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. 279

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

GPU NUCLEAR, INC.

BY: James W Langenbach
Vice President and Director, TMI

Sworn and Subscribed to before me
this 3rd day of December, 1998

Linda L. Ritter
Notary Public

Notarial Seal
Linda L. Ritter, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Feb. 26, 2002
Member, Pennsylvania Association of Notaries

ENCLOSURE 1

TMI-1 Technical Specification Change Request No. 279 Safety Evaluation

No Significant Hazards Consideration and

Proposed Technical Specification Revised Pages

I. Technical Specification Change Request No. 279

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

- Revised Technical Specification Pages: 2-4a and 2-4c

These pages are attached to this enclosure.

II. Reason for Change

The purpose of this Technical Specification Change Request (TSCR) is to revise the TMI-1 Core Protection Safety Limits and Core Protection Safety Bases, as specified on Technical Specification Figures 2.1-1 and 2.1-3, to provide more restrictive limits which reflect the decrease in reactor coolant system flow resulting from the analysis of increased once-through steam generator (OTSG) tube plugging limits.

The TMI-1 OTSGs have 15,531 tubes each. Currently, TMI-1 OTSGs have 5.5% (total of 1695) of the tubes removed from service. This consists of 1300 tubes plugged in "A" OTSG (approximately 8%) and 395 tubes plugged in the "B" OTSG (approximately 2.5%). The current maximum allowable plugging limit is a total of 2000 tubes (6.4%) as documented by NRC memorandum, H.L. Thompson (NRC) to R. Starostecki (NRC), "Operation of TMI-1 with 2000 Plugged Steam Generator Tubes", June 10, 1985. However, increased tube plugging has been seen at many of the recent outages at the B&W plants. It is expected that this trend will continue as the B&W plants continue to age. The B&W Owners Group is presently investigating the causes of, and mitigation methods for this type of corrosion. GPU Nuclear has performed an analysis to justify an average 20% of the TMI-1 steam generator tubes removed from service in anticipation of the need to plug more than the currently analyzed limit. This analysis results in more restrictive core protection safety limits and bases as provided in the proposed revisions to TMI-1 Technical Specification Figures 2.1-1 and 2.1-3. This proposed change to the tube plugging limits would allow a maximum tube plugging of 25% in any one OTSG and a maximum plugging asymmetry of 15% between the two OTSGs. These tube plugging limits will include actual plugged tubes and the equivalent plugged tubes resulting from other repairs such as sleeving.

III. Safety Evaluation Justifying Change

TMI-1 Technical Specification Section 2.1.1 specifies the combination of reactor system pressure and coolant temperature which shall not be exceeded as established in Figure 2.1-1. The curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR or greater is predicted for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in TMI-1 Technical Specification Figure 2.1-3. Technical Specification Figure 2.3-1 represents the

reactor protection system (RPS) maximum allowable setpoints, which are formed by the low pressure, high pressure, and high temperature trip setpoints. These setpoints are currently more restrictive than the setpoints based on the limits established by Figure 2.1-1. For 20% steam generator tube plugging (SGTP), the proposed revised limits of Figures 2.1-1 and 2.1-3 are more restrictive. However, they remain bounded by the current RPS trip setpoints of Figure 2.3-1, as described below.

OTSG tube plugging decreases reactor coolant system (RCS) flow (due to increased flow resistance), reduces RCS inventory and decreases primary-to-secondary heat transfer. The effects of 20% average SGTP limits on FSAR Chapter 14 accident analyses and on setpoint determination have been evaluated as described below. In addition, changes in operating parameters resulting from the increased tube plugging limit have been evaluated for operational considerations and component structural integrity.

A. FSAR Chapter 14 Transient and Accident Analysis

The TMI-1 Updated Final Safety Analysis Report (UFSAR) Chapter 14 accidents were reviewed assuming a 20% average steam generator tube plugging, to determine the effect on acceptance criteria, and thus margin of safety. The 20% SGTP is predicted to result in a small change in operating parameters as shown on Table 1. For 20% SGTP, the RCS volume decreases by about 427 ft^3, or 3.7%, and the steam generator heat transfer area is reduced by about 15.5% from the present plugging conditions. The TMI-1 UFSAR will be updated to reflect the reanalysis of limiting transients such as Loss-of-Coolant Accidents and Loss of Main Feedwater. To determine the safety consequences of the increased tube plugging required re-analysis of the limiting accidents and re-evaluation of others.

The following UFSAR events are discussed below:

1. Uncompensated Operating Reactivity Changes
2. Startup Accident
3. Rod Withdrawal at Rated Power
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. Main Steam Line Break
10. Steam Generator Tube Rupture
11. Fuel Handling Accidents
12. Rod Ejection Accident
13. Large and Small Break Loss of Coolant Accidents
14. Maximum Hypothetical Accident
15. Waste Gas Tank Rupture
16. Loss of Main Feedwater
17. Fuel Cask Drop Accident
18. Anticipated Transient Without Scram

Uncompensated Operating Reactivity Changes

During normal operation, the core reactivity changes due to fuel burnup and changes in the fission product poison concentrations. This results in an increase or decrease in the RCS average temperature. An imbalance between heat added to and heat removed from the reactor coolant causes the reactor coolant to either heatup or cooldown. Given a heat rate imbalance, the reactor coolant will heatup and cooldown slightly faster with tubes plugged than without tubes plugged because of the reduced RCS inventory. Reduced heat transfer in the steam generators will increase the RCS heatup rate and decrease the RCS cooldown rate. These reactivity changes are slow enough to allow the operator to detect and compensate for them. Additionally, if the changes are uncompensated for by the operator, then the Integrated Control System is designed to maintain T_{ave} at its setpoint. Therefore, a reduction in the initial RCS inventory and steam generator heat transfer capacity due to tube plugging will have an insignificant impact on this event.

Startup Accident

The startup event is defined as an uncontrolled withdrawal of control rods from a critical zero power condition, resulting in an increase in thermal power and a pressurization of the primary system. This event results in the largest mismatch between core power and steam generator heat removal, and is the most limiting event with respect to RCS overpressure. The consequences of the startup event are mitigated by the high flux 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). For slow RIRs, the reactor trips on high RCS pressure. For more rapid RIRs, the reactor trips on high flux before the high RCS pressure trip setpoint is reached. Rod withdrawal from zero power bounds other initial power level rod withdrawals because more integrated power seconds of energy are deposited in the coolant before the pressure trip and flux trip are reached simultaneously.

The analysis of this transient conservatively assumes that all the heat produced in the core remains in the primary system, i.e. no steam generator heat transfer. The 20% OTSG tube plugging will result in a small decrease in the RCS volume, which will cause the system to heat up at a faster rate. The startup accident was re-analyzed with 20% OTSG tube plugging and bounding analysis assumptions with 102% RCS flow (GPU Nuclear Calculation C-1101-224-E610-068). A spectrum of cases were analyzed, varying the RIR to determine the RIR at which the high pressure trip and high flux trip setpoints were reached nearly simultaneously. This RIR was the most limiting with regard to peak RCS pressure.

The results showed that an RIR of 2.13E-4 ($\Delta k/k$)/sec resulted in the highest RCS pressure. In this re-analysis, the pressurizer safety valves (PSV) lift to relieve system pressure which overshoots the PSV lift pressure as the PSVs open for the first time. Thereafter, any increase in the RCS pressure is limited to the PSV lift setpoint.

Even with a conservative 3% lift tolerance on the PSV setpoint, the peak RCS pressure, which is in the RV lower plenum, reached 2708 psia. The peak thermal power was 57.8% . The acceptance criteria for this event are that the reactor thermal power be limited to 112% rated power and RCS pressure be limited to 2765 psia. Based on these results, it is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach the design overpower condition, and the peak pressure stays below the design limit of 2765 psia.

Rod Withdrawal at Rated Power

A rod withdrawal accident (RWA) is the accidental withdrawal of a control rod group while the reactor is at rated power. This uncontrolled withdrawal, through operator error or equipment failure, results in positive reactivity addition. As positive reactivity increases, the power level increases, the reactor coolant and fuel rod temperatures increase and the consequences are mitigated by high flux or high pressure reactor trips. The UFSAR results show that the thermal power from this event is limited to 109%, which is well below the maximum design power of 112%. The system pressure during this event increases by only 118 psi providing considerable margin to the event acceptance criteria of 2765 psia. The average core moderator temperature change shows an increase of only 3° F. Since the transient is terminated within 15 seconds of its initiation on a high neutron flux trip, a reduction in steam generator heat transfer capacity will have an insignificant effect on the peak pressure. The pressure response is bounded by the startup accident. Thermal power will not be affected since the doppler and moderator feedback will remain approximately the same. Therefore, the acceptance criteria will be met with considerable margin for 20% tube plugging per steam generator.

Moderator Dilution Accident

The moderator dilution accident is a relatively slow over-pressurization transient in which positive reactivity is inserted by boron dilution. The small change in RCS volume and reduced heat transfer resulting from plugging 20% of the tubes per OTSG will slightly increase the reactivity insertion rate as well as the heat-up rate. The analyses in the UFSAR demonstrate large margins to each of the acceptance criteria, with a peak pressure of 2435 psia, a peak thermal power of 107.3% of rated and an average reactor coolant temperature change of about 2° F. In the UFSAR analysis, an initial condition was established such that the secondary heat removal rate exactly matched the RCS heat addition rate (i.e., reactor power). The secondary heat removal rate was maintained constant throughout the transient. This is a conservative assumption and is unaffected by 20% SGTP. The results will remain well within the acceptance criteria for 20% tube plugging per OTSG. This is a slowly developing transient, and the highest rate of dilution can be handled by the automatic control system, which inserts rods to maintain the power level and the reactor coolant system temperature. The UFSAR analysis conservatively did not credit any control system

action.

The moderator dilution accident in the refueling mode is not affected by OTSG tube plugging.

Cold Water Accident

The analysis of this accident in the UFSAR assumed that the plant was operating at 50% power with one pump in each loop, when the remaining two pumps were started. The resulting decrease in the core average temperature, combined with a negative moderator temperature coefficient results in positive reactivity feedback. The maximum power and pressure are reached in less than 15 seconds after the pumps are started, with large margins to the acceptance criteria. Due to the short duration of the transient, the reduced RCS volume and heat transfer area resulting from 20% OTSG tube plugging will have an insignificant effect on the results.

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 lower flow rate resulting from tube plugging include:

1) One Pump Coastdown (4→3)
2) Four Pump Coastdown (4→0)
3) Locked Rotor (4→3)

The RCS flows used in the Loss of Flow events in the UFSAR are conservatively based on 106.5% of design flow. The average measured RCS flow rates are 110 % of design flow (GPU Nuclear Calculation C-1101-220-5360-037). In order to re-analyze these events for 20% tube plugging, analyses were performed to determine RCS loop flow rates and pump coastdown flow rates. The RCS flow rate was calculated using the FSPLIT thermal hydraulics computer code for the generation and solution of steady state flow networks (FTI Proprietary Document 32-1234876-02). FSPLIT solves for momentum, elevation, friction, and form loss pressure drops. Analyses were performed for various plugging and RC pump operating conditions. To conservatively bound future fuel assembly design effects on RCS flow rates, the analysis was performed for a Mk-B10 core with fuel debris filter plates. The analysis results show that for 20% average tube plugging, the total flow is 106.4% of design flow, or a reduction of 3.6% flow.

Investigation of various asymmetric plugging combinations showed that a slightly lower RCS flow of 106.28% would result from a 15%/25% asymmetric plugging. Asymmetric tube plugging impact will be conservatively accounted for by using an allowance of 0.5%. The flow measurement uncertainty was determined to be 1.8% per the above referenced GPU Nuclear calculation. The lowest 4 pump RCS flow with 20% tube plugging is calculated to be 106.4% –1.8% –0.5% = 104.1% of design

flow. For additional conservatism in DNB calculations, a minimum flow of 102% of design flow was used.

Similarly, for 3-pump operation, the total flow is 80.2% of design flow, which is a flow reduction of 2.5% for 20% tube plugging. The 3-pump 2.5% vs. 4-pump 3.6% change in flow is primarily due to the slope of the head capacity curve, where a percent of added head in the 3-pump range flow (per pump) reduces flow less than a percent head in the 4-pump flow range (per pump). Applying a 1.8% flow measurement uncertainty and 0.5% for asymmetric tube plugging, the minimum 3-pump flow would be 80.2% – 1.8% – 0.5% = 77.9%. For additional conservatism, a minimum flow of 74.5% of design flow was used.

The effects of steam generator tube plugging on pump coastdown flow rates were evaluated (FTI Proprietary Document 32-1269014-00). The RELAP computer code was used with plugging fractions of 0, 10, 20, and 30%, and with asymmetric plugging of 0% in one SG and 30% in the other. The effects of tube plugging on the normalized coastdown flow rates were determined to be small, and consequently, the most limiting normalized flow rate among all those analyzed was conservatively selected.

A reanalysis of the one pump coastdown and four pump coastdown transients with 20% tube plugging initial flow and flow coastdown characteristics resulted in MDNBR's with the BWC correlation which were conservatively higher than the BWC correlation limit of 1.18. For the 4-0 loss of flow coastdown event, the MDNBR was 1.669 and for the 4-3 loss of flow event the MDNBR was 1.484 (GPU Nuclear Calculation C-1101-202-E620-365).

The locked rotor event is a rapid flow reduction event that leads to a minimum DNBR within a couple of seconds. The event triggers 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 Technical Specifications require that the moderator temperature coefficient shall not be positive at power levels above 95% of rated power. Therefore, the 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 74.5%) at the initial full power. The resulting MDNBR was 1.276 which provides considerable margin to the MDNBR limit of 1.0 (limiting fault)(GPU Nuclear Calculation C-1101-202-E620-365). Based on this re-analysis, it is concluded that the results of the Loss of Coolant Flow events are acceptable with 20% SGTP and 15%/25% asymmetric plugging.

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

The worst case rod misalignment transient results from a dropped rod. The UFSAR analysis of this event shows that following a dropped control rod, the neutron power decreases, causing a rapid decrease in both the core moderator temperature and fuel temperature. This temperature decrease overcompensates for the worth of the control rod, and the power rises until the reduced steam generator demand begins to increase the inlet temperature and decrease the power. A reduction in steam generator heat transfer capacity will have a slight effect on the pressure response, while a reduction in RCS inventory will have a slight effect on both the pressure response and thermal power response. However, the peak pressure increase for this transient is only 100 psi and thermal power never gets higher than the initial value. Since 20% SGTP has a small effect on the results of this accident, significant margins to the acceptance criteria will be maintained.

Loss of Electric Power

A loss of electric power results in gravity insertion of the control rods and trip of the turbine valves. The reactor coolant pumps, main feedwater and condensate booster pumps will also trip. Emergency Feedwater (EFW) is provided by the Turbine Driven Pump only, and will control steam generator levels at 50% of the operating range. Excessive temperatures and pressures in the RCS are prevented by natural circulation, with excess steam relief through the main steam line safety valves and the atmospheric dump valves (turbine bypass valve steam relief is lost due to loss of power to the condenser cooling water circulating pumps). The ability to transition to natural circulation with an average 20% SG tube plugging or a 25%/15% asymmetric plugging was analyzed. The results showed that adequate heat removal is available and RCS pressure is maintained well below the code limits (GPU Nuclear Calculation C-1101-202-E610-069).

Main Steam Line Break

The UFSAR analysis of this event is the double-ended rupture of a 24 inch main steam line from rated power. For a main steam line break (MSLB), a larger initial secondary system inventory in the steam generator associated with the break will lead to a higher integrated heat removal. The larger the heat removal, the lower the resultant reactor coolant temperature. For the same heat rate imbalance, the reactor coolant will cool down slightly faster with tubes plugged than without tubes plugged because of the reduced RCS inventory. However, this phenomenon will be more than offset by the 15.5% decrease in steam generator heat transfer area resulting in a slightly slower cooldown rate. Since the UFSAR analysis assumed a conservatively large initial steam generator inventory of 55,000 lbm per steam generator, and since this bounds the steam generator mass inventory with 20% tube plugging (conservatively estimated to be less than 45,000 lbm), the RCS cooldown remains bounded by the UFSAR analysis. The UFSAR analysis also calculates dose consequences resulting from MSLB induced OTSG tube leakage. Because of the bounding steam generator secondary mass

assumed in the UFSAR analysis and the reduced primary-to-secondary heat transfer as a result of 20% SGTP, the OTSG tube axial tensile loads, and resulting tube leakage, would be less than the existing UFSAR values. Thus, the acceptance criterion that the core will remain intact for effective cooling will be maintained, and the dose consequences reported in the UFSAR are bounding for this event and are below 10 CFR 100 limits.

Steam Generator Tube Rupture

The UFSAR acceptance criterion is that the offsite dose be less than the limits specified in 10 CFR 100. This transient assumes a double-ended rupture of one OTSG tube with an initial leak rate of 435 gpm and offsite power available, which allows for the use of the Turbine Bypass System to cool the unit. The tube leak rate was calculated assuming critical flow based on the upstream and downstream conditions. Following break initiation, the system will depressurize as primary inventory is lost out of the tube break. Reactor trip will occur on low RCS pressure and the subsequent trip will cause the main steam safety valves (MSSVs) to lift. The RCS will depressurize to the High Pressure Injection (HPI) setpoint, allowing for inventory losses to be balanced. The Turbine Bypass Valves (TBVs) are used to cool the unit down to 250°F with the intact steam generator. The reduced RCS inventory will not have any impact on this event. The reduced primary-to-secondary heat transfer post trip should result in less release through the MSSVs since the secondary side will not heat up as fast, allowing the TBVs more time to relieve pressure and redirect primary system activity to the condenser. Therefore, it can be expected that primary activity released through the MSSVs will be less for higher tube plugging. ~~Following the reseat of the MSSV~~s, the system is cooled down to 250°F at a maximum rate of 100°F/hr using the TBVs. A slightly longer cooldown time may be required with a maximum of 25% SGTP in the unaffected OTSG as a result of the decreased steam generator heat transfer area. However, the cooldown time is expected to remain less than 10 hours, which was assumed in the UFSAR analysis. Therefore, the offsite dose consequences with an average 20% SGTP are bounded by the UFSAR analysis.

Fuel Handling Accidents

These accidents only occur during fuel assembly shuffling and transfer operations between the reactor, the spent fuel pool, and the fuel shipping casks. Therefore, OTSG tube plugging will have no impact on these accidents.

Rod Ejection Accident

The Rod Ejection accident 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 the UFSAR

assumes that any fuel rod that experiences DNB is considered failed and contributes to the offsite dose consequences.

The acceptance criteria listed in the UFSAR are that the rod ejection will not further damage the RCS (met if peak fuel enthalpy limit is not exceeded) and that the offsite dose be within the 10CFR100 limits. Since DNB occurs within the first 2 seconds, reduced primary-to-secondary heat transfer will not contribute to the amount of fuel in DNB. The small break LOCA aspect of this transient is bounded by the analysis described below.

Thus, it is concluded that an average 20% SGTP will have no effect on the results of the Rod Ejection Accident, and the amount of fuel damage will remain unchanged.

Large and Small Break Loss of Coolant Accidents

The large and small break LOCA were re-analyzed with OTSG average tube plugging levels of up to 20 percent (15 percent in the intact leg, 25 percent in the broken leg) and a minimum RCS flow of 102 percent of design flow. The calculations were performed by Framatome Technologies Incorporated (FTI) using the NRC approved RELAP5-based EM, described in BAW-10192PA. FTI prepared RELAP5/MOD2-B&W, REFLOD3B, and BEACH input models that represent the TMI-1 plant. The analyses were performed in compliance with the EM methods and the limitations and restrictions stated in the NRC Safety Evaluation Report (SER) on BAW-10192PA, at a conservative power level of 102% of 2772 MWt.

TMI-1 specific sensitivity studies were performed to confirm that the most limiting set of plant boundary conditions were applied to the licensing analyses for both large and small LOCAs. After completion of these studies, the required spectrum of ECCS analyses of the limiting break sizes were performed at a variety of core operating parameters that envelope the entire range of core operation.

Large break sensitivity studies and break spectrum studies performed with the RELAP5-based EM for the 177-FA lowered loop show that the double-ended guillotine break at the cold leg pump discharge (CLPD) with a discharge coefficient of 1.0 and minimum ECCS injection is the limiting case (FTI Topical Report BAW-10222P, Rev. 0). Table 2 summarizes the results of this postulated accident when the axial location of peak power is varied along the length of the pin at any time during the warranted fuel burnup range for the Mk-B9 fuel design. TMI-1 Cycle 13 will also load a modified Mk-B9 fuel design (designated Mk-B10P) and the Mk-B8 fuel design. The revised 20% SGTP LOCA Linear Heat Rates for Mk-B8, Mk-B9 and Mk-B10P will be used in the determination of core operating limits for Cycle 13 reload. The revised LOCA LHRs will also be placed in the TMI-1 Core Operating Limits Report (COLR) prior to Cycle 13 startup.

Reactor Building sump pH following a postulated LOCA will remain in the range of 8.0 to 11.0. A calculation of sump pH following a maximum hypothetical accident

shows the sump pH would be in the range of 8.0 to 9.0 (GPU Nuclear Calculation C-1101-900-E610-070). A reduction in RCS volume will result in less boric acid in the sump and consequently a slightly higher sump pH.

A sensitivity study was performed to evaluate the effect of various tube plugging configurations. The following tube plugging configurations were evaluated (SG2 is in the loop with the break):

1. 0% SGTP
2. 15% SG1 - 25% SG2
3. 25% SG1 - 15% SG2
4. 0% SG1 - 25% SG2

The results of the sensitivity study indicated that increased resistance in the broken loop has approximately a five times greater effect on the predicted peak clad temperature (PCT) than a similar increase in the intact loop. Therefore, it is conservative to model asymmetric tube plugging with greater numbers of plugged tubes in the affected (broken) loop. The LOCA analyses establish a unit average SGTP limit of 20% and a SGTP limit of 25% on a single OTSG, supported by the specific analyses. Based on the above sensitivity studies, the LOCA analysis does not establish the tube plugging asymmetry limits.

Small break sensitivity studies performed with the RELAP5-based EM for the 177-FA lowered loop show that the most limiting break location is in the bottom of the cold leg piping between the reactor vessel inlet nozzle and the HPI nozzle. Two special break cases, an HPI line break and a core flood tank (CFT) line break, were also analyzed. Table 3 summarizes the results of the Mk-B9 small break LOCA calculations performed for a spectrum of cold leg pump discharge breaks, an HPI line break, and a CFT line break.

10 CFR 50.46 specifies that the emergency core cooling system for a commercial nuclear power plant must meet five criteria.

1. The calculated peak cladding temperature (PCT) is less than 2200° F.
2. The maximum calculated local cladding oxidation is less than 17.0 percent.
3. The maximum amount of core-wide oxidation does not exceed 1.0 percent of the fuel cladding.
4. The core remains amenable to cooling.
5. Long-term cooling is established and maintained after the LOCA.

The large and small break LOCA calculations demonstrate compliance with these criteria for breaks up to and including the double-ended severance of the largest primary coolant pipe for the TMI-1 plant with 20% SGTP (FTI Proprietary) Document Nos. 32-1266343-00 and 32-1244481-00).

Maximum Hypothetical Accident

The analysis of this accident in the UFSAR assumes a large release of fission products to the reactor building. The release is not mechanistic and a means for it to occur is not postulated. The intent of the analysis is to evaluate the radiological consequences. Due to the non-mechanistic nature of this event, steam generator tube plugging will have no effect on the event.

Waste Gas Tank Rupture

The waste gas tank is located in the auxiliary building and the analysis of its rupture is not related to the steam generators' function.

Loss of Main Feedwater

A loss of feedwater (LOFW) may result from abnormal closure of the feedwater isolation valves, abnormal control valve failure or main feedwater pump failure. A loss of normal flow through the secondary system will result in a reduction in secondary heat removal, causing the RCS temperature to increase. Due to this temperature increase, the reactor coolant begins to expand causing the RCS pressure to increase.

Increasing RCS temperature and pressure could result in the RCS filling solid, a failure of the RCS, or a fuel cladding failure. Reactor protection for these events is provided by the high RCS pressure trip function of the reactor protection system (RPS). When the RCS pressure reaches the high RCS pressure setpoint, the reactor is tripped. Shortly after reactor trip, only decay and pump heat are added to the reactor coolant. Initially, the EFW flow rate is not able to keep up with decay and pump heat. Therefore, the reactor coolant will continue to expand until the EFW heat removal rate matches the decay and pump heat input rate. Subsequent to this point in time, RCS pressure will decrease and the reactor coolant will contract.

The existing acceptance criteria for the loss of feedwater accident are:

1. The RCS pressure will be limited to less than 110% of design pressure or 2750 psig.
2. The reactor thermal power shall not exceed 112% of rated power.
3. The pressurizer will not become water solid

The LOFW was re-analyzed for 20% average SGTP with bounding analysis assumptions(GPU Nuclear Calculation C-1101-224-E610-070). Two cases were analyzed:

1) Peak pressure case – no PORV, no spray
2) Peak pressurizer level case – PORV, spray

The pressurizer power operated relief valve (PORV) is a non-safety grade component. Therefore it is not usually modeled in safety analysis. However, in the case of a

LOFW, actuation of the PORV to control system pressure would aggravate the liquid insurge to the pressurizer by venting steam from the pressurizer at a lower pressure than would the pressurizer safety valves. Consequently, the PORV was included in this analysis to provide a conservative prediction of pressurizer liquid level. Similarly, pressurizer spray is a non-safety grade pressure control system. However, actuation of pressurizer spray flow could worsen the pressurizer liquid level response during the event by condensing the pressurizer steam bubble. Consequently, the LOFW accident was analyzed with pressurizer spray to provide a conservative prediction of pressurizer liquid level.

The results of the analyses show that for the peak pressure case, RCS pressure is controlled at the safety valve setpoint, which is well below the acceptance criteria of 2750 psig. For the peak pressurizer level case, the pressurizer goes water solid, however RCS pressure is below the PORV and PSV setpoint. The intent of the criteria for not having a water solid pressurizer is to assure that liquid is not passed through the pressurizer safety or relief valves. Since RCS pressure is below the setpoints of these valves at the time the pressurizer goes solid, the intent of the criteria is met. Therefore, it is concluded that the LOFW accident results in acceptable consequences with 20% average SGTP. The TMI-1 UFSAR will be updated to reflect this change in the existing accident analysis acceptance criteria.

A main feedwater line break outside containment (upstream of the feedwater check valve) is similar to a LOFW as the check valves in the main feedwater lines prevent the abrupt loss of feedwater from the steam generators. Therefore, this event will be very similar to the LOFW event discussed above, and thus similarly concluded that the consequences with 20% SGTP are acceptable.

Fuel Cask Drop Accident

The fuel cask drop accident is the dropping of a fuel cask through the maximum drop height and is not related to the steam generators' function.

Anticipated Transient Without Scram

In compliance with the ATWS Rule (10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants), TMI-1 has installed a Diverse Scram System (DSS). This significantly reduces the likelihood of failure to shutdown the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

The primary concern during an ATWS event is overpressurization, and the limiting ATWS transient was determined to be the loss of main feedwater (LOFW). The peak RCS pressure for the LOFW with failure to scram but with DSS actuation at 2500 psia, is limited to well below the system design pressure of 2765 psia. Since the transient is essentially over after a reactor scram occurs, large margins to the acceptance criteria will be maintained even with 20% steam generator tube plugging.

Containment Integrity

Loss of Coolant Accident Mass and Energy Release

The initial mass in the RCS decreases by about 3.7% as a result of 20% average tube plugging. The RCS hot leg temperature is predicted to increase about 1.25° F (from the current tube plugging configuration – see Table 1) and the cold leg temperature decreases about 1.25° F with a constant T_{ave}. The result is a reduction in the stored energy of the reactor coolant of approximately 3.7%. The change in coolant density as a result of the change in cold leg temperature is insignificant. The mass and energy release to containment following a LBLOCA would be less with plugged tubes than without. Additionally, increased tube plugging in the steam generators would increase the resistance to flow from the steam generator side of the break. The result would be a reduction in mass flowrate to containment. Therefore, the current mass and energy release rates used in the peak containment pressure analysis and the containment EQ temperature and pressure analysis remain conservative.

Main Steam Line Break (MSLB) Mass and Energy Release

Inside Containment

During the original licensing of the plant, the containment building response was bounded by the large break LOCA and not the MSLB. The small increase in OTSG secondary inventory mass as a result of tube plugging will be offset by the reduction in OTSG superheat such that the large break LOCA mass and energy release will remain bounding.

Outside Containment

The current analysis of record for main steam line break in the Intermediate Building is based on various break sizes and locations. A guillotine break of a single main steam line was analyzed with the maximum possible OTSG inventory, including a full downcomer. The total mass in a single OTSG with up to 25% tube plugging does not exceed the mass used in the current analysis. The smaller breaks were analyzed in various locations on the smaller lines branching from the main steam line, particularly the supply line to the turbine-driven EFW pump. Those breaks were analyzed using critical flow calculations because they do not cause a reactor trip. The break is terminated by operator action. The mass and energy release was calculated based on the operating conditions for the OTSG without tube plugging. Tube plugging increases OTSG secondary side fluid mass slightly, increases operating level slightly and decreases the superheat region of the OTSG. Therefore, steam superheat decreases with increasing tube plugging. Decreasing the steam superheat would reduce the energy released to the Intermediate Building. Therefore, the current analyses of the main steam line breaks in the Intermediate Building remain conservative.

B. Impacts on Reactor Trip Setpoints and Safety Limits

The TMI-1 Technical Specifications (T.S.) contain the RPS trip setpoints in T.S. Table 2.3-1, including the core P-T trip setpoints in T.S. Figure 2.3-1. The Technical Specifications also contain the core protection safety limit (T.S. Figure 2.1-1) and corresponding safety bases (T.S. Figure 2.1-3). The impacts of the 20% SGTP on these setpoints and limits were evaluated and reanalyzed. The results show that all the current RPS trip setpoints are still applicable while the core protection safety limit and its bases given in T.S. Figures 2.1-1 and 2.1-3 become slightly more restrictive than those in the current T.S.

The core protection safety limit (T.S. Figure 2.1-1) and bases (T.S. Figure 2.1-3) define the region of unacceptable operation when the conditions of the RCS approach the DNB limit. The safety criteria in determining these limits are entirely based on DNB protection. The applicable BWC CHF correlation DNB limit for TMI-1 is 1.18 or a coolant quality limit of 26%. Currently, the P-T RPS trip setpoints in T.S. Figure 2.3-1 are more restrictive than the variable low pressure (VLP) setpoints, which are derived from the core protection safety limit. This is because the VLP setpoints are outside the box established by the pressure-temperature RPS trip setpoints.

The core protection safety limits were re-analyzed with 20% SGTP based on the reduced minimum RCS design flow (102% from 106.5% of 352000 gpm). A series of VIPRE hot channel analyses were performed by varying RCS flow, inlet temperature and system pressure (GPU Nuclear Calculation C-1101-202-E620-365). The DNB analysis for each particular pump condition was performed and results are given in attached Figure 1 (new core protection safety limit) and Figure 2 (core protection safety bases) which correspond to the proposed revised TMI-1 Technical Specification Figures 2.1-1 and 2.1-3. These results are also compared against the existing safety limits and bases in T.S. Figures 2.1-1 and 2.1-3. The comparison is shown on Figure 3, which shows that the 20% SGTP limit results in a slightly more restrictive safety limit. In Figure 4, the VLP setpoints are overlaid with the TMI-1 protection system maximum allowable setpoints (T.S. Figure 2.3-1). As can be seen from this figure, the existing P-T setpoints bound the VLP setpoints. Consequently, the P-T trip setpoints in T.S. Figure 2.3-1 are still applicable with 20% SGTP limit.

The loss of coolant flow (LOCF) events were re-analyzed to evaluate the impacts of the 20% SGTP on the flux/flow setpoint and power/pump status trip setpoint in T.S. Table 2.3-1. The safety criteria of these setpoints are DNB protection. The one pump coastdown from the four pump operation (4/3) and locked rotor accident are terminated by the flux/flow setpoint (=1.08) and four pump coastdown (4/0) event is terminated by the power/pump status trip. The minimum DNBR results based on the reduced flow demonstrate that both trip setpoints are still applicable with the 20% SGTP. The new minimum DNBR for the 4/3 pump coastdown is 1.484 (against 1.18 design limit) and 1.276 for the locked rotor accident (design limit is MDNBR=1.0).

The MDNBR result for the 4/0 coastdown was 1.669 against the design limit of 1.18.

C. Other Considerations

Components Evaluation

The increased steam generator tube plugging will cause a calculated increase in RCS T_{hot} of about 1.25° F, from a nominal 602.2° F to 603.45 ° F with a constant T_{ave}. The corresponding decrease in T_{cold} will be about 1.25° F from a nominal 556.7° F to 555.45° F with a constant T_{ave}. This is a very small increase in T_{hot}. The effect of this small temperature increase on the primary system components, including the steam generators, has been evaluated and found to be negligible.

The majority of the TMI-1 primary system was designed for a primary side temperature limit of 650° F; the pressurizer was designed for a temperature of 670° F. Thus, the effects on the reactor coolant pumps, piping, seals, valves, welds, bolting, and other components are judged to be insignificant since they were all designed for higher temperatures than the postulated T_{hot} of 603.45° F. In addition, the primary system instrumentation to monitor the operation of these components was designed for the higher temperatures, and has sufficient span to monitor the primary system operation at the higher temperatures. In summary, temperatures greater than a T_{hot} of 603.45° F were anticipated in the design of the system.

While the design temperature of the primary system is 650° F or greater, there are some B&W plant analyses that utilize T_{hot} as an input. Operation at T_{hot} of 608° F nominal has been evaluated generically based on steam generator tubes virtually identical to those of TMI-1. As a result, the majority of the B&W plant generic analyses use a T_{hot} of 608.6° F in order to bound the T_{hot} conditions at all B&W plants. A number of these documents were reviewed to ensure that a T_{hot} of greater than 603.45° F was utilized. The generic plant analyses and specifications bound the proposed operating T_{hot} of 603.45° F.

The resulting flows and temperatures based on a parametric study of tube plugging in the TMI-1 steam generators (Reference FTI Proprietary Document 32-1234876-02) have been assessed for any potential impact on the existing structural analysis bases. This assessment has determined that the changes in the steam generator characteristics due to having 20% of the tubes plugged produce no significant change in the existing design analysis for the TMI-1 RCS based on the following:

1. A reduction in Cold Leg temperature can affect the magnitude of pressure waves traveling through the RCS internals during a LOCA, thus increasing loads. The impact of the reduction in the Cold Leg temperature due to plugging on LOCA break severity has been addressed and concluded to be insignificant.

2. Primary piping thermal expansion is also a concern when RCS temperatures change. The Cold Leg temperature decreased slightly and the Hot Leg temperature

increased slightly as a result of 20% plugging. Thermal expansion stresses in the Cold Leg, therefore, will go down and stresses in the Hot Leg will go up. The thermal expansion increase in the Hot Leg is negligible.

3. The system flowrate decreases as a result of increased plugging in the OTSGs. In general, system through-wall temperature gradients will be effected by this decrease in flowrate since film coefficients will be reduced, and thus, heat transfer is slowed at the fluid/metal interface. Through-wall temperature gradients are therefore reduced by a decrease in system flowrate.

4. The small increase in Hot Leg temperature due to 20% plugging of the OTSGs has been concluded to have a negligible effect on TMI-1 operating transients. Therefore, the present stress ranges analyzed, including radial gradients and discontinuities, remain valid.

Increasing T_{hot} without an increase in the RCS pressure will decrease the top-to-bottom temperature difference in the pressurizer, and will reduce the stresses induced by thermal stratification in this vessel. In the power operating range, the top-to-bottom temperature difference is small and affects only the thermal striping in the pressurizer surge line. However, the reduction in top-to-bottom temperature difference by the increase in T_{hot} will also reduce the fatigue usage due to thermal stratification and striping in the surge line.

The small increase in operating T_{hot} will not affect the TMI-1 operating P/T limits because, in this higher temperature range, the reactor vessel materials have adequate ductility and toughness so as not to be a concern with respect to non-ductile failures.

There are many chemical reactions that will increase in rate with temperature. An increase in T_{hot} will theoretically lead to an increase in corrosion rates of those primary system components which are affected by a T_{hot} increase. However, the small incremental increase from 602.2 °F to 603.45° F is not expected to result in a significant increase in corrosion rates.

Given the above, the structural integrity of the primary system should not be adversely effected by the small calculated increase in T_{hot}. Previous analyses and designs have bounded the impact of the slightly higher primary side temperature upon primary side components.

LOCA Hydraulic Forces

The LOCA hydraulic forces were originally evaluated at nominal 100% power conditions with no tube plugging and were not reanalyzed for 20% tube plugging. The plant operating pressure will not change, and there is a small increase (approximately 1.25° F) in the hot leg temperature and a small decrease (approximately 1.25° F) in the cold leg temperature. The only effect on the hydraulic

forces would be from the change in the saturation pressure at the break location, which would be negligible. Thus, a 20% SGTP would have a negligible effect on the LOCA hydraulic forces and the existing analysis is considered adequate.

Post-LOCA Long Term Core Cooling

The LOCA analysis as described previously demonstrated that the core is quenched, and the cladding temperature is returned to near saturation temperature. Thereafter, long-term cooling is achieved by the ECCS, and these systems are redundant and are able to provide a continuous flow of cooling water to the core fuel assemblies so long as the coolant channels in the core remain open. This assures, that long term, the core temperatures will be maintained at an acceptably low value, and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

To ensure that the post-LOCA core heat load is limited to decay heat, a 1% shutdown margin is maintained in the core. The cycle specific minimum required BWST boron concentration calculation is verified against the Core Operating Limits Report (COLR) minimum requirements prior to each refueling. The calculation is performed assuming the entire RCS volume, including all OTSG tubes, is mixed with the other sources of water following the LOCA. Plugging steam generator tubes reduces the available RCS volume, which would reduce the BWST boron concentration required to assure 1% shutdown margin. Therefore, the shutdown requirement calculation methodology is conservative with respect to the reduction in RCS volume caused by tube plugging.

Boron Precipitation Prevention

The boron concentration calculations that demonstrate the prevention of boron precipitation are performed using control volumes that are limited to the reactor vessel downcomer, lower plenum, core region and outlet plenum. The boron concentration in the reactor vessel following the blowdown, reflood and refill phases of a LOCA is not affected by the inventory on the primary side of the OTSGs. Reduced RCS volume from plugging tubes would increase the RB sump boron concentration following a LOCA by a small amount, but will remain less than the boron concentration of the BWST. The LPI pump suction is switched from the BWST to the RB sump when the BWST is depleted. However, the boron concentration calculations assume a constant boron concentration from the injection source. Therefore, the boron concentration calculations are unaffected by OTSG tube plugging.

Steam Generator Overfill

The maximum plugging for any one steam generator is limited to 25% and will result in a secondary side inventory increase of approximately 13% or 5200 lbm. With the increased initial steam generator level, the time for the operator to respond to the high

level alarm will be shortened slightly, but since the symptoms of overfill are unambiguous, prompt operator response is expected to prevent its occurrence. In addition, there is no safety concern with overfill, as a stress analysis has been performed on the consequences of flooding the TMI-1 main steam line. The results of dead weight, internal pressure, and thermal expansion analyses show that the main steam piping can withstand these effects. Therefore, operating a single steam generator with up to 25% plugged tubes will not present a safety concern with respect to steam generator overfill.

Control Rod Guide Tube Boiling

Increased tube plugging reduces the total RCS flow. The reduction in RCS flow will result in a reduction of flow through the control rod guide tubes. Bulk boiling inside these tubes under normal long-term full power operation is to be avoided. Therefore, an analysis was performed based on 20% SGTP to determine the margin to bulk boiling in the limiting control rod guide tubes. The analysis assumed a conservative power level of 102% of 2772 MWt with a radial peak of 1.8. The RCS flow rate is expected to be greater (106.4%) than 102% design flow with 20% tube plugging. With a minimum of 102% of RCS design flow, there was margin to boiling in the limiting control rod guide tube. Therefore, the proposed increased tube plugging would not cause bulk boiling in the control rod guide tubes.

Asymmetric Tube Plugging Effects

Plugging more tubes in one steam generator than another causes a mismatch in flow between the loops. The flow rate will be greater in the loop with fewer plugged tubes. This flow arrangement can cause an asymmetry of flow at the inlet to the core. An assessment of reactor vessel mixing has shown that there is little difference in inlet velocity and temperature distribution at the core inlet as a result of the loop flow asymmetry. This review was based on a RCS loop configuration with a large tube plugging asymmetry of 20%/0%. A constant, bounding inlet velocity and radially asymmetric temperature distribution was applied at the core inlet to evaluate the impact on power peaking factors. Evaluations using a three-dimensional nuclear design code indicate negligible differences in 3-D power peaking throughout a fuel cycle depletion. In addition, core thermal hydraulic analyses for 20% SGTP included a 0.5% flow penalty to conservatively account for asymmetric tube plugging. The generation of maximum allowable peaking (MAP) limits for DNB margin analyses will also account for asymmetric tube plugging. MAP limits are generated as part of the standard reload licensing analysis.

Fuel Rod and Fuel Assembly Mechanical Analysis

An average SGTP of 20% will result in about 3.6% core flow reduction and approximately 2.5° F larger axial temperature difference across the core (including a 1.25° F increase in T_{hot}). The impact on cladding corrosion calculations are performed as part of the reload analysis for each operating cycle. For each sub-batch of fuel in

the cycle design, the pin power history for the peak burnup pin is calculated and input into an approved fuel performance code, which is used to calculate cladding corrosion. RCS flowrate and core inlet temperature are also input to the code. The cladding corrosion results are compared against acceptance criteria. Exceeding the acceptance criteria would require a change to the cycle design in order to alter the pin power histories such that acceptable corrosion results can be obtained. Therefore, any impact due to tube plugging will be addressed by the cycle-specific analyses.

The fuel assembly mechanical analyses which could be effected by these changes are: guide tube corrosion and growth, spacer grid corrosion and growth, fuel assembly holddown force, and hydraulic loads on the fuel assembly. The reduction in RCS flow rate will result in lower hydraulic lift forces and thus increased margin in holddown spring force and grid restraint loads. Resulting compressive guide tube loads during operation will be slightly higher, however sufficient margin exists to accommodate the minimal load increase. The temperature differential assumed in current vendor analyses bounds the increase resulting from operation with 20% SGTP. With the 1.25° F T_{hot} increase, the effect on guide tube and grid corrosion and growth will be minimal and is bounded by existing analyses. Therefore, the effects of the 20% average SGTP will be small, and the structural integrity of the fuel assemblies will be maintained.

TABLE 1

20% SGTP Calculated Change In Operating Parameters

Parameter	CY 12 SGTP**	20% SGTP
Power, MWt	2568	2568
RCS Pressure, psig	2170	2170
RCS Flow (% design flow)	110%	106.4%
T_{hot}, °F	602.2	603.45
T_{avg}, °F	579.5	579.5
T_{cold}, °F	556.7	555.45
OTSG Pressure, psia	925	925
OTSG Superheat, °F	50	44*

* Remains above design value of 35° F.

** Cycle 12 has 1300 tubes plugged in the A steam generator and 395 tubes plugged in the B steam generator. (In addition, there are 248 sleeved tubes inservice in the "A" OTSG and 253 sleeved tubes inservice in the "B" OTSG.)

TABLE 2

Summary of Mk-B9 LHR Limits

TMI-1 LBLOCA 20% Tube Plugging

Elevation	LOCA LHR Limit, kW/ft (PCT, F) BOL	LOCA LHR Limit, kW/ft (PCT, F) MOL (40000 MWD/mtU)	LOCA LHR Limit, kW/ft (PCT, F) EOL (60000 MWD/mtU)
0.0 ft.	15.96	15.96	12.0
2.506 ft.	16.8 (2007.9)	16.8 (1859.2)	12.0 (1527.0)
4.264 ft.	17.1 (2020.2)	17.1 (1862.0)	12.0 (<1700)
6.021 ft.	17.5 (1993.3)	17.5 (1890.1)	12.0 (<1700)
7.[illegible] ft.	17.3 (1954.2)	17.3 (1857.8)	12.0 (<1700)
9.536 ft.	16.8 (1901.6)	16.8 (1804.5)	12.0 (<1700)
12.0 ft.	15.96	15.96	12.0

Notes:

1. Linear interpolation for LHR limits is allowed between 40000 MWD/mtU and 60000 MWD/mtU.

2. For the BOL and MOL columns, the LHR limits below 2.506 feet are reduced linearly to $0.95*LHR_{2.506}$ at 0.0 feet. The LHR limits above 9.536 feet are reduced linearly to $0.95*LHR_{9.536}$ at 12.0 feet.

3. Analyses at BOL and MOL used a steady state energy deposition factor (EDF) of 0.973 for initial core energy deposition and a transient EDF of 0.973. The EDF is used to relate nuclear source power (used in maneuvering analyses) to thermal source power (used in LOCA analyses).

$$LHR_{Thermal} = LHR_{Nuclear} * EDF$$

Analyses at EOL used a steady state EDF of 1.0 and a transient EDF of 1.1. To report the LHR given above on a basis consistent with the BOL and MOL LHRs, it was necessary to divide the EOL LHR of 11.7 kW/ft by an EDF of 0.973 to obtain a reported LHR of 12.0 kW/ft. Therefore, all of the LHRs given above are based on nuclear source power with an EDF of 0.973.

4. LHRs are valid for fuel enrichments of 5.1 weight percent (maximum).

TABLE 3

Summary of Calculated PCTs for Mk-B9 SBLOCA Analyses

TMI-1 SBLOCA 20% Tube Plugging

Break Size (ft^2)	Peak Clad Temperature (PCT) (F)	Time of PCT (s)
0.75	859	98
0.50	845	150
0.30	790	270
0.15	922	388
0.10	1334	1045
0.09	1354	1267
0.08	1375	1480
0.07	1331	1577
0.06	1357	1818
0.05	1412	2550
0.04	1361	3314
0.03	1287	5126
0.01	715	0.0
0.44 (CFT Line)	715	0.0
0.02463 (HPI Line)	1297	5570

Figure 1
TMI-1 Core Protection Safety Limit

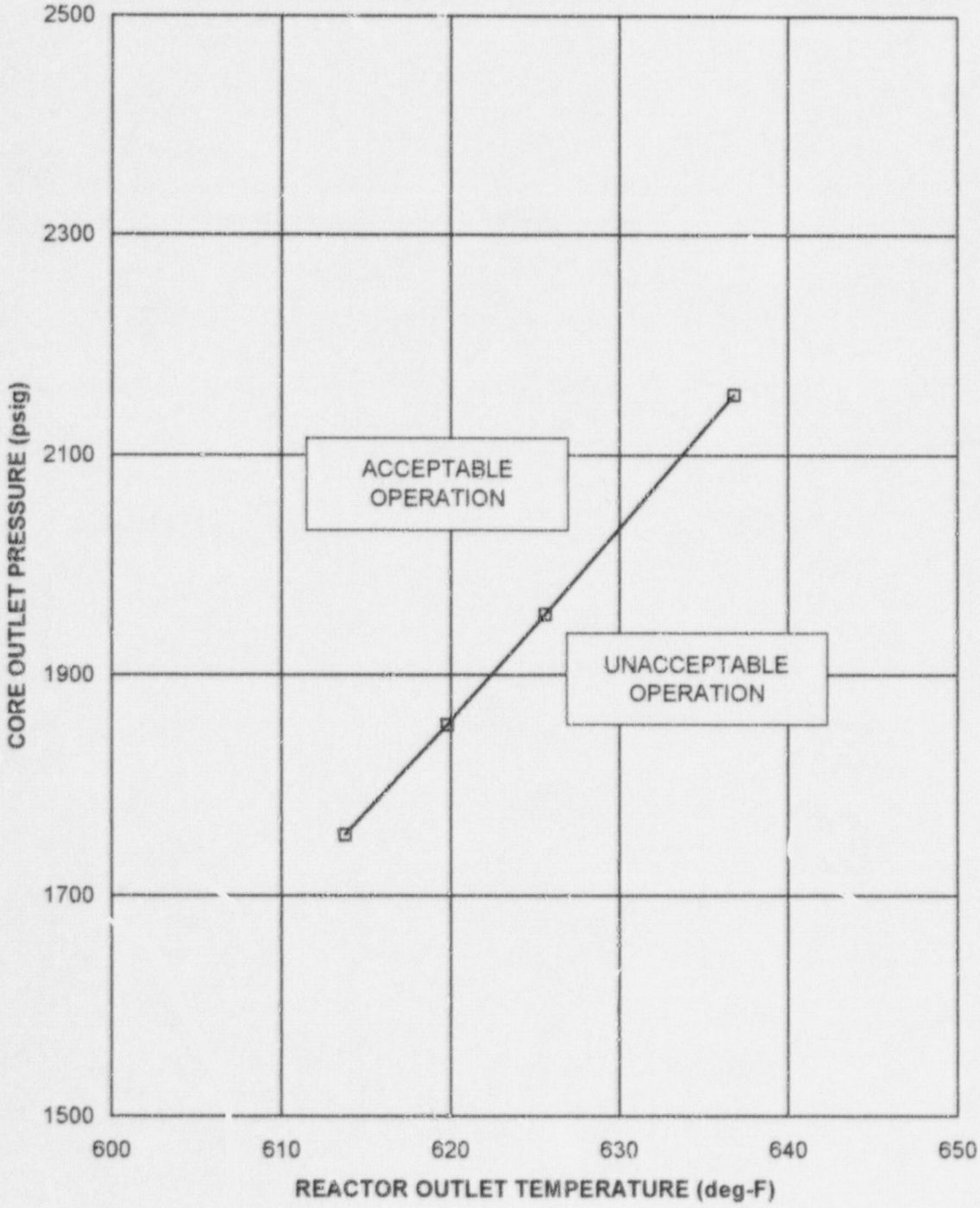


Figure 2

TMI-1 Core Protection Safety Bases

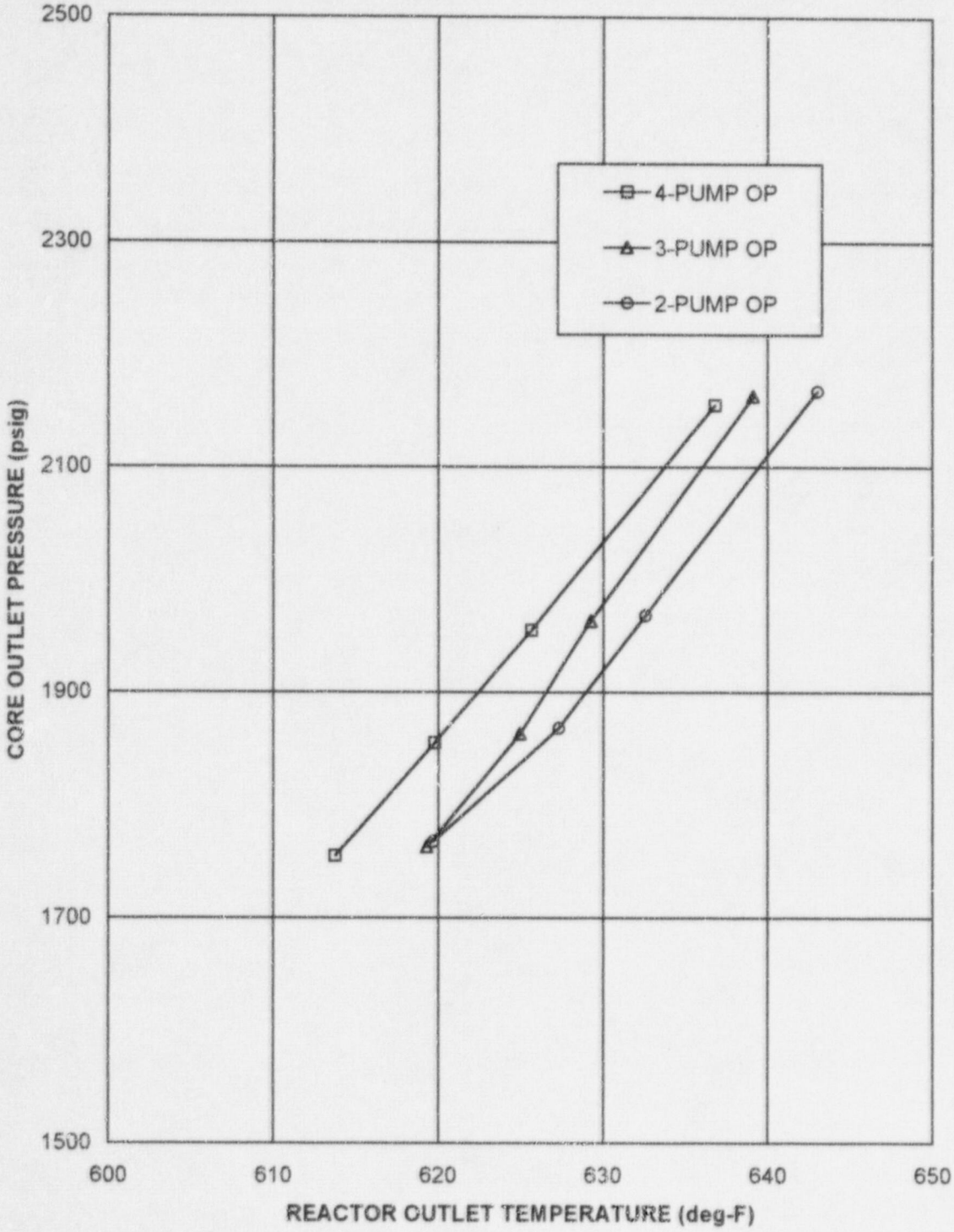


Figure 3

TMI-1 Core Protection Safety Limit Comparison

(With 20% Average OTSG Tube Plugging)

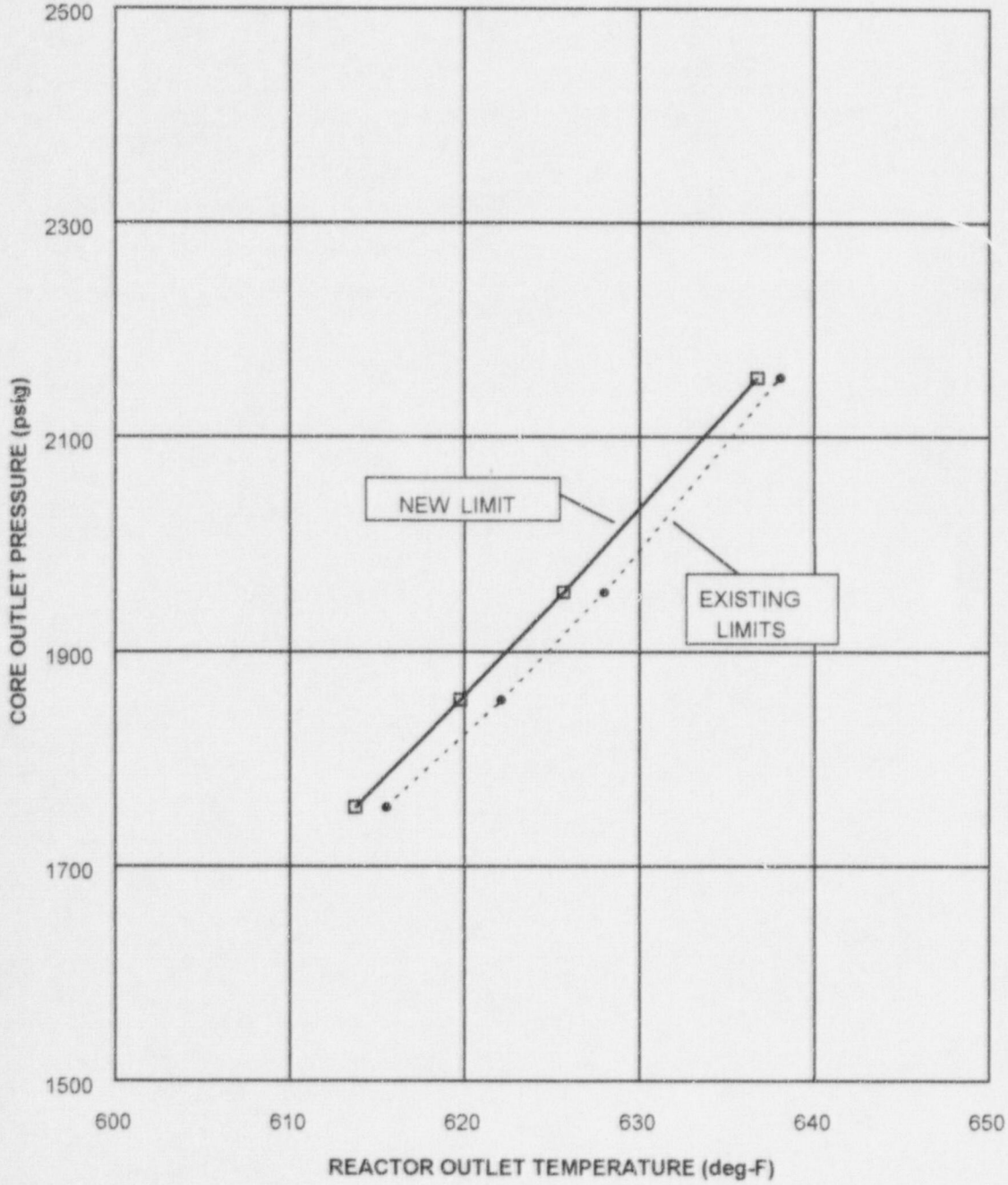
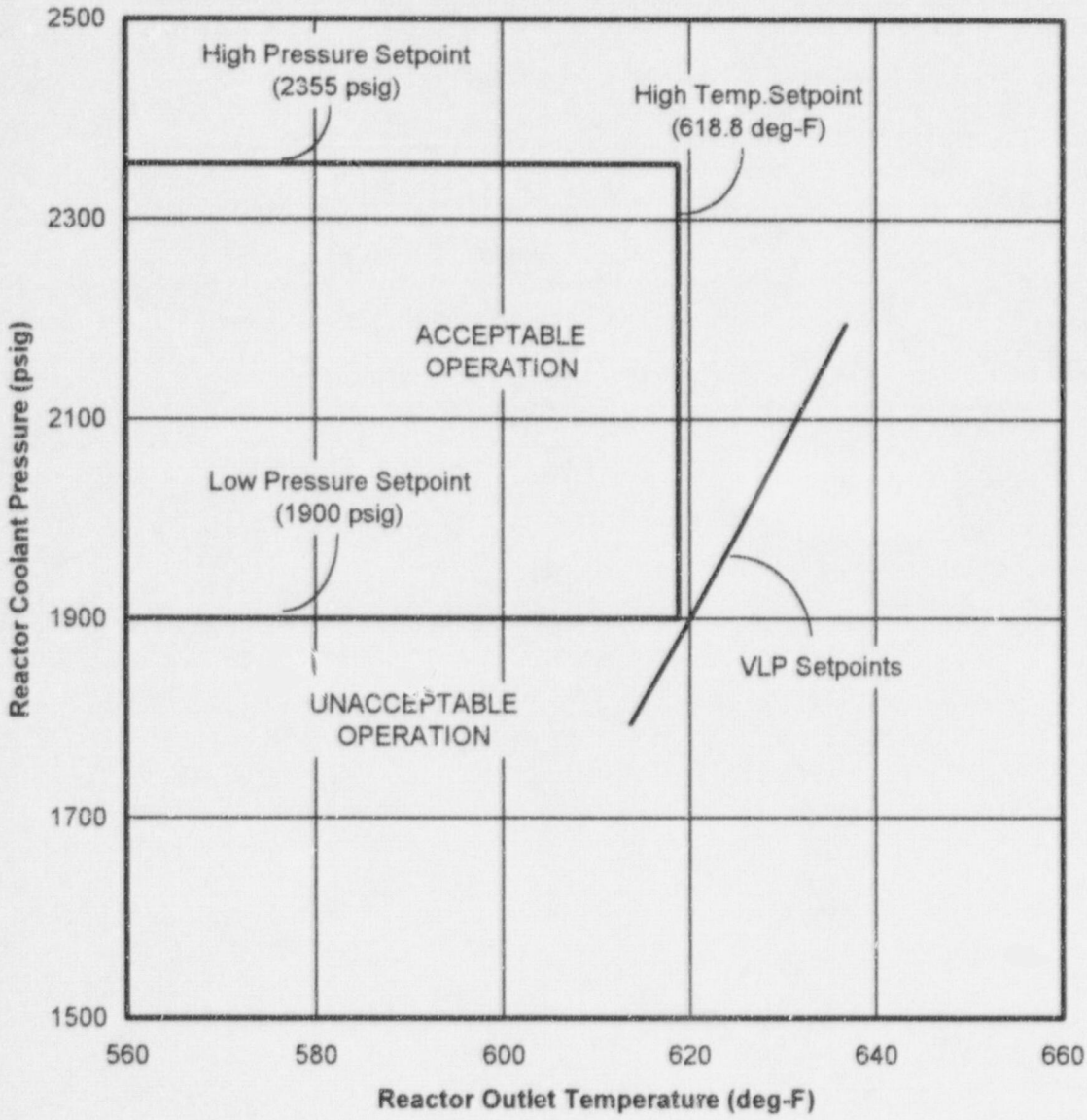


Figure 4

Protection System Maximum Allowable Setpoints
(With 20% Average OTSG Plugging)



IV. No Significant Hazards Consideration

GPU Nuclear has determined that this Technical Specification Change Request poses no significant hazards consideration as defined by 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 the consequences of an accident previously evaluated. An increase in the average steam generator tube plugging (SGTP) level to 20% results in a small reduction of reactor coolant system (RCS) flow rates and primary to secondary heat transfer. These changes result in small changes to the primary and secondary side operating parameters, and do not result in any additional challenges to plant equipment. The proposed Technical Specification Changes resulting from the increase in allowable tube plugging limits are more restrictive but remain bounded by the existing reactor protection system (RPS) trip setpoints. The assessment of the NSSS primary components, including the reactor pressure vessel, reactor core, reactor coolant pump, steam generator, pressurizer, control rod drive mechanisms, and RCS piping concluded that the integrity of these components will be unaffected by the increase in average SGTP level.

   A re-analysis of the bounding Updated Final Safety Analysis Report (UFSAR) Chapter 14 accidents, specifically the startup accident, loss of coolant flow, loss of feedwater, and large and small break LOCA demonstrated compliance with the acceptance criteria. The RCS pressure boundary is not challenged, and the DNBR and peak clad temperature values remain within the specified limits of the licensing basis. An analysis of the loss of electric power accident demonstrated the ability of the plant to transition smoothly to natural circulation with an average 20% SGTP or with asymmetric plugging. It was also determined that the current mass and energy release data used for containment integrity and equipment qualification remain bounding. Since the design requirements and safety limits continue to be met, system functions are not adversely impacted, and the integrity of the RCS pressure boundary is not challenged, the radiological consequences remain unchanged. Therefore, this activity does not involve a significant increase in 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 previously evaluated. The proposed Technical Specification changes are more restrictive core protection safety limits but remain bounded by the existing RPS trip setpoints. This proposed change assures safe operation commensurate with the effects of steam generator tube plugging. The increase in the average level of SGTP to 20% will not introduce any new accident initiator mechanisms. No new failure modes or limiting single failures have been identified. Since the safety and design requirements continue to be met and the integrity of the RCS pressure boundary is not challenged, no new accident scenarios have been created. This change does not add any new equipment, modify any interfaces with any existing equipment, or change the equipment function or the method of operating the equipment. Reactor core, RCS, and steam generator

parameters remain within appropriate design limits during normal operation. Therefore, this activity does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The existing RPS trip setpoints bound the proposed Technical Specification changes resulting from 20% SGTP. This change assures safe operation commensurate with the effects of steam generator tube plugging. The TMI-1 DNB design basis, RCS pressure limits, peak clad temperature limits and dose criteria are maintained for all UFSAR transients. Therefore, this activity does not reduce the margin of safety.

V. Implementation

GPU Nuclear requests that the amendment authorizing this change become effective immediately upon issuance.