50-289

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 19, 1999

Mr. James W. Langenbach, Vice President
and Director, TMI
GPU Nuclear, Inc.
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGE REQUEST NO. 279 - CORE PROTECTION SAFETY LIMIT WITH AN AVERAGE OF 20 PERCENT STEAM GENERATOR TUBES PLUGGED (TAC NO. MA4301)

Dear Mr. Langenbach:

The Commission has issued the enclosed Amendment No. 214 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1) in response to your application dated December 3, 1998, as supplemented by letters dated March 26, April 16, May 7, May 21, June 4, June 15, and June 29, 1999. In addition, the staff considered information provided in your letters dated February 5 and May 12, 1999, in making its determination.

The amendment revises TMI-1 Technical Specification Figure 2.1-1 "Core Protection Safety Limit," and Figure 2.1-3 "Core Protection Safety Bases" to reflect a decrease in reactor coolant system flow resulting from a revised analysis to allow operation of the TMI-1 facility with an average of 20 percent of the steam generator tubes plugged, and no more than 25 percent plugged in either generator.

As discussed with your staff, this amendment is contingent upon GPU Nuclear, Inc., demonstrating that the emergency feedwater system (EFWS) pumps can deliver the flow rate assumed in the analyses, and that the developed head at this flow rate is at least that calculated in the your evaluation of the EFWS, as described in the enclosed safety evaluation and will be effective upon completion of that demonstration. You are requested to notify the NRC staff upon completion of this surveillance.

The staff notes that in your December 3, 1998, application for the sale and license transfer of TMI-1 to AmerGen, this amendment was identified as one that must be completed by April 15, 1999, in order to support the sale and license transfer. Sufficient justification was not provided in either the December 3, 1998, application for sale and license transfer or in the December 3, 1998, application for this amendment to support such a high priority and accelerated review schedule. The staff, nonetheless, attempted to support your schedule. However, several requests for additional information were required as well as resolution of emergent issues having potential effect on your analyses before the staff could complete its review. The response to the requests for information were less timely than expected given your requested review schedule. This indicates that the review was not as straightforward as indicated in your submittal. The above information may result in a Plant Issues Matrix (PIM) entry by Region I when that document is forwarded to you following completion of its semi-annual performance review of your facility.

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A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Timothy G. Colburn

Timothy G. Colburn, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures: 1. Amendment No. 214 to DPR-50
2. Safety Evaluation

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A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Timothy G. Colburn, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
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DATED: August 19, 1999

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR, INC.

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission or NRC) has found that:

   A. The application for amendment by GPU Nuclear, Inc., et al. (the licensee) dated December 3, 1998, as supplemented March 26, April 16, May 7, May 21, June 4, June 15, and June 29, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 214 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date, as documented by the licensee in plant records, that surveillance of the emergency feedwater system (EFWS) has demonstrated that: (a) The EFWS pumps can deliver a minimum flow rate of 550 gpm (assumed in the licensee's analyses supporting the amendment) and (b) the developed head at the minimum flow rate specified in item (a) is at least 2,662 feet plus instrument uncertainties. This amendment is to be implemented within 30 days of the date of the demonstration of the flow rate and head, as documented by the licensee, as stated above. The licensee plans to perform the demonstration described above during the 13R refueling outage, which is scheduled to begin on September 10, 1999; should this demonstration, however, not be made on or before December 31, 1999, this amendment shall be void and of no effect.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 19, 1999

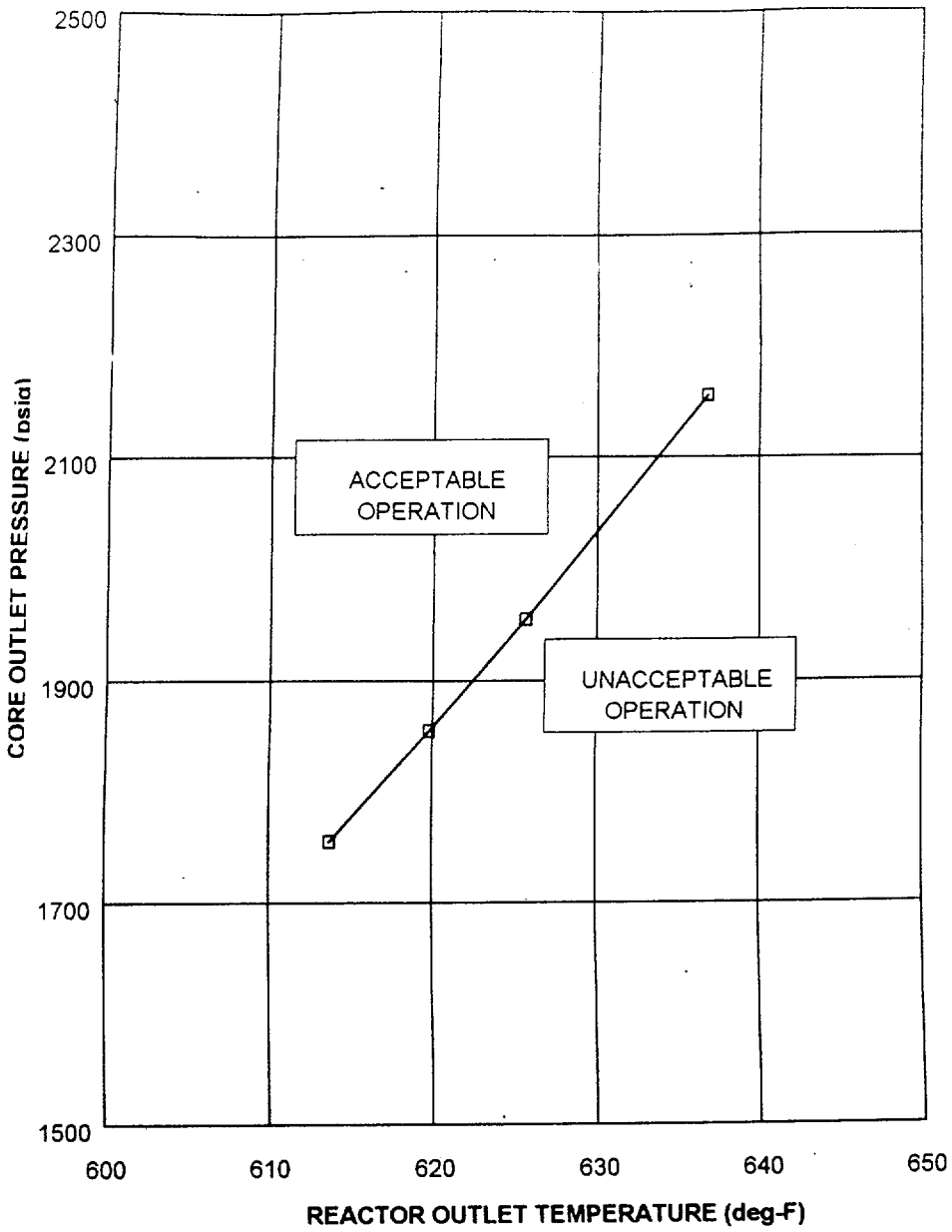
ATTACHMENT TO LICENSE AMENDMENT NO. 214

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change.

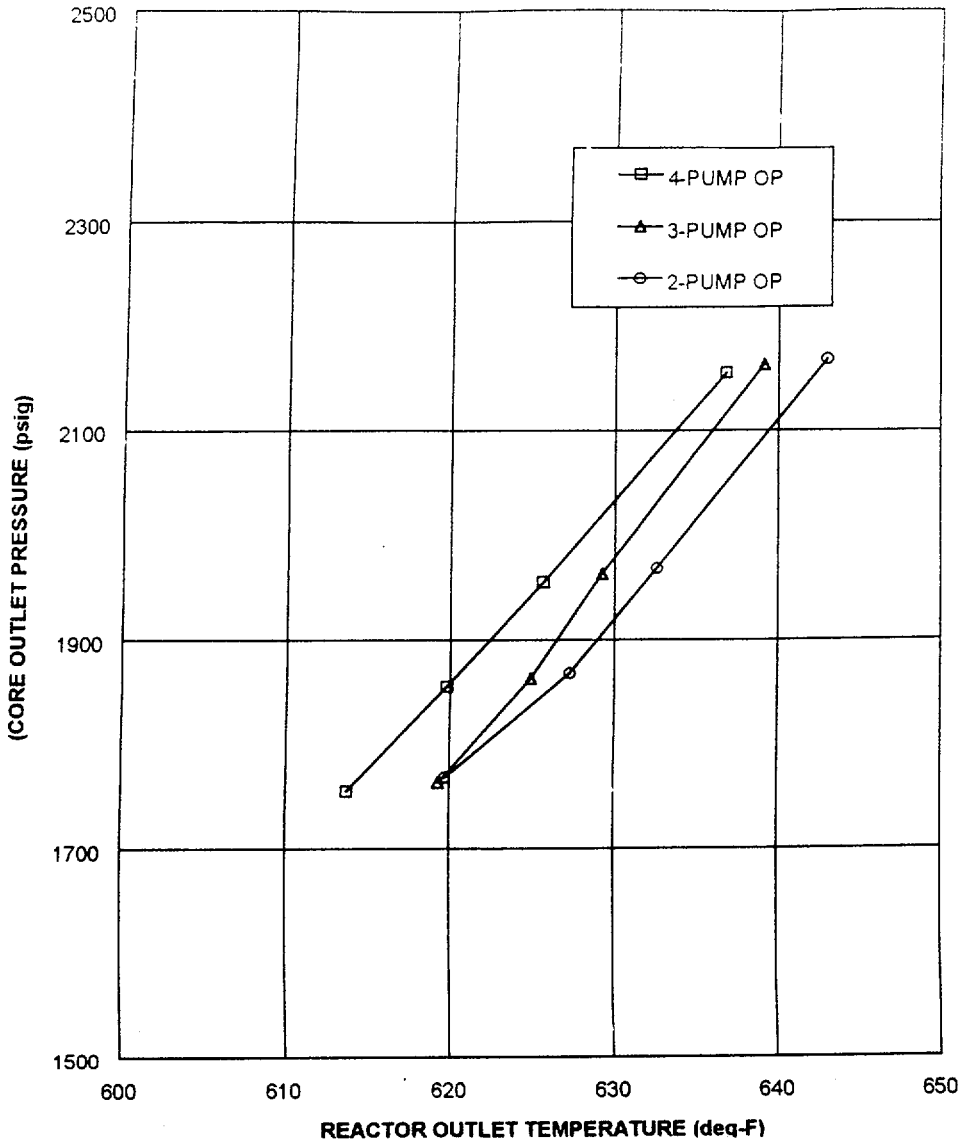
| Remove | Insert |
|---|---|
| 2-4a (Figure 2.1-1) | 2-4a (Figure 2.1-1) |
| 2-4c (Figure2.1-3) | 2-4c (Figure 2.1-3) |




CORE PROTECTION SAFETY LIMIT
TMI-1
Figure 2.1-1



Amendment No. ~~50, 142, 167~~, 214




| RC Pumps | Reactor Coolant Flow (lbs/hr) | Power | Pumps Operating (Type of Limit) |
|---|---|---|---|
| 4 | $134.02 \times 10^6$ | 112% | Four Pumps (DNBR Limit) |
| 3 | See COLR | See COLR | Three Pumps (Quality Limit) |
| 2 | See COLR | See COLR | One Pump in Each Loop (Quality Limit) |

CORE PROTECTION SAFETY BASES
TMI-1
FIGURE 2.1-3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR, INC.

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

## 1.0 INTRODUCTION

By letter dated December 3, 1998, as supplemented by letters dated March 26, April 16, May 7, May 21, June 4, June 15, and June 29, 1999 (References 1, 3, 4, 5, 7, 8, 9 and 10), GPU Nuclear, Inc. (the licensee) requested changes to Figure 2.1-1, "Core Protection Safety Limit," and Figure 2.1-3, "Core Protection Safety Bases," in the Three Mile Island Nuclear Station, Unit 1 (TMI-1) Technical Specifications (TSs). The revised figures reflect a decrease in reactor coolant system (RCS) flow resulting from analyses to allow operation of TMI-1 with an increased level of steam generator (SG) tubes plugged. Currently, TMI-1 has 1,300 tubes plugged in SG "A" (approximately 8 percent) and 395 tubes plugged in SG "B" (approximately 2.5 percent), resulting in 5.5 percent of the tubes removed from service. The proposed changes would allow a maximum average tube plugging of 20 percent in both SGs, a maximum tube plugging of 25 percent in any one SG, and a maximum plugging asymmetry of 15 percent between the two SGs. These limits would include actual plugged tubes and equivalent plugged tubes resulting from other repairs such as sleeving. The licensee also provided information pursuant to 10 CFR 50.46 reporting requirements by letters dated February 5 and May 12, 1999 (References 2 and 6), that was used in the staff's evaluation of the requested change. These letters and the supplements provided additional information and did not affect the staff's proposed no significant hazards determination (63 FR 71967) published in the Federal Register on December 30, 1998.

SG tube plugging decreases RCS flow (due to increased flow resistance), reduces RCS inventory, and decreases primary-to-secondary heat transfer. The thermo-hydraulic effects of increasing the tube plugging limits on the Updated Final Safety Analysis Report (UFSAR) Chapter 14 accident analyses and on setpoint determinations are evaluated below. The staff has also evaluated the licensee's submittal regarding the proposed changes as they relate to the integrity of the reactor coolant pressure boundary and the characteristic curves for the emergency feedwater (EFW) pump.

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The licensee's request also addressed the impact of 20 percent SG tube plugging on the mass and energy releases to the reactor building (containment) used in the current licensing bases for the calculation of the maximum reactor building pressure. The impact on the equipment qualification temperature and pressure profile was also addressed. The licensee's assessment of 20 percent SG tube plugging on the mass and energy releases developed for containment performance licensing analyses, for both loss-of-coolant accidents (LOCAs) and main steam line breaks (MSLBs), are also evaluated below.

The licensee also evaluated the effect of the proposed amendment on the radiological consequences for two design basis accidents, SG tube rupture and MSLB, to determine the potential effects of additional tube plugging on the calculated radiological consequences analyzed in Chapter 14 of the TMI-1 UFSAR. The licensee concluded that the effects of the proposed amendment were bounded by the existing analysis. The staff's evaluation of these radiological consequences is also addressed below.

## 2.0 EVALUATION

### 2.1 Revised accident analyses

TS 2.1.1 specifies the combination of reactor system pressure and coolant temperature which must not be exceeded as shown in Figure 2.1-1. The curve presented in Figure 2.1-1 represents the conditions at which the minimum predicted departure from nucleate boiling ratio (DNBR) does not violate the DNBR limit of 1.18 obtained with the BWC critical heat flux correlation for the limiting combination of thermal power and number of operating reactor coolant pumps (RCPs). This curve is the most restrictive of all possible RCP-maximum thermal power combinations shown in TS Figure 2.1-3. TS 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.

The core protection safety limits were reanalyzed with 20 percent SG plugging based on a reduced minimum RCS design flow of 102 percent of 352,000 gpm. The 102 percent value includes the effects of 20 percent SG tube plugging, asymmetries in tube plugging, and flow measurement uncertainty. The DNB hot channel analyses were performed with the approved VIPRE code for various flow, inlet temperature and system pressures for each particular pump condition. The results indicate that although the core protection safety limit and its bases given in TS Figures 2.1-1 and 2.1-3 become slightly more restrictive than those in the current TMI-1 TS, all of the current RPS trip setpoints remain applicable.

The licensee evaluated the TMI-1 UFSAR Chapter 14 safety analyses to determine the effect of 20 percent average SG tube plugging on the compliance of the various transients and accidents with the acceptance criteria. The calculated change in current operating parameters (Cycle 12) due to 20 percent SG tube plugging is a decrease in RCS flow (percent design flow) from 110 percent to 106.4 percent, an increase in $T_{hot}$ from 602.2 °F to 603.45 °F, and a decrease in $T_{cold}$ from 556.7 °F to 555.45 °F. There is also a reduced steam temperature calculated as a result of the proposed change. The operating RCS pressure remains unchanged.

The reactivity initiated events evaluated were the uncompensated operating reactivity change, the startup accident, the rod withdrawal at rated power, the moderator dilution accident, the cold water accident, the stuck-out, stuck-in, or dropped control rod accident, and the rod ejection accident. The results indicate that for those events in which the RCS pressure boundary must not be challenged and fuel integrity must be maintained, the RCS pressure remains below the acceptance criterion in the SRP for RCS pressure during these events (110 percent of design pressure) and the minimum DNBR remains above the limiting value. In addition, the amount of fuel damage in the rod ejection accident remains unchanged with the increased SG tube plugging. Therefore, the results of the reactivity initiated events meet all of the acceptance criteria with a 20 percent average SG tube plugging. The staff reviewed the licensee's evaluations for these events and concurs with the conclusions.

In addition, the licensee evaluated the loss of coolant flow (LOCF), loss of all alternating current (AC) power, steam line break, steam generator tube failure, loss-of-feedwater (LOFW), anticipated transient without scram (ATWS), small-break loss-of-coolant accident (SBLOCA), and large-break loss-of-coolant accident (LBLOCA) with respect to the effects of the proposed changes to the SG tube plugging limit. The licensee concluded that the steam line break and steam generator tube failure events were bounded by the current analyses of record and that the ATWS event retained large margins to the acceptance criteria. The staff reviewed the licensee's evaluations for these events and concurs with the licensee's conclusions. The licensee further concluded that the LOCF, LOFW, loss of all AC power, SBLOCA, and LBLOCA events were impacted by the proposed changes. Accordingly, the licensee reanalyzed these events with initial conditions and assumptions consistent with the proposed changes. Conservative values for instrument errors and system response times were used in these analyses in order to bound expected values and ensure bounding results. A discussion of each of these events follows.

The LOCF events were identified as the most limiting with respect to DNBR. These events were reanalyzed using the VIPRE-01 MOD02 computer code and the BWC correlation, both of which have been approved by the staff (Reference 11). The licensee confirmed that the operational parameter ranges for the LOCF events analyzed were within the range of the restrictions/limitations in the staff safety evaluation report (SER) for the VIPRE-01 code with the BWC correlation. Three LOCF scenarios were analyzed. These were a coastdown of all four RCPs, a coastdown of a single RCP, and a locked rotor. In order to analyze these events, the licensee calculated RCS flow rates for four pumps in operation and for three pumps in operation. These calculations included the effects of 20 percent SG tube plugging, asymmetries in tube plugging, and flow measurement uncertainty. In addition, in order to conservatively bound future fuel assembly design effects, these calculations were performed for an Mk-B10 core with fuel debris filter plates. The resulting flow rates were 102 percent of design flow for the four pumps in operation case and 74.5 percent of design flow for the three pumps in operation case. For analysis of the coastdown events, the licensee also investigated the effects of the proposed changes on RCP coastdown characteristics. RCP coastdown was analyzed for tube plugging levels of 0, 10, 20, and 30 percent and with asymmetric plugging of 0 percent in one SG and 30 percent in the other SG. The most limiting normalized flow rate was used in the analyses. The four pump and single pump coastdown events were mitigated by the power/pump monitors trip and the flux flow trip, respectively. These events resulted in minimum DNBR values of 1.669 and 1.484, respectively. Both values were higher than the limit for the BWC correlation of 1.18.

The locked rotor event was analyzed using a single state point analysis to determine the minimum DNBR behavior of the core using a minimum locked rotor transient flow fraction at the initial full power level. This event was mitigated by the flux flow trip and resulted in a minimum DNBR value of 1.276. This value is higher than the limit for the locked rotor event of 1.0.

Based on the above, the staff finds the licensee's LOCF analyses to be acceptable.

The LOFW and loss of all AC power events were reanalyzed using the RETRAN-02 MOD005.2 computer code and a TMI-1 RETRAN model, both of which have been approved by the staff (Reference 2). The licensee confirmed that the reanalysis for each of these transients: (1) utilized modeling options consistent with the approved TMI-1 RETRAN model, and (2) met the restrictions outlined in the RETRAN-02 SER. Two scenarios were analyzed for the LOFW event. Both were conservatively assumed to be initiated by closure of the feedwater control valves. One scenario was performed with assumptions to maximize the resulting pressurizer liquid level while the other was performed with assumptions to maximize the resulting RCS pressure. For the peak pressurizer liquid level analysis, it was assumed that the pressurizer power-operated relief valve (PORV) and the pressurizer spray system were available for pressure control. This assumption reduces the resulting pressure thereby allowing more insurge into the pressurizer. This assumption is conservative with respect to the peak pressurizer level. For the peak pressure analysis it was conservatively assumed that the pressurizer PORV and the pressurizer spray system were not available. In this case, the pressurizer safety valves were relied on for overpressure protection. The pressurizer safety valves were assumed to lift at 2,575 psig which includes 3 percent for tolerance. In addition to the above assumptions, the licensee determined that asymmetric tube plugging will show the same results as 20 percent average tube plugging for both scenarios. Therefore, the analyses were performed with an assumption of 20 percent tube plugging in both SGs. The LOFW analyses were mitigated by the high RCS pressure trip function of the RPS and the emergency feedwater system (EFWS). Assuming a single failure, the EFWS must be able to deliver the required flow to the SGs using two of the three pumps and control valves. All three of the possible two-pump combinations (i.e, two motor-driven pumps (MDP), the turbine-driven pump (TDP) and MDP "A", and the TDP and MDP "B") must be capable of delivering the required flow to the SGs. The EFWS was assumed to actuate on a low SG level signal with actual flow initiated after a 43-second delay.

Initially, the licensee used overly conservative assumptions in modeling the EFWS. This resulted in a low EFWS flow rate which led to the pressurizer becoming water solid in the peak level analysis. As a result of a staff concern with the water solid condition, the licensee performed a revised peak level analysis with higher EFWS flow rates (linear interpolation between 550 gpm to the SGs at 1,065 psia and 500 gpm to the SGs at 1,090 psia). In the revised analysis, the pressurizer did not become water solid. In addition, at the time that peak pressurizer level occurs, the RCS pressure is decreasing and is well below the opening setpoints of the PORV and the pressurizer safety valves.

In order to ensure that the surveillance program periodically demonstrates the capacity of the EFWS consistent with the assumptions in the revised analysis, the licensee derived new acceptance criteria for the EFWS surveillance tests. To derive the new acceptance criteria, the licensee developed a more realistic and detailed model of the EFWS. The model was developed with the RELAP5 computer code and included all of the piping and fittings from the condensate storage tanks to the once-through steam generators, the pump recirculation lines,

and pump bearing cooling lines. The model was benchmarked against data taken from several surveillance tests. Use of the new model allowed the licensee to remove overly conservative assumptions that were used in the original model.

The new EFWS model was utilized, in conjunction with system flow assumptions from the revised RETRAN-02 peak level analysis, to derive individual EFWS pump flow and head requirements for the EFWS surveillance tests. The licensee proposed a surveillance test which will be used to test each pump. As required by the proposed surveillance test, each pump will be individually tested with a control valve throttled until the required flow is achieved. Pump discharge pressure will then be measured at locations down-stream of the pump recirculation and bearing/seal cooling water branch lines. This pressure is then compared with pump suction pressure to produce a measured total head. As long as the measured EFWS flow and total head are equal to or greater than the required values (required values equal the calculated values using the RELAP5 model plus the test instrumentation uncertainties), the surveillance test will be deemed acceptable. Using the RELAP5 model, the licensee calculated the total head as 2,662 feet and the flow rate for each pump as 275 gpm. These values will be adjusted for instrumentation uncertainties and the resulting head and flow will be included in the surveillance test procedures.

The staff reviewed the proposed methods for evaluating the EFWS performance and establishing surveillance test acceptance criteria using the RELAP5 model and finds them to be acceptable. The analyses for the LOFW events, as described above, demonstrated continued compliance with the acceptance criteria for these events by showing that: (1) the pressurizer does not become water solid in the limiting level analysis, (2) the maximum RCS pressure reached in the peak pressure analysis (i.e., 2,669.4 psia) was less than the limit of 2,750 psig, and (3) the thermal power remained below 112 percent. The staff reviewed these analyses and, because the licensee used NRC-approved codes in performing these analyses and the acceptance criteria are met, finds them acceptable. The staff also finds the development of the theoretical pump performance curves and the acceptance criteria for the surveillance tests acceptable. However, the licensee has not performed the surveillance tests for the EFWS pumps to demonstrate that the actual performance of the pumps meet the newly established acceptance criteria. Therefore, acceptability of the proposed changes is contingent upon the licensee demonstrating that each of the EFWS pumps can deliver the required flow rate (275 gpm plus test instrumentation uncertainties) and that the developed head at this flow rate is at least that calculated in the licensee's evaluation of the EFWS (2,662 feet plus test instrumentation uncertainties).

The loss of all AC power (i.e., station blackout (SBO)) transient was reanalyzed. This event was initiated by the loss of all unit power except the unit batteries. The loss of all AC power resulted in the trip of the reactor, the turbine stop valves, the RCPs, and the main feedwater pumps. This event was mitigated by the EFW and natural circulation cooling with the main steam safety valves (MSSVs) relieving the excess steam from the secondary system. The MSSVs were conservatively assumed to lift at the setpoint plus 3 percent to account for setpoint drift. This is consistent with the as-found limits used when testing the MSSVs. The motor-driven EFWS pumps were assumed to not operate as a result of the loss of all AC power. Therefore, the TDP was the only means of providing the required flow. The TDP was assumed to deliver a total of 350 gpm flow to the SGs at 1,065 psia and 330 gpm to the SGs at 1,090 psia to provide decay heat removal capability (linear interpolation was used for intermediate values). The analysis assumptions for this event were the same as the LOFW analysis for

peak RCS pressure with the following exceptions: the EFWS was actuated by the loss of RCPs signal, and feedwater was assumed to terminate in 2 seconds. The loss of all AC power analysis demonstrated the natural circulation capability of the plant with 20 percent average SG tube plugging. In addition, the licensee used NRC-approved codes in performing this analysis, which demonstrated continued compliance with the acceptance criteria for the event. Accordingly, the staff finds the licensee's analysis to be acceptable.

An evaluation of the TDP with respect to the loss of all AC power analysis was performed in a manner similar to that used in the evaluation of the EFWS with respect to the LOFW analysis. For the loss of all AC power event, the TDP is required to deliver 350 gpm. At this flow rate, the licensee calculated that a total dynamic head of 2,540 feet is required. The licensee evaluated the TDP performance with respect to TDP acceptance criteria for both the LOFW event and the loss of all AC power event. The licensee determined that acceptance criteria based on the LOFW event are more limiting and; therefore, the acceptance criteria for the TDP will be established based on that event. The staff agrees that the LOFW event is bounding and finds the licensee's evaluation to be acceptable.

The LOCA events were reanalyzed using the RELAP5/MOD2-B&W, REFLOD3B, and BEACH computer codes; all of which have been approved by the staff (Reference 13). The licensee has confirmed that the analyses were performed in compliance with the evaluation model methods and the limitations and restrictions stated in the staff's SERs. The LOCA analyses were performed to cover the increased SG tube plugging limit with the proposed asymmetry, and previously identified non-conservatisms in the modeling of the RCPs and the use of fuel assembly mixing-vane grids. The non-conservative modeling of the RCP was reported by the licensee to the NRC by letter dated February 5, 1999. The non-conservative assumption on the use of fuel assembly mixing-vane grids was reported by the licensee to the NRC by letter dated June 4, 1999. Tube plugging asymmetries were investigated and the most limiting configuration of 15 percent tubes plugged in the intact loop SG and 25 percent tubes plugged in the broken loop SG was utilized in the analyses. The analyses were performed at a power level of 2,827 MWt (102 percent of 2,772 MWt) and 102 percent of design flow. The limiting LBLOCA was determined to be the double-ended guillotine break at the cold-leg pump discharge with a discharge coefficient of 1.0 and minimum emergency core cooling system (ECCS) injection. This configuration was therefore analyzed for the LBLOCA. The limiting SBLOCA was determined to be a 0.05 break in the bottom of the cold-leg piping between the reactor vessel inlet nozzle and the high-pressure injection (HPI) nozzle. This configuration was therefore analyzed for the SBLOCA. Two other cases were also analyzed. These were an HPI line break and core flood tank line break. The failure of one emergency diesel generator was assumed in all analyses. This minimized ECCS injection by reducing the available injection to one HPI pump and one low-pressure injection pump. The LOCA analyses were performed for the Mk-B9 fuel design. The licensee reported the resulting peak cladding temperatures (PCTs) as 2,104 °F for the LBLOCA, 1,412 °F for the SBLOCA, 1,297 °F for the HPI line break, and 715 °F for the core flood tank line break. The licensee further stated that the analyses demonstrated compliance with the five acceptance criteria of 10 CFR 50.46(b).

In the February 5, 1999, report on non-conservative RCP modeling, the licensee reported that the previous (i.e., prior to this amendment request) TMI-1 LOCA analyses for the Mk-B9 fuel design used a conservatively low-initial pressurizer liquid inventory to account for pressurizer level uncertainty. This assumption was reported to have resulted in PCTs approximately 80 °F

to 100 °F higher than would have been produced with nominal pressurizer inventory. In the LOCA analyses for this amendment request, the licensee used nominal pressurizer inventory. In the letter dated June 15, 1999, the licensee provided further explanation of the effect of the pressurizer level uncertainty. The licensee explained that the effect at the elevation with the maximum PCT (the 7.779-foot evaluation) is insignificant. The licensee reported that the effect is only predominant at the core inlet elevation. Furthermore, the licensee noted that the limiting PCT at the 7.779-foot elevation is based on bounding LOCA linear heat rate (LHR) limit of 17.3 kW/ft. The TMI-1 current operating Cycle 12 and subsequent Cycle 13 operating limits are based on a LOCA LHR limit of 16.8 kW/ft at the 7.779-foot elevation. The calculated TMI-1 PCT at the 7.779-foot elevation based on 16.8 kW/ft is only 1,964 °F. The staff has considered the impact of pressurizer level uncertainty in its evaluation of the licensee's LOCA analyses. With the licensee's estimated impact of pressurizer inventory uncertainty added to the PCTs reported in the licensee's submittals for this amendment request, the resulting TMI-1 PCTs would still be under the 10 CFR 50.46(b)(1) limit of 2,200 °F.

TMI-1 Cycle 13 will load Mk-B10P and Mk-B8 fuel designs. The revised corresponding LHRs (including the limit to 16.8 kW/ft for the 7.779-foot elevation) will be used in determination of core operating limits for the Cycle 13 reload. The revised LOCA LHRs will be placed in the TMI-1 Core Operating Limits Reports prior to Cycle 13 startup.

Because (1) the licensee demonstrated that continued compliance with the 10 CFR 50.46(b) acceptance criteria for LOCA events will be maintained and (2) PCT will remain below 2200 °F taking into account pressurizer level uncertainty, and based on the above discussion of LOCA events, the staff finds the licensee's LOCA analyses to be acceptable.

2.2 Structural Integrity of RCS Components

As part of its justification to support the proposed plant operation, the licensee performed an evaluation of the structural integrity for the RCS and components. The key design parameters (i.e., RCS pressure, hot leg temperature, cold leg temperature, SG steam pressure, and SG outlet temperature) are provided in Table 1 on page 19 of Reference 1 for the current existing analysis and the proposed operation at the full-power condition. The licensee performed the evaluation by comparing the key inputs in the existing stress report with the proposed operating condition for the RCS coolant piping, components and their supports. The components include the reactor vessel and internals, the attached nozzles, the pressurizer, surge line (stratification), pressurizer spray nozzles, the SGs, RCPs and fuel assemblies. The licensee evaluated the effects of the increased hot leg temperature on the RCPs, seals, valves, welds, bolting, and other components mentioned above. The evaluation conservatively considered a 2.1 °F increase in $T_{hot,}$ which represents only a 0.4 percent change in the temperature differential from an initial ambient system temperature of 70 °F to the revised hot leg temperature. The licensee indicated that the evaluation of the RCS system for normal operating conditions, anticipated transients, and upset conditions shows that the through-wall thermal gradient and thermal expansion terms remain bounding for the revised design condition. In addition, the licensee noted that the design-basis stress analyses for TMI-1 are based on a 608 °F hot leg temperature which bounds the proposed hot leg temperature of 604 °F. On the basis of its

review, the licensee concluded that the increase in $T_{hot}$ due to 20 percent tube plugging does not impact the design-basis analyses for the components evaluated, with regard to stresses and fatigue usage factors. Because the licensee's analysis is bounding, the staff agrees with the licensee's conclusion.

The licensee evaluated the pressurizer for limiting locations from a structural standpoint, including the surge line, the spray nozzle, and the shell. Since the RCS pressure does not change for the proposed condition, the increase in the hot leg temperature will reduce the top-to-bottom temperature difference in the pressurizer and the thermal stratification in the surge line, thus reducing the fatigue usage factor and stresses in the pressurizer and the surge nozzle. The pressurizer spray nozzle was evaluated for the effect of a reduced cold leg temperature. The effect was determined by the licensee to be minimal since a minimum bypass flow exists through the spray line which keeps the line and the nozzle at the cold leg temperature. Therefore, the licensee concluded that the changes in hot leg and cold leg temperatures due to a 20 percent plugging of once-through steam generator (OTSG) tubes has no effect on the bounding design-basis stress and fatigue analyses for TMI-1. Because the licensee's analysis is bounding, the staff agrees with the licensee's conclusion.

The licensee also evaluated the effect of the $T_{cold}$ reduction on the hydraulic forces applied to the components. The reduction in $T_{cold}$ increases the fluid density and thus, increases the loading on the components during a postulated LOCA. The reduced cold leg temperature can also affect the magnitude of the decompression wave during a LOCA, based on the difference between the operating pressure and the saturation pressure, thus increasing loads. The increase in pressure difference is about 1.6 percent , which is considered by the licensee to have negligible effects on the LOCA loads. The increase in density (about 0.4 percent) is also considered small and insignificant. Because the increases in pressure and density are small, i.e., are not significant within the accuracy of the analysis, the staff agrees with the licensee's conclusion that a 20 percent SG tube plugging (SGTP) would have a negligible effect on the LOCA hydraulic forces. Accordingly, the staff considers the existing analysis adequate.

The licensee evaluated the SG components for a higher primary coolant fluid density (or the reduction of cold leg temperature), a decrease in steam temperature, and the potential for flow-induced vibration. The licensee's review identified certain components (i.e., tubes, secondary shell and attached nozzles) for which the loads (i.e., effects of fluid flow changes on tube vibration, the changes in transients, etc.) are affected as a result of the reduction in the cold leg temperature. The licensee indicated that changes to normal operating loads due to the reduction in the cold leg temperature are insignificant in comparison with the design basis LOCA loads resulting from an MSLB. The licensee evaluated the potential for flow-induced vibration, especially in the upper region of the SG where superheated steam flows radially and normal to the tubes. As shown in a table on page 52 of Reference 3, the changes in the calculated dynamic pressure over the proposed temperature range are negligible. Therefore, the potential for flow-induced vibration will not increase for the proposed condition. The licensee also showed in Table 1 of Reference 1 that the steam superheat for the proposed 20 percent SGTP condition is 44 °F, which represents a 6 °F reduction in the steam temperature from the current steam superheat of 50 °F, but remains above the design basis value of 35 °F. The licensee concluded that a 20 percent SGTP will not adversely affect the tube fluid-elastic stability ratio and stresses and fatigue usage factors, and that the components evaluated will remain within the allowable limits for the proposed condition. Because the licensee's analysis is bounding, the staff agrees with this conclusion.

Based on its review of the information provided by the licensee, as set forth above, the staff finds that the proposed plant operation with up to 20 percent average level of SGTP (involving a reduced RCS flow rate, a reduced RCS cold leg temperature, an increased hot leg temperature, and a reduced superheated steam temperature) have no adverse impact on the structural and pressure boundary integrity of the RCS piping, components and their supports, and is, therefore, acceptable.

2.3 Evaluation of Mass and Energy Releases for Containment Performance

The staff has evaluated the licensee's assessment of 20 percent OTSG tube plugging on the mass and energy releases developed for containment performance licensing analyses for both LOCAs and MSLBs. The staff also evaluated the licensee's assessment of the impact of changes to the mass and energy releases on the current licensing bases for the reactor building pressure response analysis and the equipment qualification temperature and pressure analysis.

The RCS parameters of importance to the mass and energy releases that are impacted by the 20 percent tube plugging are:

- decrease in the RCS volume of 427 $ft^3$ (3.7 percent decrease)
- increase in RCS $T_{hot}$ of 1.25 °F (from 602.2 °F to 603.45 °F)
- decrease in RCS $T_{cold}$ of 1.25 °F (from 556.7 °F to 555.45 °F)

The core thermal power (2,568 MWt), RCS pressure (2,170 psig) and RCS $T_{ave}$ (579.5 °F) are unchanged from the current licensing bases.

The OTSG parameters of importance to the mass and energy releases that are impacted by the 20 percent tube plugging are:

- increase in secondary side mass of 5,200 lbm (13 percent increase at 25 percent OTSG tube plugging)
- decrease in OTSG heat transfer area (15.5 percent decrease)
- decrease in OTSG outlet super heat of 6 °F (from 50 °F to 44 °F)

The OTSG secondary side pressure (925 psia) is unchanged from the current licensing bases.

The licensing bases mass and energy releases from LOCAs were obtained with the FLASH and PIRT computer programs. FLASH computes the blowdown portion of the LOCA and PIRT computes the post-blowdown period. The reactor building (containment) pressure response was obtained with the CONTEMPT computer program.

The RCS internal energy (stored in the reactor coolant) in the licensing bases analyses is about 3.7 percent higher than the RCS internal energy with 20 percent OTSG tube plugging. The coolant density change, due to the 1.25 °F decrease in $T_{cold}$, is less than 0.2 percent and, therefore, the LOCA blowdown rates will not be significantly different from the licensing bases analyses. However, OTSG tube plugging will increase the RCS loop resistance and would result in a reduced mass flowrate into the reactor building. The licensee concluded that the licensing bases mass and energy releases from LOCAs used in both the peak containment

pressure analysis and the containment equipment qualification temperature and pressure analysis remain conservative.

The licensing bases for TMI-1 showed the reactor building pressure response to be bounded by the large break LOCA and not the MSLB. The MSLB licensing analysis assumed a conservatively large initial OTSG secondary side inventory, 55,000 lbm. With 20 percent OTSG tube plugging, the actual secondary side inventory is estimated to be less than 45,000 lbm. With 20 percent OTSG tube plugging, the heat transfer area is also reduced by 15.5 percent and the OTSG super heat is reduced by 6 °F. The licensing bases mass and energy releases from the MSLB used in the peak containment pressure analysis remain conservative and the licensee concluded that the reactor building response will continue to be bounded by the large break LOCA.

Because there would be a reduced mass flowrate into the reactor building during a LOCA, and the estimated secondary side inventory is less than assumed in the analysis, the staff agrees with the licensee's conclusion that the current licensing analyses for LOCA and MSLB mass and energy releases developed for containment performance remain conservative with up to 20 percent OTSG tube plugging. Accordingly, the current reactor building pressure licensing analysis and the current equipment qualification temperature and pressure profile licensing analysis also remain conservative.

2.4 Radiological Consequences

The proposed TS change request would allow a maximum OTSG tube plugging limit of 25 percent in any one OTSG and a maximum plugging asymmetry of 15 percent between the two OTSGs. The current maximum allowable tube plugging limit is a total of 6.4 percent. The additional OTSG tube plugging will decrease RCS flow due to increased flow resistance, will reduce RCS inventory, and will decrease primary-to-secondary heat transfer. The licensee evaluated two design basis accidents, steam generator tube rupture (SGTR) and MSLB, to determine the potential effects of additional tube plugging on the calculated radiological consequences analyzed in Chapter 14 of the TMI-1 UFSAR.

The steamline break and SGTR are assumed to result in the release of the fission products contained in primary-to-secondary leakage and relatively small amounts of that contained in the secondary system prior to the accident. In its evaluation, the licensee assumed in the UFSAR that the unit has been operating with an RCS dose equivalent iodine-131 (DEI-131) activity specified in the TS (0.35 μCi/gram) prior to the accident and that at the onset of the accident, the release rate of iodine from the fuel rods to the RCS is assumed to spike by a factor of 500.

In its application, the licensee stated that the additional OTSG tube plugging would reduce primary-to-secondary leakage from the existing primary-to-secondary leakage value assumed in the UFSAR since the OTSG tube axial tensile loads and the primary-to-secondary heat transfer are reduced. Therefore, the licensee concluded that the radiological consequences analyzed in the UFSAR are still bounding for meeting the relevant dose criteria. Based on the licensee's reasoning, the staff reviewed the licensee's analysis and agrees with the licensee's conclusion that there will be no effects on the radiological consequence analyzed in the UFSAR as a result of the additional OTSG tube plugging and that the parameters used in the SGTR and MSLB analyses are still bounding for meeting the relevant dose criteria. Therefore, the staff finds that the radiological consequences of the proposed TS change are acceptable.

Based on the above, the staff finds the proposed changes to Figure 2.1-1, "Core Protection Safety Limit," and Figure 2.1-3, "Core Protection Safety Bases," in the TMI-1 TSs to be acceptable. However, these findings are contingent upon the licensee demonstrating that the emergency feedwater system (EFWS) pumps can deliver the flow rate assumed in the analyses and the developed head at this flow rate is at least that calculated in the licensee's evaluation of the EFWS (2,662 feet plus instrument uncertainties).

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 71967). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 REFERENCES

1. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Technical Specification Change Request No. 279 - Core Protection Safety Limit," December 3, 1998.

2. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, 10 CFR 50.46 Report on Significant PCT Change in ECCS Analyses," February 5, 1999.

3. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 Core Protection Safety Limit," March 26, 1999.

4. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 Core Protection Safety Limit," April 16, 1999.

5. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 Core Protection Safety Limit," May 7, 1999.

6. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, 10 CFR 50.46 Final Report on Significant PCT Change in ECCS Analyses," May 12, 1999.

7. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 - Core Protection Safety Limit," May 21, 1999.

8. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 - Core Protection Safety Limit," June 4, 1999.

9. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 - Core Protection Safety Limit," June 15, 1999.

10. Letter from J. W. Langenbach, GPU Nuclear, Inc., to USNRC, "Three Mile Island Nuclear Station, Unit 1 (TMI-1), Operating License No. DPR-50, Docket No. 50-289, Additional Information - Technical Specification Change Request No. 279 - Core Protection Safety Limit," June 29, 1999.

11. Letter from J. Norris, USNRC, to J. Knubel, GPU Nuclear Corporation, "Review of Topical Report TR-087, Entitled TMI-1 Core Thermal-Hydraulic Methodology Using VIPRE-01 Computer Code for Three Mile Island Unit 1," dated December 19, 1966.

12. Letter from B. Buckley, USNRC, to J. Knubel, GPU Nuclear Corporation, "Review of Topical Report TR-078, Revision O, Entitled TMI-1 Transient Analysis Using the RETRAN Computer Code," dated February 10, 1997.

13. Letter from J. Lyons, USNRC, to J. H. Taylor, Framatome Technologies Incorporated, "Acceptance for Referencing of Topical Report BAW-10192-P, BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," dated February 18, 1997.

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