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. 242

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.S. Broughton
Vice President and Director, TMI

Sworn and Subscribed to before me
this 22nd day of March, 1994.

Melody Kim Kulp
Notary Public

Notarial Seal
Melody Kim Kulp, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Aug. 21, 1997
Member, Pennsylvania Association of Notaries

9409260157 940322
PDR ADOCK 05000289
P PDR

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 242

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 revised maximum allowable control rod insertion time acceptance criteria for an operable control rod drive mechanism from the fully withdrawn position to 3/4 insertion at hot reactor coolant full flow conditions, specified in Technical Specification Section 4.7.1.1, for the remainder of Cycle 10.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The proposed change revises the maximum control rod insertion time to 3.0 seconds for an operable control rod drive mechanism from the fully withdrawn position to 3/4 insertion for safety and regulating rod groups, for Cycle 10 operation, unless it is demonstrated that all accident analyses meet their event safety limits.

Recent testing of control rod insertion times has resulted in the identification of initial insertion times for 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. However, the remaining control rods had drop times of $\leq$1.5 seconds. All rods have subsequently achieved drop times of $\leq$1.5 seconds, except for Rod 1-3.

Refueling outage 10R testing had in some cases required multiple drops of three (3) control rods in order to meet the acceptance criterion. The exercising of the rods typically results in improvement in the insertion time with each successive drop, and after several tests the acceptance criterion was met. Control rod insertion testing is required during each refueling outage prior to startup.

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 3.0 seconds was evaluated by adjusting the scram curve (negative reactivity insertion versus time) assumed in the affected FSAR accident analyses to reflect the assumption that all control rods insert in 3.0 seconds rather than the current value of 1.66 seconds. This approach applies a 1.34 second delay to the existing FSAR scram curve and is more conservative than redefining the scram curve over a 3 second period.

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. A review of the remaining FSAR accident analyses determined that these analyses either are not affected by the revised control rod insertion times or remain conservatively bounding for Cycle 10 operation. The following FSAR events are discussed below:

1. Uncompensated Operating Reactivity Changes
2. Startup Accident
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 Power
9. Steam Line Break
10. Steam Generator Tube Rupture
11. Anticipated Transients Without Scram
12. Fuel Handling Accident, Waste Gas Tank Rupture, Fuel Cask Drop Accident
13. Rod Ejection Accident
14. Large Break Loss of Coolant Accident/Maximum Hypothetical Accident
15 Small Break LOCA
16 Loss of Feedwater 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. <u>Startup Accident</u>

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) 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. The most severe startup event reported in the FSAR considered an RIR of 2.15E-4 Δk/k/sec and an MTC of +0.9E-5 Δk/k/F. The peak pressurizer pressure for this analysis was 2653.4 psia. Delaying the reactor trip to justify a 3.0-second rod drop test 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.

The accident was re-analyzed using NRC-approved methods with the same FSAR assumptions, but with a 1.34 second trip delay time. The results show that with a low overpower trip setpoint of 50% FP for power levels at or below 25% 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-5 Δk/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.

When the reactor power reaches 25% FP, the high flux trip can be reset to 105.1% FP. The FSAR analyses of rod withdrawals at power included a case from 25% FP with a high flux trip at 112% FP. The resulting peak pressurizer pressure was well below the 2765 psia acceptance criterion.

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 then the maximum volumetric insurge into the pressurizer, the peak PZR pressure cannot exceed 2515 psia, the PSV lift setpoint. The comparison of the insurge flow rate and the discharge flow rate 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 2652 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.

4. Moderator Dilution Event

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. Cold Water Accident

The results of this accident are acceptable without an 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. The three most DNB-limiting transients that are directly dependent on the time at which the 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 DNB-limiting Condition III event.

A reanalysis of the one pump coastdown and four pump coastdown transients using NRC-approved methods with the same FSAR assumption but with a 1.34 second trip delay time 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.

The locked rotor event is a rapid flow reduction event that leads to a minimum DNBR within a couple of seconds. The present FSAR analysis shows the minimum DNBR goes below the BAW-2 CHF correlation limit of 1.3, but does not go below a minimum DNBR of 1.0 (the acceptance criteria).

The event produces a flux/flow trip that is followed by the control rods entering the core. 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, core power will not increase prior to the control rod insertion.

An increase in the control rod drop times of 1.34 seconds further delays the time when core power and local power are reduced. 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.

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.

8. Loss of Electric Power

The pressure transient for this event is bounded by the loss of feedwater event. The MDNBR for this 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. The increased rod drop time will not affect fuel response because the condition of concern is an approach to minimum DNBR due to subcritical multiplication. This condition occurs well after reactor trip.

The increased rod drop times would result in slightly more energy addition to the reactor coolant. Blowdown data used to generate equipment qualification profiles were conservatively developed and bound the slight increase in coolant energy resulting from the increased rod drop time. 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 be expected to increase offsite releases above those assumed in the FSAR.

11. Anticipated Transient Without Scram (ATWS)

No credit was taken for control rod insertion during an ATWS. Therefore, an increase in rod drop times would not affect the outcome of the accident.

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

No credit is taken for rod insertion during these events. Therefore, an increase in rod drop times would not affect the outcome of these accidents.

13. 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 and results in a value 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 hydraulic methods. The FSAR method over predicts the number of fuel pins in DNB by a factor of two. This means that the small increase in the number of pins in DNB resulting from a 1.34 second scram delay is well within the conservatisms inherent in the FSAR. Consequently, the FSAR calculation of offsite consequences for the control rod ejection event remains bounding with an additional scram delay of 1.34 seconds.

14. 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.

15. 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 uncovering 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 rod drop delay of 3.0 seconds to 3/4 insertion will have no effect.

16. Loss of Feedwater 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 a 3.0 second drop time to 3/4 insertion on reactor trip are acceptable.

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). TMI-1 was not licensed to analyze a feedwater line break. However, the feedwater line break analysis from the Midland nuclear units was reviewed.

The peak RCS pressure reached approximately 2700 psia in the Midland analysis and approximately 2625 psia for TMI-1. This analysis included a high pressure trip setpoint indicative of degraded environment instrument errors (2443 psia). Although TMI-1 operates at 2568 MWt instead of 2452 MWt (Midland rated power) the 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 a rod drop delay of 3.0 seconds to 3/4 insertion.

The results of the above evaluation support the conclusion that, during Cycle 10, a control rod drop time acceptance criterion of 3.0 seconds to 3/4 insertion is acceptable, since there is no adverse effect on existing event acceptance limits and consequences of postulated accidents.

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% $\leq$ 25% power, that the setpoint shall be lowered to 50% Full Power. During plant shutdown, the high flux trip setpoint change to 50% power shall be initiated within 4 hours of reaching a power level at or below 25% full power.

Technical Specification Section 4.7.1.1 is also revised to note that due to uncertainty in the degradation mechanism over time, additional control rod surveillance testing will be performed at a frequency to be agreed upon between NRC and the licensee.

Technical Specification Section 4.7.1 Bases is revised to clarify that the specified trip time of 3.0 seconds 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.

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. In order to allow TMI-1 startup to proceed within the current Technical Specification requirements considering the restrictions for an inoperable control rod, the request for a notice of enforcement discretion regarding Technical Specification Section 4.7.1.1 has been provided as outlined in the cover letter to this amendment change request, to be in effect until issuance of the proposed amendment. As requested above, the TSCR should be processed on an exigent basis.