Nuclear Regulatory Commission on the above date.

Sincerely,

T. G. Broughton

T. G. Broughton
Vice President and Director, TMI

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May 20, 1994
C311-94-2069

Mr. Russell L. Sheaffer, Chairman
Board of County Commissioners of
Dauphin County
P. O. Box 1295
Harrisburg, PA 17120

Dear Mr. Sheaffer:

Enclosed please find one copy of Technical Specification Change Request No. 244 to the Operating License for Three Mile Island Nuclear Station, Unit 1.

This request was filed with the U. S. Nuclear Regulatory Commission on the above date.

Sincerely,

T G Broughton

T. G. Broughton
Vice President and Director, TMI

plp

Enclosure

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 244

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

Revised Pages: 2-10, 4-48, and 4-49

These pages are attached to this change request.

II. REASON FOR CHANGE

This change is requested to provide a revised maximum allowable control rod insertion time acceptance criterion from the fully withdrawn position to 3/4 insertion at hot reactor coolant full flow conditions, specified in Technical Specification Section 4.7.1.1, for the remainder of Cycle 10. This revised criterion would only apply to the twelve (12) control rods which initially experienced slow rod drop times during testing conducted on March 17, 1994. These control rods are as follows: 1-1, 1-2, 1-3, 3-3, 3-4, 3-5, 3-6, 4-5, 5-4, 5-7, 5-9, and 6-5. The revised interim acceptance criterion for these control rods is 2.11 seconds which is based on the expected rod drop time assuming all four (4) ball check valves in the control rod drive mechanism thermal barrier are stuck closed. The following safety analyses have been reevaluated using a bounding control rod drop time of 3.0 seconds for all rods to justify the requested acceptance criterion of 2.11 seconds for the specified twelve (12) control rods.

The results of the 3-second drop time reevaluation of the startup accident require the high flux trip setpoint to be lowered to 63% Full Power (FP) for power levels at or below 47% FP to ensure that the peak Reactor Coolant System (RCS) pressure criterion is not exceeded. This change is incorporated in Table 2.3-1.

This change is also requested to clarify that biweekly exercising of control rods is at least 2 inches of travel which is consistent with actual surveillance requirements.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The proposed change revises the control rod trip insertion time testing criterion to specify that the control rods listed below will be considered operable if the maximum control rod trip insertion time from the fully withdrawn position to 3/4 insertion does not exceed 2.11 seconds at hot coolant full flow conditions. This specification does not change the optional hot no flow test and its 1.40 second acceptance value.

Control Rods (Group-Rod)

1-1	3-3	4-5	5-4	6-5
1-2	3-4		5-7	
1-3	3-5		5-9	
	3-6			

Control rod insertion time testing conducted on March 17, 1994 resulted in the identification of initial insertion times for these twelve (12) control rods which were in excess of the Technical Specification acceptance criterion for insertion time which is 1.66 seconds to 3/4 inserted. The maximum rod drop time experienced by a single rod during the initial testing was 2.9 seconds. The remaining 49 control rods had drop times of $\leq$1.5 seconds. All rods subsequently achieved drop times of $\leq$1.5 seconds.

Refueling outage 10R testing had required multiple drops of three (3) control rods (1-3, 3-6, and 4-5) in order to meet the acceptance criterion. These three (3) rods again experienced slow drop times during the March 17, 1994 testing. Accordingly, this technical specification change request addresses the twelve (12) control rods which have experienced slow drop times for Cycle 10 testing.

The revised criterion of 2.11 seconds is the calculated drop time based on dynamic hydraulic modeling of the control rod drive mechanism assuming all four thermal barrier ball check valves are stuck closed and nominal clearances exist in the lead screw-thermal barrier bushing area. Primary system chemistry changes implemented to maintain a higher reactor coolant system pH value, and surveillance movements with greater rod travel distance (bi-weekly exercising of control rods of at least 2 inches travel) during Cycle 10 operation are expected to reduce the likelihood of crud buildup in the gap between the lead screw and the thermal barrier bushing. During 10R, a Group 7 Control Rod Drive Mechanism (CRDM) from core location H-12 was disassembled and inspected for wear. The CRDM was clean with no abnormal deposits and the check valves were free. The Group 7 CRDMs exhibit little change in drop times in either the 10R insertion trip test or the subsequent March 1994 testing. The reliable performance of Group 7 may be due to the additional motion resulting from their regulating group function. This increased motion may reduce the crud formation as a result of the exchange of reactor coolant displaced by lead screw motion.

Additional Cycle 10 control rod drop time testing will be performed to verify the effectiveness of these corrective actions, as identified in GPU Nuclear letter C311-94-2055, dated April 22, 1994. However, since the twelve (12) specified rods have demonstrated susceptibility to crud buildup and sticking of the ball check valves, these control rods may exceed the existing drop time criteria of 1.66 seconds during future Cycle 10 testing.

The FSAR Chapter 14 analyses assume that a reactor trip results in the insertion of negative reactivity consistent with the 1% shutdown margin Technical Specification, including the most reactive control rod stuck in the fully withdrawn position. The rate of negative reactivity insertion is based on the combination of an assumed rod position vs. time curve and a reactivity worth vs. position curve, both of which are conservative for the core design and control rod design. The rod position vs. time curve includes the effect of the rod drop time. An increase in rod drop time requires evaluation of the existing safety analysis.

The proposed maximum allowable control rod insertion time of 2.11 seconds for the twelve (12) designated control rods was evaluated by adjusting the scram curve (negative reactivity insertion versus time) assumed in the affected FSAR accident analyses to reflect the bounding assumption that all control rods insert in 3.0 seconds rather than the current value of 1.66 seconds. This approach assumes no control rod movement for 1.34 seconds and therefore applies a 1.34 second delay to the existing FSAR scram curve which is more conservative than redefining the scram curve over a 3 second period, because no credit for negative reactivity insertion is taken over the first 1.34 seconds.

The accident analyses in Chapter 14 of the TMI-1 FSAR were reviewed to determine the consequences of the accidents with a 1.34 second time delay in scram time. The accident analyses which are limiting in terms of control rod insertion times have been reanalyzed using Babcock and Wilcox licensed codes and methods. A review of the remaining FSAR accident analyses determined that these analyses are either not affected by the revised control rod insertion times or remain conservatively bounding for Cycle 10 operation. The following FSAR events are discussed below and Table 1 provides a summary of the event reviews:

1. Uncompensated Operating Reactivity Changes
2. Startup Accident and Rod Withdrawal at Partial Power
3. Rod Withdrawal Accident at Rated Power Operation
4. Moderator Dilution Accident
5. Cold Water Accident
6. Loss-of-Coolant Flow
7. Stuck-Out, Stuck-In, or Dropped Control Rod Accident
8. Loss of Electric Load
9. Steam Line Break
10. Steam Generator Tube Rupture
11. Fuel Handling Accident, Waste Gas Tank Rupture, Fuel Cask Drop Accident
12. Rod Ejection Accident
13. Large Break Loss of Coolant Accident/Maximum Hypothetical Accident
14. Small Break LOCA
15. Loss of Feedwater and Feedwater Line Break Accident

1. <u>Uncompensated Operating Reactivity Changes</u>

   These reactivity changes are slow enough to allow the operator to detect and compensate for them. Additionally, if the changes are uncompensated by the operator then Reactor Coolant System (RCS) temperature changes compensate for the reactivity disturbance. Therefore, increased rod drop times would not affect these transients.

2. Startup Accident and Rod Withdrawal at Partial Power

The most limiting event with respect to RCS overpressure is the startup event. This event results in the largest mismatch between core power and steam generator heat removal. The startup event is defined as an uncontrolled withdrawal of control rods from a critical zero power condition. The consequences of the startup event are mitigated by the high flux (112% Full Power (FP)) and high pressure reactor trips. The specific trip that terminates the transient depends on the Reactivity Insertion Rate (RIR) of the withdrawn control rod(s).

The FSAR analyses included a number of parametric studies, including reactivity insertion rate and Moderator Temperature Coefficient (MTC). The most severe startup event reported in the FSAR considered an RIR of 2.15E-4 $\Delta k/k$/sec and an MTC of +0.9E-4 $\Delta k/k$/F. The peak pressurizer pressure for this analysis was 2653.4 psia.

The accident was re-analyzed using NRC-approved methods with the same FSAR assumptions, but with a 1.34 second trip delay time. Delaying the reactor trip to assume a 3.0 second rod drop time would allow power production to continue an additional 1.34 seconds. The additional core power would result in the heatup and expansion of the RCS fluid. The volumetric expansion would produce a greater insurge into the pressurizer. The resulting compression of the steam bubble would then increase system pressure.

Rod withdrawal from zero power (startup accident) bounds other initial power level rod withdrawals because more integrated full power seconds of energy are deposited in the coolant before the pressure trip and flux trip are reached simultaneously. In analyzing this accident, partial power rod withdrawal accidents were also considered. The results show that with a low overpower trip setpoint of 63% FP for power levels at or below 47% FP, the peak RCS pressure for this event does not exceed 2765 psia which is an acceptance criteria for this event. This assumed an MTC of +0.9 E-4 $\Delta k/k$/F which is conservative, as TMI-1 Cycle 10 has a negative MTC for the remainder of the cycle.

The adjustment of the high flux trip setpoint is incorporated in the attached revision to TMI-1 Technical Specification Table 2.3-1, and will be administratively controlled for the remainder of Cycle 10 operation. During the plant startup when the reactor power exceeds 47% FP, the high flux trip can be reset to 105.1% FP. During plant shutdown, the high flux trip setpoint change to 63% power shall be initiated within 6 hours of reaching a power level at or below 47% full power. This time period is permissible based on the low probability of a startup accident occurring during the period. Also, this period is acceptable because of the multiple failures and/or deliberate bypass of rod pull interlocks required to initiate a startup accident.

Table 2 summarizes the startup accident reanalysis.

3. <u>Rod Withdrawal (RW) Accident at Rated Power Operation</u>

The limiting RW at power analysis in the FSAR resulted in a peak pressure of 2479 psia, which is below the pressurizer safety valve lift setpoint of 2515 psia. In the analysis, delaying the reactor trip an additional 1.34 seconds would result in maintaining the thermal power at reactor trip for an additional 1.34 seconds. An evaluation was performed to determine if the delayed trip analysis would also result in peak pressure below the acceptance criterion.

The delayed reactor trip is expected to result in the lifting of the Pressurizer Safety Valves (PSVs). If the volumetric discharge from the PSVs is greater than the maximum volumetric insurge into the pressurizer, the peak pressurizer pressure cannot exceed 2515 psia, the PSV lift setpoint. The comparison of the insurge flow rate and the discharge flow rate from the original analysis shows that upon lifting, the PSVs can more than adequately relieve the maximum insurge seen in the RW event. The peak pressurizer pressure is therefore, at most, 2540 psia (including 1% PSV tolerance). The peak pressurizer pressure was converted to peak system pressure, which is located in the reactor vessel (RV) lower plenum, and was calculated to be 2651 psia, which is well below the acceptance criterion of 2765 psia.

The delay in reactor trip could also result in thermal power exceeding 112% FP, which is the design overpower. The Departure from Nucleate Boiling Ratio (DNBR) is at or above the correlation limit if the core thermal power is at or below 112% FP. The rise in core thermal power over the 1.34 second delay on reactor trip was calculated and added to the thermal power value of the FSAR analysis. This yields a peak thermal power at the new trip time of 110.52% FP. This power is below the design overpower value of 112% FP. The minimum DNBR, therefore, remains above the correlation limit with the additional reactor trip delay of 1.34 seconds for the RW at power event.

Table 3 summarizes the rod withdrawal at rated power accident reanalysis.

4. <u>Moderator Dilution Event</u>

The moderator dilution accident is an overheating event and would be affected by the delayed rod drop time. However, this event is bounded in terms of peak pressure by the startup event. Therefore, the increased rod drop time would result in acceptable consequences for this event.

5. <u>Cold Water Accident</u>

The results of this accident are acceptable without a Reactor Protection System (RPS) trip and an increase in rod drop time would not affect this transient.

6. Loss-of-Coolant Flow

The loss of coolant flow events are the most challenging for Minimum DNBR (MDNBR). The three most limiting DNB transients that are directly dependent on the time at which the Control Rod Assemblies (CRAs) enter the core include:

1.	One Pump Coastdown (4-3)	Condition II
2.	Four Pump Coastdown (4-0)	Condition II
3.	Locked Rotor (4-3)	Condition III

The coastdown events result in the most limiting DNB conditions of any of the other Condition I and Condition II events. The locked rotor event is the most limiting Condition III DNB event.

A reanalysis of the one pump coastdown and four pump coastdown transients using NRC-approved methods with the same FSAR assumptions but with greater than 1.34 second trip delay time (1.40) resulted in MDNBR's with the BWC correlation which were higher than the BWC correlation limit of 1.18. Therefore, the additional delay time results in acceptable response for these events.

Tables 4 and 5 summarize the reanalysis of these events.

The locked rotor event is a rapid flow reduction event that leads to a minimum DNBR within a few seconds. The event produces a flux/flow trip that is followed by a rapid flow decrease to approximately 75% of the initial value. This causes a trip to occur well within one loop transit time, therefore the reactor coolant inlet temperature does not respond and can be assumed to remain constant for the duration of the DNBR analysis (less than five seconds total time).

If the core has a positive moderator temperature coefficient, then the core power will increase as the coolant temperature increases. However, TMI-1 Cycle 10 has a negative moderator temperature coefficient for the remainder of the cycle. Therefore, reactor power level will be a constant value from initiation of the event until reactor trip.

A single state point analysis was used to determine the minimum DNBR behavior of the core using a minimum locked rotor transient flow fraction (approximately 75%) at the initial full power. The resulting MDNBR was found to be acceptable. Since this analysis uses initial power and final flow, the results are not affected by an increase in the trip delay time.

Table 6 summarizes the reanalysis of this event.

7. Stuck-Out, Stuck-In, or Dropped Control Rod Accident

No credit is taken for rod insertion during a stuck-out, stuck-in or dropped rod accident. Therefore, an increase in rod drop times would not affect the outcome of the accident.

The revised rod drop time criteria recognizes that the twelve (12) specified control rods may be susceptible to slower drop times. However, the additional control rod surveillance movements every two (2) weeks at an increased extent of movement are expected to reduce the likelihood of crud buildup between the lead screw and thermal barrier bushing. Also the lithium concentration in the RCS has been raised to increase pH and reduce the rate of corrosion. These compensatory measures have mitigated the adverse affects of crud buildup and therefore there is a high level of confidence that all control rods will trip into the core if required by the Reactor Protection System. Therefore, the probability of occurrence of a stuck-out control rod does not increase.

8. Loss of Electric Load

The event sequence is one that is terminated by high RCS pressure and does not take credit for the anticipatory reactor trip signal (ARTS) for reactor trip on turbine trip. Even with an additional delay of 1.34 seconds on reactor trip, the pressure transient is less severe than a total loss of feedwater event because main feedwater is still available and heat removal through the main steam safety valves, turbine bypass valves, and atmospheric dump valves exceeds the heat generation capacity of the core.

The MDNBR for a loss of AC power event is bounded by the total loss of flow event (4-0 RCPs).

9. Steam Line Break

The Main Steam Line Break accident is an overcooling transient that has the potential of resulting in a core return to power. For TMI-1, the limiting Steam Line Break (SLB) accident is analyzed from full power as this maximizes the steam generator inventory which causes the most overcooling. Following a SLB, the reactor would trip on low RCS pressure in approximately three seconds and feedwater would be isolated on low steam line pressure in less than one second. A delayed reactor trip provides additional heat input to the primary and would lessen the severity of the overcooling transient. However, additional heat input to the primary system is transferred to the secondary system which results in additional energy released to the containment during the SLB blowdown phase. This additional energy released to the containment could potentially increase containment pressure and temperature. However, the SLB blowdown data used to generate the containment equipment qualification profiles were conservatively developed and bound the slight increase in coolant energy resulting from the increased rod drop time.

The increased rod drop time will not affect fuel response since the neutron power decreases initially because the drop in RCS pressure causes the moderator density to decrease. As the colder water from the broken steam generator enters the core, the neutron power begins to increase. However, the reactor trips on low RCS pressure before the neutron power reaches the initial value. Even with an increase in the scram delay of 1.34 seconds, the neutron power would not exceed the design overpower value. Consequently, the core thermal power would remain near the initial value until the control rods insert. The condition of concern is an approach to minimum DNBR due to subcritical multiplication. This condition occurs well after reactor trip. The effect of increased drop times on this accident is therefore acceptable.

10. Steam Generator Tube Rupture (SGTR)

The SGTR is an overcooling event in which reactor trip occurs on low RCS pressure. Since dose consequences were deterministically calculated based on conservatively high and constant tube leak rate, the increased rod drop times would not increase offsite releases above those assumed in the FSAR.

11. Fuel Handling Accident, Waste Gas Tank Rupture, Fuel Cask Drop Accident

Rod insertion is not relevant during these events. Therefore, an increase in rod drop times would not affect the outcome of these accidents.

12. Rod Ejection Accident

The rod ejection event is a rapid ejection of a Control Rod Assembly from the core region. The resulting power excursion, due to the rapid increase in reactivity, is limited by the Doppler effect and terminated by the RPS high flux or high pressure trip. The rod ejection event is limiting with respect to peak fuel enthalpy. Also, the number of pins predicted to experience DNB is important because it affects the calculated offsite doses. The effects of the delayed reactor trip on fuel enthalpy and number of pins in DNB is calculated.

The integrated neutron power response during the 1.34 second scram delay is used to calculate the peak fuel enthalpy. Based on adiabatic heatup, a peak fuel enthalpy of 193 cal/gm is calculated which is below the event threshold of 200 cal/gm.

An evaluation of pins in DNB showed that the point kinetics/adiabatic heatup method used in the FSAR was very conservative compared to results calculated using three-dimensional neutron kinetics and core thermal-hydraulics methods (BWKIN). The FSAR method overpredicts the number of fuel pins in DNB by a factor of two. With the conservative adiabatic heatup calculation, the number of pins in DNB is calculated to be 10% higher due to the delayed reactor trip assuming a bounding 3.0 second drop time.

This would result in a 10% higher offsite dose, which would remain well below 10 CFR 100 dose limits and which would be bounded by the Maximum Hypothetical Accident dose consequences.

Table 7 summarizes the reanalysis of this accident.

13. Large Break Loss of Coolant Accident/Maximum Hypothetical Accident

No credit is taken for rod insertion during these events and results are independent of reactivity addition rates. Therefore, an increase in rod drop times would not affect the outcome of the accident.

14. Small Break LOCA

The analysis of small break LOCAs show that there are no clad temperature excursions because the core remains covered at all times during the transient. Since the core remains covered, the consequences of a small break LOCA are bounded by large breaks. The additional 1.34 second delay on reactor trip would not add sufficient energy to result in core uncovery for any small breaks. Since the core would remain covered, no clad temperature excursion would occur. Therefore, the results of the small break LOCA analyses would not be affected by the additional trip delay.

The additional delay in reactor trip could result in a change in the break size that defines the transition from small break to large break analysis methodology. However, since these sizes are not limiting for peak clad temperature (bounded by large break LOCAs), the assumed rod drop delay of 3.0 seconds to 3/4 insertion has no effect.

15. Loss of Feedwater and Feedwater Line Break Accident

The loss of main feedwater accident results in a mismatch between core power and secondary heat removal. The mismatch causes the RCS to heatup and pressurize. In order to determine the effect of the reactor trip delay, the RCS pressurization rate was determined from existing analyses, and multiplied by the trip delay of 1.34 seconds to yield the increase in peak RCS pressure. Adjusting to the maximum pressure location in the RCS resulted in a peak RCS pressure below the acceptance criterion limit of 2765 psia. Therefore, the consequences of a loss of main feedwater with an assumed 3.0 second drop time to 3/4 insertion on reactor trip are acceptable.

Table 8 summarizes the reanalysis of this accident.

A rupture of the main feedwater line to one steam generator results in a rapid reduction in feedwater flow. The decrease in secondary heat removal causes the RCS to pressurize and eventually a reactor trip on high RCS pressure is actuated. The feedwater line break is a limiting fault transient. The acceptance criterion for peak RCS pressure is that the pressure shall remain below ASME service level

C limits (3125 psia). The feedwater line break inside containment was not part of the original licensing basis for TMI-1. However, the feedwater line break analysis from the Midland nuclear units was reviewed as well as a TMI-1 plant specific feedwater line break analysis that had been performed previously.

The peak RCS pressure reached approximately 2700 psia in the Midland analysis and approximately 2625 psia for TMI-1. The Midland analysis included a high pressure trip setpoint indicative of degraded environment instrument errors (2443 psia). The TMI-1 peak RCS pressure would not exceed level C limits even with an additional reactor trip delay of 1.34 seconds. Therefore, the feedwater line break transient results are acceptable with an assumed rod drop delay of 3.0 seconds to 3/4 insertion.

Summary of Safety Evaluation

The results of the above bounding evaluation supports the conclusion that, during Cycle 10, a control rod drop time acceptance criterion of 2.11 seconds to 3/4 insertion is acceptable at hot coolant full flow conditions for the twelve (12) specified control rods, since there is no adverse effect on existing event acceptance limits and consequences of postulated accidents. Inserting all rods at rates faster than 3.0 seconds ensures that local power at any location in the core will be less than that generated in the 3 second analyses due to the greater insertion of negative reactivity at every control rod location in the core.

Technical Specification Table 2.3.1 was revised at item 1 for Nuclear power, max % of rated power, to include footnote (6), which requires during plant startup from 0% $\leq$ 47% power, that the setpoint shall be lowered to 63% Full Power. During plant shutdown, the high flux trip setpoint change to 63% power shall be initiated within 6 hours of reaching a power level at or below 47% full power. This time period is permissible based on the low probability of a startup accident occurring during the period. Also, this period is acceptable because of the multiple failures and/or deliberate bypass of rod pull interlocks required to initiate a startup accident.

Technical Specification Section 4.7.1 Bases is revised to clarify that the specified trip time of 2.11 seconds for the specified twelve (12) control rods for Cycle 10 is based on reanalysis of limiting safety analyses. Technical Specification Page 4-49 is editorially revised only to accommodate revisions to Page 4-48, and to clarify that bi-weekly exercising of control rods is "at least" two inches of travel to be consistent with actual surveillance requirements.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION

GPU Nuclear has determined that this Technical Specification Change Request involves no significant hazards consideration as defined by NRC in 10 CFR 50.92 because:

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The revised acceptance criterion assures the ability of the control rods to mitigate design basis accidents. Specifically, the revised acceptance criterion assures that the negative reactivity insertion rate maintains the event acceptance criteria of the safety analysis. Therefore, this change does not increase the probability of occurrence or the 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 since the revised acceptance criteria will not create any failure modes not bounded by previously evaluated accidents.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The revised acceptance criteria for TMI-1 Cycle 10 assures that the negative reactivity insertion rate is sufficient to maintain the margin of safety in the accident analysis. The margin of safety is defined as the margin between the safety limit and fission product barrier failure. Since none of the accident analyses exceed the event safety limit, the margin of safety is not reduced.

V. IMPLEMENTATION

It is requested that the amendment authorizing this change become effective upon issuance. As requested above, the TSCR should be processed on an exigent basis to preclude the possible need for future Enforcement Discretion should control rod drop times for any of the specified control rods be greater than 1.66 seconds but less than 2.11 seconds.