GPU NUCLEAR

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June 23, 1997
6710-97-2242

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
GPU Nuclear Response to Inspection Report (IR) 96-201

A design team inspection was performed at TMI-1 by the Special Inspection Branch of Nuclear Reactor Regulation (NRR) and its contractor Stone & Webster Engineering Corporation (SWEC) during the period November 12, 1996 through January 10, 1997. The inspection team performed a comprehensive, in-depth examination of the design and licensing basis documentation for the Makeup and Purification (MU&P) and Decay Heat Removal (DHR) Systems. The MU&P System includes high pressure injection (HPI) and the DHR System includes low pressure injection (LPI). A public exit meeting was held at TMI on January 30, 1997.

By letter dated April 15, 1997 NRC provided Inspection Report (IR) 96-201, "Three Mile Island - Unit 1, Design Inspection." The cover letter stated that the team noted that the design documents for the reviewed systems appropriately implemented the intent of the design and licensing basis except for the specific cases identified in the report. Appendix A of the report listed the open items, which were categorized as either unresolved items (URIs) or inspection follow-up items (IFIs). The letter requested that GPU Nuclear provide a schedule for completion of our corrective actions for the open items within 60 days. GPU Nuclear's request for an additional day was granted by Mr. Robert M. Gallo, Chief Special Inspection Branch by telephone on June 20, 1997. Enclosed is the GPU Nuclear response to that request.

The NRC's letter of April 25, 1997 identified those open items from IR 96-201 that were considered to be potential violations and needed to be discussed in a predecisional enforcement conference. During the conference which was held on May 22, 1997, GPU Nuclear agreed with the NRC's assessment of many of the issues that were raised. In our presentation we provided some additional perspective on the issues raised in the report and related some of the important actions that are being taken to prevent a recurrence of those findings including: the implementation of a new process-based engineering organization, development of a new self assessment program, and the initiation of a new corrective action process (CAP), as well as many of the actions that are being taken to address the specific findings in IR 96-201.

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This letter provides an update on the corrective actions, including a current status and schedule for completion of the remaining items. The information provided in Attachment 1 does not significantly differ from the comments provided by GPU Nuclear at the predecisional enforcement conference except that after further discussion with the NRC staff our positions in response to the findings represented in OIs 96-201-14 and 96-201-18 have been revised. GPU Nuclear has the resources and is committed to completing these actions within the current schedule or as close to these dates as possible.

Sincerely,

RWKeaten

R. W. Keaten
Vice President and Director, Engineering

MRK
Attachment

cc: Administrator, NRC Region I
TMI Senior NRC Resident Inspector
TMI Senior NRC Project Manager

Attachment 1

GPU Nuclear Response to NRC Inspection Report (IR) 96-201

IR 96-201 included 30 Open Items (OIs) listed in Appendix A (OIs 96-201-01 through 96-201-28 including the three part OI 96-201-17A, B and C). For each of these OIs, this attachment gives the short caption for the finding from Appendix A, followed by A) a summary description of the finding, B) a discussion in response to the finding, C) the planned corrective actions, and D) GPU Nuclear's current schedule for completion of the remaining actions. Note that the caption for each of the OIs in Appendix A of the IR includes (in parentheses) a reference to the specific section in the IR where the OI is discussed.

IFI 96-201-01 "Letdown line Break in the Auxiliary Building (Section E1.2.2.2.a)"

A. Description of the finding:

The inspection team noted that a properly approved analysis of the consequences of a letdown line break in the Auxiliary Building (AB) was not available. The team referred this issue to the technical review branch in the Office of Nuclear Reactor Regulation (NRR) staff for review regarding the extent to which TMI-1 was required to consider the effects of a letdown line break in the auxiliary building. The staff review concluded that the TMI-1 licensing basis for pipe breaks includes the postulation of full diameter breaks in the letdown line between the containment penetration and the breakdown orifice as described in Appendix 14A to the FSAR. Therefore, the design of safety-related equipment in the affected areas should consider the conditions resulting from these breaks.

B. Discussion:

GPU Nuclear has been unable to retrieve the 1973 analysis, which is referred to in Updated FSAR Appendix 14A, Section 4.3, page 14A-11. In the absence of this analysis, GPU Nuclear is evaluating whether the consequences of this event are acceptable. Our evaluations to date indicate that the plant could be safely shut down if such an event occurred because:

1. With respect to pipe whip and jet impingement:

   a. The postulated letdown line break does not produce sufficient energy to cause the letdown line to whip. MU-V3 and its associated operator, power and control signals would survive the break because of 1) the relative location of this equipment to the postulated breaks, 2) the physical protection provided for the operator by an intervening platform, and 3) the large size of the pneumatic operator considering the break energy.

   b. Postulated breaks are not in locations that could affect other safety-related equipment by either pipe whip or jet impingement.

2. With respect to the environmental consequences of postulated breaks:

   a. Safety related equipment in the area, Equipment Qualification (EQ) zone 7, is qualified to survive the post Loss of Coolant Accident (LOCA) environment inside containment.

   b. The static pressurization analysis performed by the architect engineer, Gilbert Associates, Inc. (GAI) in 1973, while simple, yields a valid conclusion that pressurization in the area is very small as long as letdown is isolated in a short time (minutes) after the break.

   c. Based on operator actions, the event would be terminated rapidly enough to prevent an environment requiring qualification under 10CFR 50.49.

Evaluations to date show that the consequences of the letdown line break do not result in consequences that would cause fuel damage or off-site releases that exceed 10 CFR 100 limits. The statements in the FSAR have been substantiated by our reviews to date. Manual actions, and possibly automatic isolation signals used to protect the core during Small Break LOCAs would be effective during this event and the effect of these consequences on the Reactor Coolant System (RCS) would be bounded by previously analyzed cold leg breaks.

C. Corrective Actions:

1. Perform analysis of the enviro umental effects of a letdown line break between the containment and the block orifice.

2. Determine the effects of this environment on safety related equipment in the affected areas.

3. Revise the FSAR to reference the new analysis.

D. Schedule for Completion of Corrective Actions:

1. Analysis is scheduled for completion by October, 1997.

2. Evaluation of the effects on safety-related equipment will be completed by December 31, 1997. We expect that the results will be acceptable.

3. Revision of the FSAR will follow in the next FSAR update following completion of the analysis and evaluations.

IFI 96-201-02 "Evaluation of Simultaneous Start of MU&P Pump MU-P-1C and Suction and Discharge Valves (Section E1.2.2.2.b)"

A. Description of the Finding:

In the normal operating configuration of the Makeup and Purification System, Makeup Pump MU-P-1C is isolated from the Makeup Tank (MUT) by manual valves and isolated from the Borated Water Storage Tank (BWST) by Emergency Safeguards (ES) motor operated valve MU-V-14B. Makeup Pump MU-P-1C is not lined up to any water source until MU-V-14B opens. The inspection team was concerned that the effect on the pump due to a slow opening suction valve combined with a rapid start of the pump and a fast opening high pressure injection valve had not been analyzed.

B. Discussion:

As described in the inspection report, GPU Nuclear committed to reduce the design basis valve stroke time for MU-V-14A/B in the Surveillance Procedure (SP) 1300-3H,"IST of MU Pumps and Valves," from 22 seconds to 13 seconds and to analyze the effect on system performance of concurrent maximum suction valve stroke time, minimum pump startup time and minimum discharge valve stroke time.

1. The simultaneous start of Makeup Pump MU-P-1C with opening of MU-V-14B was tested during initial startup testing in 1974. MU-V-14B stroke time was recorded at 9 seconds during that test.

2. The valve stroke time of MU-V-14A/B (motor operated stop check valves) has remained 9 seconds or less as verified by a review of test records.

3. The initial startup test results are consistent with a reasonable expectation for the valve and pump combination. The minimum flow rate for a TMI-1 Makeup Pump is 40 gpm which can be met at less than full open valve positions. After 3 seconds of valve travel, MU-V-14B permits a flow area of approximately 27 sq. in. Using an equivalent pipe size for this flow area, Crane Handbook 410, "Flow of Fluids through valves, fittings, and pipe," shows approximately 100 gpm flowrate with a pressure drop near 1 psid. Therefore, simultaneous operation will provide sufficient flow to meet the minimum pump flow requirements which are significantly less than full ECCS flow.

4. MU-P-1C is presently operable.

5. If MU-V14B stroke time increased by 45% (from 9 seconds to 13 seconds), the pump would still have sufficient flow to meet the pump minimum flow requirements during a startup transient at less than full open valve position.

C. Corrective Actions:

1. Surveillance procedure 1300-3H, "IST of MU Pumps and Valves," was revised in Revision 46, effective March 26, 1997. Data sheet D now identifies the design basis stroke time for MU-V-14A/B as 13 seconds.

D. Schedule for Completion of Corrective Action:

No additional corrective action is required to resolve this issue.

IFI 96-201-03 "Evaluation of Gas Accumulation in Suction Piping for MU&P Pump MU-P-1C (Section E1.2.2.2.b)"

A. Description of the Finding:

MU-P-1C suction piping is not normally lined up to a source of water. The inspection team identified the potential for accumulation of non-condensables, such as hydrogen, released from the stagnant water in the suction line because of the physical configuration of the line. During the inspection, the vent valve in the high point of this pipe section was opened and it was verified that there was no gas accumulation in the suction pipe. The inspection team considered that a positive pump suction pressure was not necessarily an indication of absence of gas accumulation in the piping. (E1.2.2.2.b)

B. Discussion:

As described in the inspection report, GPU Nuclear has instituted periodic checks of the MU-P-1C suction pressure as an interim measure. An operator checks the MU-P-1C suction pressure once per day as required on the Primary Auxiliary Operator (AO) Log sheet. If the suction piping pressure is maintained at greater than 30 psig, no mechanism has been identified which could cause the accumulation of non-condensables in this pipe section.

C. Corrective Actions:

1. A long term resolution of this concern is being developed.

D. Schedule for Completion of Corrective Actions:

1. Action to determine a long term resolution is being tracked by Licensing Action Request (LAR) 97052.02 which is scheduled for completion prior to startup following the Cycle 12 Refueling Outage scheduled to begin in September 1997.

URI 96-201-04 "Adequacy of Makeup Tank Pressure/Level Curves (Section E1.2.2.2.c)"

A. Description of the Finding:

The NRC inspection team identified several non-conservative assumptions in the calculations that serve as the basis for the Makeup Tank (MUT) Net Positive Suction Head (NPSH) and gas entrainment curves.

B. Discussion:

The lower pressure/level curve on Figure 1 of Operating Procedure (OP) 1104-2, "Makeup and Purification System," is designed to ensure adequate Makeup Pump NPSH when taking suction from the MUT and is based on a limiting scenario where a Reactor Coolant System (RCS) break occurs on the normal makeup line. The operator starts the second Makeup Pump and opens MU-V-217 following a reactor trip in accordance with Abnormal Transient Procedure (ATP) 1210-1, "Reactor Trip." This scenario is limiting because of the high Makeup Pump flow rates and rapidly decreasing NPSH available. The challenge is over upon Emergency Safeguards (ES) actuation or operator action to open MU-V-14. The primary concern raised with calculation C-1101-211-5360-003, "Makeup Pump NPSH," involved the use of a 1600 psig backpressure versus a 0 psig backpressure at the break on the normal makeup line.

Since lower backpressure will increase the drawdown rate from the MUT, GPU Nuclear agreed with the inspection team on makeup line back pressure and acted expeditiously to review these concerns. Preliminary calculations and sensitivity analyses were used as part of the operability review which concluded that the system was operable but degraded based on the short period of time with degraded NPSH. Procedure changes were promptly initiated to place operational limitations on makeup flow with MU-V-217 open to preclude the potential for degraded NPSH.

Preliminary analyses to support the operability review has demonstrated that the Makeup Pumps would be in a degraded NPSH condition for less than thirty seconds. Makeup Pump performance is not required for core cooling during the period of degraded NPSH. It was determined that the effect of operation with the degraded NPSH for the limited time period would not prevent the pump from performing its design function following ES actuation. Therefore, when the suction valves from the BWST open, the system is restored to a fully functional condition and there is no adverse impact on nuclear safety.

The upper pressure/level curve on Figure 1 of OP 1104-2 is designed to prevent Makeup Pump gas entrainment when taking suction from the MUT during high pressure injection prior to the MU-V-14 valves opening. The inspection team raised concerns with several non-conservative design inputs in calculation C-1101-211-5310-047, "Makeup Tank Drawdown During LOCA." These included the omission of some Makeup Pump suction pipe and fittings and underestimating Makeup Pump flow, minimum Borated Water Storage Tank (BWST) level, BWST water temperature, and MUT useable volume.

GPU Nuclear agrees that these concerns result in minor non-conservative conclusions in the calculation. However it should be noted that the BWST temperature and MUT volume concerns have minimal impact and that the other observations are overcome by the overall conservatism of the calculation.

The gas entrainment analysis conservatively postulates a large break Loss of Coolant Accident (LOCA) and assumes full flow from both trains of Low Pressure Injection (LPI) and Building Spray (BS). In such a scenario the High Pressure Injection (HPI) System is not required for accident mitigation. Small and intermediate size breaks result in slower BWST/MUT drawdown rates. The slower drawdown rates afford the operator ample time to make a successful switchover of suction to the BWST. Gas entrainment of the Makeup Pumps is not a concern following switchover because of the large inventory of water available as a suction source. Therefore, the HPI System will perform its intended safety function when required to do so and there is no adverse impact on nuclear safety.

C. Corrective Actions:

1. Operating Procedure (OP) 1104-2, "Makeup & Purification System." Revision 107, effective January 30, 1997 and Abnormal Transient Procedure (ATP) 1210-01, "Reactor Trip," Revision 36, effective January 31, 1997 were revised to limit makeup flow to less than 500 gpm when taking manual action to open MU-V-217 following a reactor trip. This satisfies the vendor recommended NPSH requirements throughout the postulated Makeup line break scenario.

2. Calculations C-1101-211-5360-003, "Makeup Pump NPSH," Revision 1 and C-1101-211-5310-047, "Makeup Tank Drawdown During LOCA," Revision 0 are being revised to address the relevant non-conservative assumptions identified during the inspection.

3. Operating Procedure (OP) 1101-1, "Plant Limits and Precautions," will be revised to delete the irrelevant information about useable MUT volume.

4. The System Design Basis Document (SDBD) will be revised to clarify information on MUT volume.

5. Programmatic improvements in the calculation preparation and control processes are addressed in OI 96-201-28, Corrective Action C.4.

D. Schedule for Completion of Corrective Actions:

1. Revisions of calculations C-1101-211-5360-003, "Makeup Pump NPSH," Revision 1 and C-1101-211-5310-047, "Makeup Tank Drawdown During LOCA," Revision 0 are scheduled for completion by September 30, 1997.

2. The System Design Basis Document (SDBD) is scheduled to be updated by August 30, 1997.

3. Procedure 1101-1, "Plant Limits and Precautions," is scheduled to be revised by December 31, 1997.

URI 96-201-05 "Design Basis Valve Stroke Times in Surveillance Procedure (Section E1.2.2.2.d)"

A. Description of the Finding:

FSAR Table 14.2-14 states that the Emergency Core Cooling System (ECCS) delay time assumed in the Loss of Coolant Accident (LOCA) accident analysis is 35 seconds. The licensee stated that this delay time is composed of 1 second for instrumentation lag, 10 seconds for start of the emergency power source if offsite power is not available, and 24 seconds for system response (pump acceleration and valve stroke time). FSAR Section 6.1.3.1 states that the system is designed to be in full operation within 25 seconds after receiving an actuation signal, and Surveillance Procedure (SP) 1300-3H, "IST of MU Pumps and Valves," Revision 44, provides a design basis stroke time of 25 seconds for the makeup pump recirculation isolation valves MU-V-36&37 and the injection isolation valves MU-V-16A through D. The team observed that the 25 second startup delay time added to the 11 second delay of the actuation signal results in a total delay time of 36 seconds, which would be an unanalyzed condition as stated in the System Design Basis Document (SDBD). This change to the facility had not apparently been reviewed in accordance with 10 CFR 50.59. The licensee initiated a work request to revise the design basis stroke times in procedure 1300-3H and initiated a change to the FSAR.

B. Discussion:

The inspection identified documentation discrepancies between the FSAR, SDBD and SPs associated with the High Pressure Injection (HPI) System response time. The inconsistencies between these documents were all related to whether a 1 second delay for instrumentation lag was consistently applied. The FSAR states that the Makeup pumps and valves need to operate within 25 seconds. SP 1300-3H, "IST of MU Pumps and Valves," identifies a design basis stroke time for HPI valves of 25 seconds.

A 35 second HPI System response time is assumed in the LOCA analysis which verifies ECCS performance in accordance with 10CFR50.46. This time includes an instrumentation and actuation logic system delay (1 sec.), an emergency diesel generator startup & load time delay (10 sec), and allowance for HPI components to start and valves to stroke (24 seconds).

The valve stroke times in question have always been well within the design basis. The slowest actual valve stroke time is approximately 15 seconds vs. the design basis allowable of 24 seconds. The discrepancies identified had not resulted in any inappropriate design changes or inadequate maintenance or testing. The FSAR and SP are being corrected. The inaccurate identification of the valve design basis stroke time in the SP and FSAR had no adverse impact on nuclear safety.

C. Corrective Actions:

1. SP 1300-3H has a maximum stroke time of 18 seconds for MU-V-16A through D and a maximum stroke time of 19 seconds for MU-V-36 and MU-V-37. These maximum stroke times do not need to be changed. Exceeding these maximum stroke times would have caused the affected component to have been declared inoperable and would have started the Technical Specification time clock in accordance with Administrative Procedure (AP) 1041, "IST Program Requirements." The affected design basis stroke times in SP 1300-3H have been revised to 24 seconds to address the documentation discrepancies.

2. FSAR Section 6.1.3.1 states that the Makeup & Purification System is designed to be in full operation within 25 seconds after receiving an actuation signal. A Proposed FSAR Update (PFU) has been prepared (PFU-98-T1-124) to change this to read as follows: "The LOCA analysis assumes HPI operation 35 seconds after actuation. This allows 10 seconds to energize the on site emergency power source, 1 second for instrumentation response and 24 seconds for Makeup System component response time."

D. Schedule for Completion of Corrective Actions:

1. PFU-98-T1-124 will be included with FSAR Update 14, which is currently due in April 1998.

IFI 96-201-06 "Consequences of Failure of Auxiliary Steam Piping (Section E1.2.2.e)"

A. Description of the Finding:

The consequences of failure of non safety-related Auxiliary Steam (AS) piping in the Auxiliary Building (AB). Failure of the auxiliary steam piping due to a seismic event could result in degradation of safety-related equipment classified for a mild environment.

B. Discussion:

Although GPU Nuclear has been unable to retrieve the final report documenting the analysis of pipe breaks in the AB, a draft of the report has been located.

The AS lines in the Fuel Handling and Auxiliary Buildings are low pressure and thus were investigated for cracks only. The pressure is too low to achieve critical flow, and the steam jet force from the largest postulated steam line crack is small. No damage to electrical or Instrumentation and Control (I&C) equipment will result.

A preliminary calculation, which was completed before the end of the inspection, shows that leakage from a crack (1/2 pipe ID x 1/2 thickness) is 580 lbm/hr at 11 psig line pressure. This is equal to a heat load of approximately 700,000 btu/hr, which in our engineering judgment is not sufficient to cause a harsh environment.

The routing of the AS lines in the AB is shown in Updated FSAR Figures 14A-3 and 14A-4. The conclusions in the Updated FSAR have been substantiated by reconstituted analysis although this analysis has not yet design verified.

The finding expressed in IR 96-201 raised the additional concern of failure of the AS line due to a seismic event. Based on the preliminary results from a letdown line break in the AB, it does not appear that the failure of the AS line would result in a harsh environment before the break could be detected and isolated.

C. Corrective Actions:

1. Analysis will be performed to a) confirm our engineering judgment that the heat load from a crack break is insufficient to cause a harsh environment, and b) the consequences of a seismically induced failure of the AS line is bounded by the letdown line break.

2. The effects of an AS line break in the AB on safety-related equipment will be evaluated.

3. Documentation will be completed to support statements in the Updated FSAR regarding the consequences of an AS System line break in the AB.

D. Schedule for Completion of Corrective Actions:

1. The analysis is scheduled for completion by October, 1997.

2. Evaluation of the effects on safety-related equipment will be completed by December 31, 1997. We do not expect the results will show any effects on safety related equipment.

3. Changes to the FSAR will be submitted in FSAR Update 14, which is currently due in April 1998.

IFI 96-201-07 "Loss of Pressure in MU&P Tank due to Letdown Line Break (Section E1.2.2.c)"

A. Description of the Finding:

Evaluate the impact of loss of pressure in the Makeup Tank (MUT) on the Net Positive Suction Head (NPSH) for the High Pressure Injection (HPI) Pumps due to letdown line break or crack combined with the failure of the check valve in the line to the MUT.

B. Discussion:

1. A break upstream of the block orifice large enough to depressurize the line to less than MUT pressure would result in Emergency Safeguards (ES) actuation and automatic closure of MU-V-3 & MU-V-2A and MU-V-2B. Check valve MU-V-115 would prevent backflow from the MUT to the break. If this check valve failed, the gas overpressure in the MUT would be lost. The loss of MUT pressure is not an immediate threat to Makeup Pump operation. Adequate NPSH is available to the operating Makeup Pump without any gas overpressure and minimum operating water level. The pressure and level in the MUT are continuously monitored and if the required pressure and level in the MUT and the Reactor Coolant System (RCS) pressurizer could not be maintained, a controlled plant shutdown and cooldown would be initiated.

2. The letdown line downstream of the block orifice is Seismic I piping at low pressure and temperature. There is no licensing or design requirement to postulate a break in this line.

C. Corrective Actions:

No additional action is required.

IFI 96-201-08 "M&UP Pump NPSH When Taking Suction From BWST or Makeup Tank (Section E1.2.2.c)"

A. Description of the Finding:

GPU Nuclear was asked to review the available Net Positive Suction Head (NPSH) for the High Pressure Injection (HPI) pumps when taking suction from the Borated Water Storage Tank (BWST) at "Low-Low Level" or during potential vacuum conditions in the makeup tank.

B. Discussion:

Abnormal Transient Procedures (ATPs) 1210-6, "Small Break LOCA Cooldown," Revision 23; 1210-7, "Large Break LOCA Cooldown," Revision 23;" and 1210-10, "Abnormal Transient Rules, Guides, and Graphs," Revision 32, were effective on March 19, 1997. These procedures ensure that the "piggy-back" mode is established prior to reaching the BWST "Low Level" alarm at 9 ft 6 in. Therefore, with 9 ft 6 in of level in the BWST, the Makeup Pumps have adequate NPSH under maximum flow operating conditions.

With MUT level and pressure maintained in the normal operating band, there is no mechanism or specific sequence of events that could result in a vacuum in the MUT during tank drawdown. The operator is required to open the suction valve from the BWST if MUT level reaches 55 inches. MUT indicated level would have to be less than 0 inches before vacuum conditions develop. Therefore, the potential for vacuum conditions in the MUT is precluded by system design and operational controls.

C. Corrective Actions:

No additional action is required.

IFI 96-201-09 "Incomplete DC System Voltage Drop Calculations (Section E1.2.3.2.a)"

A. Description of the Finding:

Calculation C-1101-734-5350-004, "TMI-1 DC System Calculation," Revision 1, provides individual DC circuit voltage drop analyses for the redundant station battery circuits. The circuits for Makeup System isolation valve MU-V-18 and numerous other DC circuits were listed in the calculation, but the voltage drop analyses were not performed. The calculation also concluded that further investigation was required to determine the adequacy of terminal voltages at various DC equipment.

B. Discussion:

A preliminary voltage drop analysis was performed during the inspection that demonstrated adequate voltage was available at the terminals of MU-V-18.

C. Corrective Actions:

1. Calculation C-1101-734-5350-004, "TMI-1 DC System Calculation," will be updated to address the available voltage at MU-V-18 and the remaining circuits. The updated calculation is expected to also confirm adequate voltage is available at the terminals of MU-V-18.

D. Schedule for Completion of the Corrective Actions:

1. The revision to calculation C-1101-734-5350-004, "TMI-1 DC System Calculation," is scheduled to be completed by December 31, 1997.

URI 96-201-10 "Alignment of MU-V-18 DC Power Supply (Section E1.2.3.2.b)"

A. Description of the Finding:

During the inspection GPU Nuclear told the inspection team that the loss of 1A or 1B 125 VDC power concurrent with an High Pressure Injection (HPI) line break downstream of the last check valve on the injection line would prevent the HPI system from performing its design Emergency Core Cooling System (ECCS) function if the power source lineup and the valve positions in the cross-connect line between Makeup Pumps A and B were not controlled. If the system is not properly aligned, the loss of a 125 VDC power train could result in failure of MU-V-18 to close, thus, degrading one HPI train and the other HPI train would be inoperable due to the loss of the 125 VDC breaker control power from the same source. The team was told that this problem had been discovered before the inspection team arrived at the site. But when the alignment was checked on January 8, 1997, it was discovered that the electrical power alignment was incorrect. The licensee issued temporary orders to control room staff to perform checks of the cross-connect valve positions and DC power source alignment every shift. The licensee issued PFU 98-T1 to revise the FSAR and initiated changes to the operating procedures. The team noted that the requirements in 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" regarding prescribing activities affecting quality by documented instructions, procedures or drawings and accomplishing the activities in accordance with these instructions, procedures, or drawings were not apparently met.

B. Discussion:

The alignment of the power supply for MU-V-18 is not required to ensure nuclear safety. Further review following the design inspection determined that regardless of the power alignment for 1M DC distribution panel and MU-V-18, there is no single failure which can reduce HPI capability to less than that assumed in the ECCS Loss of Coolant Accident (LOCA) analysis.

The concern at the time of the inspection was for a break in the normal makeup line at the Reactor Coolant System (RCS) nozzle and an assumed failure of the "A" or "B" DC distribution system, that only one HPI Pump would be operating and that the minimum HPI flow would not be supplied to the RCS. This concern was based on the MU-V-18 valve failing open concurrent with the redundant train HPI Pump failing to start on loss of DC power.

A more detailed analysis concluded that these events (MU-V-18 "open" with HPI Pump failure) would not occur for the postulated single failure. Two scenarios were considered. One where the Emergency Safeguards (ES) actuation occurs prior to the loss of DC power and a second where the ES actuation occurs after the loss of DC power.

In the first case, both trains of HPI are initiated and then MU-V-18 fails "open" when DC is lost. The running HPI Pump does not trip when DC power is lost. With two HPI Pumps operating, the flow through MU-V-18 does not prevent HPI from performing its nuclear safety function.

In the second case, the 1M DC distribution panel transfers to the alternate power source when DC is lost and MU-V-18 will close upon ES although only one HPI pump will start. One HPI train is capable of providing the minimum HPI flow assumed in the ECCS LOCA analysis.

There was no adverse impact on nuclear safety from this finding. This concern was based on an incomplete analysis of the events at the time of the inspection.

C. Corrective Actions:

1. FSAR Section 6.1.3.1 will be revised.

D. Schedule for Completion of Corrective Action:

1. A change to FSAR Section 6.1.3.1 has been prepared (PFU 98-T1-119) and will be incorporated into FSAR Update 14 which is currently due in April 1998.

URI 96-201-11 "Makeup Tank Level Instrument Loop Tolerances (Section E1.2.4.2.a)"

A. Description of the Finding:

The team reviewed the instrument data sheets in Surveillance Procedure (SP) 1302-5.17, Revision 17, "Make-up Tank Level Instrumentation." The data sheet for instrument loop MU14-LT specified a loop error (tolerance) of ±1%. However, calculation C-1101-662-5350-049, "TMI-1 Makeup Tank Level Error for Accident Conditions (LT-778Loop)," Revision 0, estimated a loop error of ±1.23%. Although the loop error in the data sheet was more restrictive, the team questioned the inconsistency in the two documents. The licensee stated that for all safety-related instrument loops a conservative loop error of ±1% was assumed. Documentation justifying this assumption could not be retrieved.

After further review of surveillance data sheets and calculations, the team identified three other instances of inconsistency in the documents. The data sheets for control room indicator instrument loop MU-LI-778A and computer point A0498 instrument loop MU14-LT specified a loop error of ±1% each, but calculation C-1101-662-5350-049, "TMI-1 Makeup Tank Level Error for Accident Conditions (LT-778LOOP)," Revision 0, determined that the loop errors should be limited to ± 0.64% and ± 0.73% respectively. Calculation C-1101-624-5350-002, "Makeup Tank Level Error for Accident Conditions (MU-14-LT Loop)," Revision 1, determined that the loop error should be ± 0.57% for MU14-LR in instrument loop MU14-LT, but the data sheet specified a loop error of ±1.0. The calculated allowable loop errors were more restrictive than the instrument loop data sheets. The licensee initiated action to resolve the team's concerns.

B. Discussion:

Multiple calculations of Makeup Tank (MUT) level instrument loop accuracy are on record. These calculations were identified with different system designations. Tolerances determined by these calculations did not match the acceptance criteria in the SP 1302-5.17.

Calculation C-1101-624-5350-002, "Makeup Tank Level Error for Accident Conditions (MU-14-LT Loop)," is for MU-V14-LT loop errors. Revision 2 of this calculation was to support a modification that would have replaced the analog Integrated Control System/Non Nuclear Instrumentation (ICS/NNI) System with a Digital Control System, but the modification was not installed.

Calculation C-1101-662-5350-049 "TMI-1 Makeup Tank Level Error for Accident Conditions (LT-778 LOOP)," is for LT-778 loop errors.

Calculation C-1101-211-5350-057, "Makeup Tank Level Instrument Drift," which was intended for MU-14-LR and MU-LI-778A Instrumentation Drift, was not found until after the inspection. It supports the 1% tolerance for some of the loop outputs.

The surveillance program has maintained the instrument calibration to a high degree of accuracy (allowable error was 1%). Calculation C-1101-211-5350-057, 'Makeup Tank Level Instrument Drift," supports the existing surveillance tolerance for the critical loop outputs.

In order to assure that any potentially similar problems with other process measurements and setpoints are addressed, plans for a setpoint basis update program was initiated in response to the NRC's 50.54(f) letter of October 9, 1996. This program will provide a data base for reference to the design basis document (i.e. setpoint calculation) for each Nuclear Safety Related (NSR) and Regulatory Required (RR) setpoint. If a design basis document can not be found, an engineering task will be initiated to have the setpoint design basis established. Once a design basis document is established, the document will be referenced in the setpoint program database as a source document. This program will ensure that design basis references for setpoints are captured and easily retrievable.

C. Corrective Actions:

1. A new calculation which includes all the outputs from MU-14-LT and MU-LT-778 will be prepared.

2. SP 1302-5.17, "Make-up Tank Level Instrumentation," will be revised to agree with the tolerances in the revised calculation.

3. The calculation process upgrades will allow quicker review and identification of calculations which support key plant parameters. (See OI 96-201-28 Corrective Action C.4)

D. Schedule for Completion of Corrective Actions:

1. The new calculation for Makeup Tank level instrument loop accuracy is scheduled to be completed by July 30 1997.

2. SP 1302-5.17, "Make-up Tank Level Instrumentation," is scheduled to be revised by August 30, 1997.

URI 96-201-12 "FSAR Discrepancies (Section E1.2.7)"

A. Description of the Finding:

The team identified four discrepancies between the FSAR and other plant documentation. These four items were identified as follows:

1. FSAR Table 14.2-18, Sheet 1, indicated that "total flow" for a High Pressure Injection (HPI) line break at a Reactor Coolant System (RCS) pressure of 1800 psig is 347.5 gpm rather than 397.5 gpm.

2. FSAR Section 14.2.2.4.3.a indicated that an open issue existed with regard to tripping the Reactor Coolant Pumps (RCPs) during a Small Break Loss of Coolant Accident (LOCA) but did not state that a manual trip on a loss of subcooling margin was acceptable.

3. FSAR Table 9.1-2 and the GMS-2 database contain conflicting design data for MU&P system components.

4. FSAR Section 8.2.3.1.b describes the Emergency Diesel Generator (EDG) continuous rating as 2600 kw instead of the correct rating of 2750 kw stated in the vendor manual, VM-TM-0191, Revision 29.

B. Discussion:

The resolution of these discrepancies requires a revision of the FSAR. The GPU Nuclear process for FSAR updates requires that a Preliminary FSAR Update (PFU) be submitted. A PFU includes the recommended text revision and the required safety review. PFUs have been submitted for all of the discrepancies identified.

Each of these discrepancies is discussed as follows:

1. Table 14.2-18 indicated HPI flow of 347.5 gpm rather than 397.5 gpm. No analysis or testing/surveillance activities were identified where this error was continued. The HPI flow test and LOCA analysis both used the proper value of 397.5 gpm.

2. Section 14.2.2.4.3.a indicated there is an open issue with regard to tripping RCPs. This error had no impact on nuclear safety because the "open issue" had been resolved. This omission in the FSAR could only have caused someone to raise the question as the inspection team has done.

3. Table 9.1-2 contained erroneous Makeup System component data. The erroneous component data were corrected to reflect the equipment installed. A review of these errors did not identify any installed equipment that is not qualified for its service.

4. Section 8.2.3.1.b indicated an EDG continuous rating of 2600 kw rather than 2750 kw. Had the erroneous value of 2600 kw been used as the EDG rating in an evaluation, the conclusion reached would have been conservative because the actual EDG rating is 2750 kw.

In September 1996, a new emphasis was placed on the quality and completeness of the FSAR. FSAR section owners have been established as a step to improve the quality.

C. Corrective Actions:

1. As stated in the inspection report, the following PFUs were prepared to correct the four discrepancies identified above: PFU 98-T1-113; PFU 98-T1-123; PFU 98-T1-127; and PFU 98-T1-135. It is noteworthy that after a PFU has been submitted, the change is placed in a database for immediate use by reviewers in performing safety evaluations.

D. Schedule for Completion of Corrective Actions:

1. The above listed PFUs will be included in FSAR Update 14, which is currently due in April 1998.

URI 96-201-13 "Adequacy of BWST Setpoint for DHRS Pump Switchover to RB Sump (Section E1.3.2.2.a)"

A. Description of the Finding:

The design inspection team identified five concerns with a 1994 calculation provided to demonstrate the adequacy of the plant design for Borated Water Storage Tank (BWST) switchover. The analysis covered the evolution where it is required to switch the Emergency Core Cooling System (ECCS) pump suction to the Reactor Building (RB) sump when the BWST is depleted in a Loss of Coolant Accident (LOCA) scenario. The NRC concerns were: (1) failure to consider instrument error in the alarm, (2) use of a non-conservative RB pressure assumption, (3) the validity of the assumed time for operator response, (4) disagreement between the valve stroke times used and the design basis time stated in the valve stroke test procedure, and (5) improper selection of a constant and invalid flow assumption in the application of methodology for vortex determination.

B. Discussion:

The concerns of the inspection team were evaluated. A brief assessment of the error and the significance of that error in the analysis results is evaluated as follows:

1. The failure to consider instrument error was caused by a misconception in that, since the "Lo Lo Level" alarm setpoint was determined using instrument error, the error was already accounted for and would not need to considered again in this analysis. The actual instrument error was measured on December 21, 1996 and found to be very small.

2. The RB pressure assumption was the best available information and was felt to be an adequate assumption for the purpose of the analysis being performed. The information that confirmed that this was not a conservative assumption was not available until December 1996 during the inspection. The change in RB pressure had a significant non-conservative effect on the analysis.

3. The operator response times were taken from simulator performance records. The appropriateness of data from planned versus unplanned activities in the simulator was not considered. There were some misinterpretations of the data due the manner in which plant computer data is collected and presented. The operator response times used were reasonable.

4. The valve stroke times used in the analysis were appropriate. However, the surveillance procedures which test the stroke time of these valves had acceptance criteria greater than the assumed stroke time.

5. The calculation of the minimum level at a specified flow rate required to prevent air entrainment due to vortexing was based on TMI-1 startup data and a methodology from an industry periodical. The startup test was not performed at a constant flowrate and this was not obvious in the presentation of the results of the test. Also, the empirical method described in the periodical was valid with only a limited range of constants. The TMI vortex analysis included errors associated with the flowrate from the TMI-1 startup test and use of a constant outside the prescribed range for the methodology used. These errors produced a small non-conservative change in the result.

The overall impact of these errors was evaluated when the concerns were raised. The above items 1,3,4 & 5 together would not have changed the conclusion of the analysis. However, the new RB pressure assumption resulted in a significant decrease in the time available to complete the evolution and meet the acceptance criteria. GPU Nuclear initiated work on this problem immediately on Dec. 20,1996 when the preliminary drawdown analysis using the revised RB pressure indicated a potential problem. At 4:00 am on December 21, 1996, GPU Nuclear determined that current analysis methods could not definitively conclude that some quantity of air would not be drawn into the suction of the Decay Heat (DH) and Building Spray (BS) pumps in the Large Break LOCA scenario with conservative assumptions for the plant conditions. Therefore, the DH and BS pumps were declared inoperable.

A reportable event was declared and reported to NRC by phone and by submittal of licensee event report (LER 96-002).

Prior to expiration of the Technical Specification Allow Outage Time (AOT), an alternative procedure had been implemented which allowed GPU Nuclear to conclude that air entrainment in the ECCS pumps would not occur during the postulated LOCA.

C. Corrective Actions:

1. A revised switchover sequence was developed which increased the time for operator action to complete the transfer to the RB recirculation mode prior to reaching the point of vortexing. Temporary procedure changes were implemented on December 21, 1996 for the revised switchover sequence. Abnormal Transient Procedures (ATPs) 1210-6,"Small Break LOCA Cooldown," Revision 23; 1210-7,"Large Break LOCA Cooldown," Revision 23, 1210-8,"RCS Superheated," Revision 20; and 1210-10, "Abnormal Transients Rules, Guides, and Graphs," Revision 32, were effective March 13, 1997 to replace the temporary changes.

2. All licensed operators were briefed on the procedure changes.

3. The new switchover sequence was successfully performed at the TMI-1 Replica Simulator using the revised procedural guidance.

4. All persons who are certified to perform calculations or design verifications have received additional training on the calculation and design verification process with special emphasis on validation of inputs and assumptions.

5. Formal analysis will be completed to document the new procedure for switchover.

D. Schedule for Completion of Corrective Actions:

1. The revised BWST switchover analysis will be completed by December 31, 1997. Other actions listed above are complete.

URI 96-201-14 "Adequacy of Safety Evaluation of an FSAR Change (Section E1.3.2.2.b)"

A. Description of the Finding:

Safety Evaluation (SE) No. 115403-004, Rev. 0, did not identify that, because the required Net Positive Suction Head (NPSH) for the Decay Heat Removal (DHR) Pumps would not be met without taking credit for containment overpressure, the probability of occurrence of malfunction of the DGR Pumps previously evaluated in the safety analysis report may be increased, and thus, a potential Unreviewed Safety Question (USQ) as defined in 10CFR50.59 was involved.

B. Discussion:

Safety Evaluation SE-115403-004, Rev 0, considered that the assumption of no credit for containment overpressure above the sump vapor pressure only applied to the licensing basis accident analysis condition of 3000 gpm Low Pressure Injection (LPI) flow rate and 1500 gpm RB Spray flow rate, and adequate NPSH was demonstrated at these conditions. A flow rate limited to 3300 gpm by Abnormal Transient Procedure (ATP) 1210-7, "Large Break LOCA Cooldown," was considered to be beyond the licensing basis as presently defined by the TMI-1 FSAR and the NRC Safety Evaluation Report (SER) dated July 11, 1973 for TMI-1. Since this condition was interpreted as being beyond the licensing basis, it was determined that use of conservative but more realistic assumptions in terms of containment overpressure was acceptable.

While applying containment overpressure could legitimately be viewed as a design control concern under Appendix B to 10 CFR 50, at the preliminary enforcement conference on May 22, 1997, GPU Nuclear questioned whether SE No. 115403-004, Revision 0, should have determined that an unreviewed safety question existed by considering assumptions beyond the licensing basis. Safety Evaluation 115403-004, Revision 0, explicitly documented the rationale substantiating the safety evaluation conclusions, and conservatively incorporated instrument error not previously considered in the licensing basis. GPU Nuclear requested further consideration and guidance from the NRC Staff on this question. We suggested that safety evaluations should be performed in a manner consistent with licensing basis assumptions because the objective of those evaluations is to determine whether a change preserves the plant's licensing basis. Based on additional discussion with NRC Staff, GPU Nuclear understands that it is the NRC's position that use of containment overpressure for the 3300 gpm LPI flow rate is a deviation from the existing licensing basis. GPU Nuclear understands that 10CFR50.59 and USQ criteria interpretation is an evolving issue and is continuing to develop based on issuance of NRC document, SECY 97-035. We are participating in various industry groups related to this issue and continue to monitor these activities to fully understand how to more effectively address 10 CFR 50.59 criteria.

GPU Nuclear understands that this open item reflects the NRC staff's concern that the NPSH calculation for the DHR Pumps should assume LPI flow of 3150 gpm (the LPI flow at 0 psig presented in FSAR Table 14.2-27 and derived from the flow curves used in the large break LOCA analysis) and an appropriate allowance for instrument error. In contrast, Section 6.4.2 of the FSAR reflects a licensing basis calculation that assumes LPI flow of 3000 gpm (the nominal flow rate used in the accident analysis) with no allowance for instrument error.

We believe that this NPSH calculation, using the licensing basis assumptions, is not sufficiently conservative. We also believe that the NPSH for the DHR Pumps should be based on an LPI flow of 3300 gpm, based on the ATP 1210-7 procedure flowrate limit, coupled with an appropriate allowance for instrument error. This will bound the 3150 gpm flowrate described in FSAR Table 14.2-27.

The existing licensing basis only addresses the accident analysis assumed LPI flow rate of 3000 gpm. Safety Evaluation 115403-004, Revision 0, recognized that ATP 1210-7 allowed a higher LPI flow rate. This higher LPI flow rate provides additional margin beyond the accident analysis value in terms of core cooling (10 CFR 50.46) concerns. The value used for the assumption on containment overpressure was conservative but reflected an expected Reactor Building response to the postulated design basis accident. Safety

Evaluation 115403-004, Revision 0, determined the NPSH available to the DHR Pumps to be adequate at both the 3000 gpm accident analysis value and the 3300 gpm procedurally limited flow value.

The calculated NPSH available for the DHR Pumps at conditions of 3300 gpm LPI and 1500 gpm RB Spray flow was approximately 0.5 ft. (0.22psi) less than required with no credit for containment overpressure based on a 1990 calculation and the associated safety evaluation. Abnormal Transient Procedure (ATP) 1210-07, "Large Break LOCA Cooldown," directs the operator to turn off the RB Spray Pumps at a RB Pressure of 4.0 psig. Under this condition there is an excess available NPSH of 2.9 ft. Thus additional NPSH margin is provided when considering the expected plant and operator response to the postulated design basis accident.

No immediate corrective actions were required since this condition only involves a reduction in the margin included in the NPSH determination and does not represent a safety or operability issue.

C. Corrective Actions:

1. GPU Nuclear will submit a license amendment application for NRC approval to revise the licensing basis so that the NPSH for the DHR Pumps is calculated based on an LPI flow rate of 3300 gpm and an appropriate allowance for instrument error. The application will include use of containment overpressure in the calculation of available NPSH for the maximum pump flow rate of 3300 gpm. The calculation will be revised accordingly.

   Because this application will change the plant's licensing basis and NPSH calculation for the DHR Pump, and will be approved by license amendment, it will also eliminate any USQ that may exist as a result of the revised profile evaluated in Safety Evaluation 115403-004, Revision 0.

D. Schedule for Completion of Corrective Actions:

1. The DHR Pump NPSH calculation using 3300 gpm LPI maximum flow and an appropriate allowance for instrument error will be completed to support submittal of a license amendment application by August 1997 for NRC approval prior to startup following the Cycle 12 refueling (12R) Outage, which is scheduled to begin in September, 1997.

IFI 96-201-15 "Technical Specification Discrepancy (Section E1.3.2.2.c)"

A. Description of the Finding:

Technical Specification (TS) 3.3.1.1.f states: "The two reactor building sump isolation valves (DH-V-6A/B) shall be either manually or remotely operable." This requirement was not appropriate because only remote operation of the valves would support the design basis assumptions for the transition time of the Decay Heat Removal (DHR) Pump suction switchover from the Borated Water Storage Tank (BWST) to the Reactor Building (RB) Sump during post-accident conditions. In addition, manual operation of these valves during an accident may not be possible because the DHR System pump rooms may not be accessible due to high radiation conditions.

B. Discussion:

Valve surveillance tests require these valves to be tested for remote operation. TMI-1 has not been operated with these valves in a condition where they were only manually operable.

C. Corrective Actions:

1. Technical Specification Change Request (TSCR) No. 263 has been submitted for NRC review and approval. This TSCR deletes the words "either manually or" and includes other unrelated changes that were requested prior to startup for Cycle 12 operation.

C. Schedule for Completion of Corrective Actions:

1. This action will be complete upon issuance by the NRC of the amendment requested in TSCR No. 263.

URI 96-201-16 "Discrepancy Between FSAR and Technical Specifications Regarding DHRS Leakage (Section E1.3.2.2.d)

A. Description of the Finding:

- FSAR Sections 6.4.3, 6.4.4 and Table 6.4-3 stated the design basis leakage in the Auxiliary Building (AB) from the Decay Heat Removal (DHR) System and Building Spray (BS) System as 2255 cm^3/hr (0.6 gal/hr). FSAR Section 14.2.2.5 and Table 14.2-20 documented a 2 hour dose for engineered safeguards leakage of 2255 ml/hr of 0.037 rem.

Technical Specifications (TS) 4.5.4, "Decay Heat Removal System Leakage," allowed 6 gal/hr leakage from the DHRS and referred to FSAR Section 6.4.4 and Table 6.4-3. The TS bases stated that dose was 0.39 rem from the 6 gal/hr leakage and used FSAR Section 14.2.2.5(d) as a reference for this dose. There were no Technical Specifications for BS System leakage. The team noted that leakage control in the DHRS was more important than for the BS system since the BS system operates for a few hours after the accident while the DHR System operates over the entire duration of the accident.

The discrepancy between the FSAR and TS was identified by the licensee as inspection observation 212-61 in report TDR 1092 during a self-assessment of the DHR System performed in 1992. The preliminary safety significance review, documented in memorandum C320-92-1287, stated that the observation was not safety significant because the dose of 0.39 rem "still represents a negligible portion of the total two-hour thyroid dose of 189 rem." No action was taken to resolve the discrepancy.

The licensee stated the FSAR would be revised as part of licensing Action Request (LAR) 95076.01. The team noted that the LAR was initiated May 25, 1996. The above discrepancies had not been corrected and the FSAR updated to assure that the information included in the FSAR contained the latest material as required by 10 CFR 50.71(e).

B. Discussion:

GPU Nuclear acknowledges that the discrepancy identified is valid and has initiated a corrective action item to evaluate this discrepancy and revise the FSAR and other documents as required. The discrepancy between the FSAR and TS basis for Emergency Core Cooling System (ECCS) leakage has a very minor impact on the Maximum Hypothetical Accident (MHA) dose consequence result with minimal impact on nuclear safety as assessed at the time of the Low Pressure (LPI) Safety System Functional Inspection (SSFI).

C. Corrective Actions:

1. The MHA dose consequence analysis (Calculation C-1101-202-E260-329) is being revised and NRC review of a change will be requested to include a more conservative leakage assumption consistent with the TS.

2. Changes to TS 4.5.4 will be requested to include leakage from all "recirculation" systems.

3. The affected FSAR sections will be revised.

D. Schedule for Completion of Corrective Actions:

1. Submittal of the revised assessment of dose consequences from the MHA using the revised leakage assumptions is scheduled for submittal by early July 1997.

2. The Technical Specification Change Request to revise the TS Surveillance Specification Section 4.5.4 is scheduled for submittal by early July 1997.

3. The affected FSAR sections will be revised in the next FSAR update following completion of these actions which require NRC approval. LAR 95076.01 will track the completion of the FSAR Update.

URI 96-201-17A "Leak Testing of DHRS Pump Suction Check Valves (Section E1.3.2.2.e)"

A. Description of the Finding:

The function of the Decay Heat Removal (DHR) System valves, DH-V-14A and DH-V-14B, had been evaluated as part of an Inservice Testing (IST) program scope review. A calculation was performed which accurately determined that there was inadequate pressure to force post accident water through this check valve flowpath and become a potential source of offsite dose. However, the calculation was documented only in a memo. The inspection team questioned the Reactor Building (RB) pressure assumption in the memorandum. The Equipment Qualification (EQ) RB pressure profile had been revised after the memorandum had been prepared.

B. Discussion:

The revised RB pressure profile did result in higher RB pressure which would potentially challenge the DHR System check valves using the conservative assumptions typically assumed for such analysis. Additional RB pressure analysis was performed which demonstrated that with more limiting operating restrictions of RB pressure, Borated Water Storage Tank (BWST) temperature and RB fan cooler availability, it could again be assured that leakage through these check valves could not reach the environment. Interim operating restrictions were established to ensure that there would be sufficient margin pending the development and completion of appropriate IST.

The potential safety issue is leakage past this check valve in a post Loss of Coolant Accident (LOCA) condition where that leakage then flows into the BWST which is vented to atmosphere. Although not formally proceduralized as a test, the DH-V-14A and DH-V-14B check valves have been functionally tested each time the plant transitioned to DHR operations. During this evolution, Reactor Coolant System (RCS) pressure (greater than 200 psig) is applied across this check valve and recent experience (the last couple of outages) has been that no leakage was observed. Leakage is indicated by lifting relief valve DH-V-57 or observation of BWST level increasing. DH-V-14A/B are disassembled and inspected as part of the IST program and significant wear would be detected and repaired.

C. Corrective Actions:

1. Training has been provided to the engineering staff who may perform design basis calculations. This training emphasized procedural compliance and specifically the use of the EP-006 format for all design calculations which includes the identification of other affected calculations.

2. The operational restrictions put in place December 20, 1996 will remain in effect until the Cycle 12 Refueling (12R) Outage which is scheduled to begin in September, 1997. Thereafter, beginning in the 12R outage, valves DH-V-14A and DH-V-14B will be tested as part of the IST program.

D. Schedule of Corrective Actions:

1. Leakage testing of the DH-V-14A/B will be implemented as part of the IST program during the 12R Outage tests.

URI 96-201-17B "Leak Testing of DHRS Pump Discharge Check Valves (Section E1.3.2.2.e)"

A. Description of the Finding:

In a design basis accident, if a Decay Heat (DH) Pump fails, the discharge cross connect valves DH-V-38A and DH-V-38B are opened and flow is provided through both low pressure injection paths. In this condition the discharge check valve (DH-V-16) on the failed pump would need to close to ensure adequate Low Pressure Injection (LPI) flow. The Inservice Testing (IST) program did not recognize and test this "Close" function of DH-V-16A and DH-V-16B.

B. Discussion:

The DH-V-16A/B valves are tested for their "open" function quarterly and during refueling outages. GPU Nuclear agrees that these valves have a "close" safety function. A test procedure was written and performed within 24 hours of discovery of the condition to verify the "close" safety function.

These valves are periodically disassembled and inspected as part of the preventative maintenance program. These inspections have shown that the valves are in good operating condition with no significant signs of degradation. The "open" function of these valves has been periodically tested as part of the IST program. Therefore, we believe that the valves would have functioned as required in the past. The IST program will continue to ensure the "close" functionality of these valves in the future.

C. Corrective Actions:

1. A test on December 19, 1996 verified that valves DH-V-16A and DH-V-16B seated in the "closed" position. Both valves performed acceptably during the test.

2. IST procedures will be revised to include "closed" function testing of the DH-V-16A and DH-V-16B valves.

3. "Close" testing of DH-V-16A/B has been added to the work scope for cold shutdown activities in the event of an unscheduled cold shutdown prior to the IST procedure updates being completed.

D. Schedule for Completion of Corrective Action:

1. Changes to IST procedures to add "closed" function testing of the DH-V-16A and DH-V-16B valves will be completed prior to the tests required during the Cycle 12 Refueling (12R) Outage which is scheduled to begin in September 1997.

URI 96-201-17C "Inspection of DHRS Pump Vault Floor Drain Check Valves (Section E1.3.2.2.e)"

A. Description of the Finding:

As described in FSAR section 6.4.5 a leak in a Decay Heat (DH) vault will not affect more than one train of equipment because there are check valves in the floor drains to prevent water from backing up into the vaults. There are two floor drains from each DH vault to the Auxiliary Building (AB) Sump. The inspection team identified that the procedure which verifies the operability of these check valves did not include both valves.

B. Discussion:

When the team identified the omission from the procedure, action was begun immediately to disassemble and verify the condition of the valves which were not being surveilled. Also, a procedure change was prepared to add these previously omitted valves to the procedure.

The procedure in question covered all of the floor drains in the auxiliary and fuel handling buildings. Each room in these buildings has one of a typical floor drain arrangement with a swing check valve and special access for inspection and cleanout. The procedure covered all the drains of this type. The Emergency Core Cooling System (ECCS), Decay Heat, Building Spray and Makeup Pump Rooms each have an additional drain with a level alarm. These drains have a ball float type check valve. These ball float type check valves were not being inspected. The purpose of the procedure clearly addresses the IE Circular 78-06 which identified the issue years ago. It also recognizes that the concern is "potential flooding of ECCS rooms."

C. Corrective Actions:

1. These valves were inspected and found to be operable.

2. The procedure U-17, "Zurn Floor Drain Inspection" was revised in Revision 5, effective February 20, 1997 to include testing the ball check valves.

3. Each of the ECCS pump room drain valves will be added to the Inservice Testing (IST) program as augmented IST.

4. Flow Diagram 302-719, will be revised to identify the drains associated with the Building Spray, Decay Heat and Makeup pump rooms.

D. Schedule for Completion of Corrective Actions:

1. The addition of ECCS pump room floor drain valves to the Augmented IST program will be completed before startup following the Cycle 12 Refueling (12R) Outage which is scheduled to begin in September 1997.

2. Flow diagram 302-719, "Reactor & Auxiliary Bldg. Sump Pump and Drainage System" is scheduled to be revised by December 31, 1997.

URI 96-201-18 "Timeliness of Action on SSFI Open Items (Section E1.3.2.2.f)"

A. Description of the Finding:

This item deals with timeliness of action on Safety System Functional Inspection (SSFI) Open Items. As stated in the inspection report, GPU Nuclear performed an SSFI of the Decay Heat Removal (DHR) System in 1992. Technical Data Report (TDR) No. 1092, "Low Pressure Injection System - Safety System Functional Inspection," was approved on January 12, 1993. The team selected about 30 open items to verify the status of corrective actions. In the following instances, the licensee delayed actions on SSFI open items that had been pending for up to four years:

1. SSFI Items 212-1 through 212-11 concern design basis piping analysis and related matters.

2. SSFI Items 212-12 and 212-13 relate to potentially non-conservative RB pressure and sump liquid temperatures used in the Low Pressure Injection (LPI) Pump Net Positive Suction Head (NPSH) calculation.

3. SSFI Item 212-42 questioned why the DHR Pump vent valves, DH-V-75A&B and DH-V-76A&B, were not in the Equipment Qualification (EQ) program.

4. SSFI Item 212-61 deals with discrepancies between the FSAR and the Technical Specifications on DHR System allowable leakage.

B. Discussion:

The NRC has identified this URI as an example of potentially not satisfying 10 CFR 50 Appendix B, Criterion XVI in that conditions adverse to quality had not been corrected promptly. At the May 22, 1997 Predecisional Enforcement Conference GPU Nuclear agreed that prompt action had not been taken on all LPI SSFI open items. GPU Nuclear is committed to prompt closeout of significant issues according to relative priorities and available resources. GPU Nuclear agreed to review this issue and provide additional information.

A broader assessment of this URI has been conducted which agrees that the items identified during the design inspection and discussed in IR 96-201 are conditions adverse to quality. These items should have been addressed in a more timely manner. Each are briefly described below. Although they are judged to represent low safety significance, they are examples of conditions which need to be corrected.

GPU Nuclear offers the following additional information regarding the SSFI items listed in the inspection report:

1. SSFI Items 212-1 through 11: These items represent problems associated with locating design information, consistency between calculations and plant configuration, and the thoroughness of documenting the basis of design decisions. Many of these relate to work performed in the mid 1980s. The initial review found these to have low safety significance, and as these items are being completed, we are finding that the initial assessments in 1992 are being confirmed. The physical plant design has proven to be conservative.

2. SSFI Items 212-12 and 13: These items have been resolved and closed with no safety impact or concerns. Delay in completing the NPSH calculation, once the required information was available, was due to resource limitations.

3. SSFI Item 212-42: This item resulted from not researching design information sufficiently to understand the design basis of Decay Heat Pump vent valves. An early assessment indicated these should be in the EQ program, however later paperwork (not located during the Inspection) and assessment has indicated this was not warranted. These valves were put in the EQ Program to be

conservative and remain since it is beneficial, however they are not required to satisfy 10 CFR 50.49 criteria involving electrical equipment.

4. SSFI Item 212-61: This item is discussed as OI 96-201-16. With regard to Criterion 16, this is an example of inconsistency between the Technical Specifications and the FSAR and is a condition adverse to quality that was not closed in a timely manner.

GPU Nuclear has had an extensive program of self-assessment through SSFIs and Service Water System Operational Performance Inspections (SWSOPIs) as well as design basis reconstitution through the preparation of Design Basis Documents (DBDs). These efforts, from 1989 to 1996, identified over 450 issues that involved some form of follow-up action. Approximately 400 items have been closed with the balance being managed through the Corrective Action Process (CAP). GPU Nuclear places great value in finding and resolving issues as discussed at the Predecisional Enforcement Conference. Placing the items in the new CAP Process was an action initiated prior to Inspection 96-201. Improving the management of issues was one of the reasons for initiating the major organizational changes in engineering in 1996. The change accomplished many objectives such as establishing a programs group to better focus attention and resources on Configuration Management and in following up activities on various initiatives like DBDs and SSFIs.

The results of work on the open items so far indicates that these represent issues of effective closeout tracking and documentation; there appears to be no substantial impact on nuclear safety associated with the delayed completion of these particular items. Based on the work completed to date, there is nothing to challenge the conclusions of the original evaluations of the safety significance of these issues.

To promote timely response to action items (such as those cited in URI 96-201-18), all open items from the System Design Basis Document (SDBD) program and the SSFI and SWSOPI Inspections (a total of 65 items) have been entered into the Corrective Action Process (CAP) as committed to in the response to the NRC's 50.54(f) letter dated October 9, 1996. All open items entered into the CAP and the Engineering Division Task Tracking System will be tracked to completion.

C. Corrective Actions:

SSFI Items 212-12 and 212-13 are complete. SSFI Item 212-42 is complete. Valves DH-V-75A&B and DH-V-76A&B were added to the EQ List as of February 5, 1997.

D. Schedule for Completion of Corrective Actions:

1. SSFI Items 212-1 through 212-11. Those items that are not already closed are targeted for completion by December 31, 1997. GPU Nuclear is in the process of contracting out work on several of these items. Therefore, the target date of December 31, 1997 could slip if external resources are unavailable.

2. SSFI Item 212-61 will be completed as described in response to OI 96-201-16.

IFI 96-201-19 "Evaluation of Reactor Building Sump Screen for Reverse Flow (Section E1.3.2.2.h)"

A. Description of the Finding

The team questioned whether the Reactor Building (RB) sump screens had been analyzed to verify that they were designed to withstand pressures due to reverse flow when the Reactor Coolant System (RCS) drop line is opened during post-accident conditions to control boron precipitation in the reactor core.

B. Discussion:

This concern results from opening the Decay Heat drop line to the RB Sump during a large break Loss of Coolant Accident (LOCA) as a means of avoiding core boron precipitation.

C. Corrective Actions:

1. Analysis is being performed to determine the maximum pressure the RB Sump screens can withstand if the RCS drop line were opened during post-accident conditions to control boron precipitation in the reactor core.

D. Schedule for Completion of Corrective Actions:

1. The current phase of the analysis is expected to be completed by July 31, 1997. Depending on the results, additional work could be required.

URI 96-201-20 "Testing of Molded Case Circuit Breakers (Section E1.3.3.2.a)"

A. Description of the Finding:

Molded case circuit breakers in two safety related Motor Control Centers (MCCs) for ventilation systems used during refueling had not been periodically tested since their installation in 1986 because they were not included in the original maintenance database until 1993. They are now scheduled to be tested in 1997 and 1998. Also, in 1993, the testing frequency for a feeder circuit breaker in 1B Emergency Safeguards (ES) Valves MCC Unit 7A for the 1B ES Engineered Safety Features (ESF) Vent was deleted from GMS-2. As of this inspection the circuit breaker in the 1B ES Valves Unit 7A was still within its four year maintenance cycle.

CAP T1997-0006 was written to address the failure to include these breakers in the Preventive Maintenance (PM) program. The 1A & 1B ESF Vent MCC feeder breakers and all breakers supplying ESF Ventilation system loads were tested during February 1997.

B. Discussion:

GPU Nuclear concurs that the ESF Vent MCC molded case circuit breakers were omitted from the PM program. The 1A and 1B ESF Vent MCCs and associated molded case circuit breakers were not added to the PM Program when they were installed in 1986. With the change from GMS-1 (a task oriented system) to GMS-2 (a component oriented system) at the end of 1989 through the beginning of 1990, discrete components (circuit breakers, motors, pumps, etc.) were loaded into the GMS-2 component database. The correct component-specific tasks were then established to test these individual circuit breakers. As part of the task restructuring project, however, the task to te the breaker for 1B ESF Vent MCC Feeder breaker (1B ESV Unit 7A) was inadvertently canceled.

The loads powered from the 1A & 1B ESF Vent MCCs are non-NSR loads. The ESF Ventilation System is classified as Regulatory Required (RR) and the system is shutdown by an ES signal. The Borated Water Storage Tank Heaters (BW-H-0001 & BW-H-0002) are also feed from the ESF Vent MCCs. These heaters are tripped and locked out by an ES signal concurrent with a loss of offsite power. Components powered from these MCCs are not required for ES conditions. The molded case breakers feeding the MCCs were always within their test interval. The breakers were operable to isolate the ESF Vent MCCs from either the 1A or 1B ES Valves MCC in the event of a fault on the 1A or 1B ESF Vent MCC. Nuclear Safety was not impacted since an electrical fault on a non-Nuclear Safety-Related (non-NSR) component powered from either ESF Vent MCC would have been isolated by a tested NSR breaker.

In 1988, the Maintenance Assessment group was formed and this review was assigned as part of that group's responsibilities. This resulted in the process becoming more formalized. Specific individuals are now required to perform this review function, as outlined in attachments to Administrative Procedures (APs) 1021 and 1043 for modification closeouts; omissions such as this are less likely to occur with the process in place today.

C. Corrective Actions:

1. A review was conducted of the molded case breaker PM tasks to ensure that all NSR MCC molded case breaker are included in the test program. Twenty breaker tasks were updated as a result of this review which was completed on June 3, 1997.

2. Tests have been completed on the three breakers (BW- FH-1, BW-H-001 and BW-H-002) which are powered from the 1A & 1B ESF Vent MCCs. These tests had not been completed by the close of the design inspection.

3. A GMS-2 Task to test the molded case feeder breaker to the 1B ESF Vent MCC has been re-established.

D. Schedule for Completion of Corrective Actions:

1. Review of the Nuclear Safety Related (NSR) MCC breaker PM tasks will be completed by July 31, 1997. The other actions described above have been completed.

URI 96-201-21 "Static Head Correction for Borated Water Storage Tank (BWST) Level Transmitter (Section E1.3.4.2.a)"

A. Description of the Finding:

Surveillance Procedure (SP) 1302-5.19, "Borated Water Storage Tank Level Indicator," data sheets show the head correction (due to elevation differences between the level transmitters and the bottom of the BWST) and span correction (due to the specific gravity of borated water) for the BWST level transmitters. No engineering documentation was found that details the assumptions and references used to establish the calibration data.

B. Discussion:

Although the span and head correction values in the procedure are correct, the lack of a reference to a source document for these values is a weakness in the procedure. No formal setpoint calculation exists for the BWST level indication loops. A setpoint calculation would include the information, references, and sketches necessary to define the basis for the setpoint. Without the setpoint calculation, alternate sources were used to establish the calibration data in the surveillance procedure as discussed below.

Field sketches show the proper head correction. Also, the span and head correction information used in the surveillance procedure was obtained from engineering evaluations that show the proper head correction. However, the span correction value used in these evaluations did not include a source reference; and the head correction information from the field sketches in the engineering evaluations was not incorporated into appropriate design documents, Piping and Instrumentation Drawings (P&IDs).

The SP did not contain any errors related to head or span correction. This finding involves maintaining proper documentation of the basis for the values being used.

In order to assure that any potentially similar problems with other process measurements and setpoints are addressed, plans for a setpoint basis update program was initiated in response to the NRC's 50.54(f) letter of October 9, 1996. This program will provide a data base for reference to the design basis document (i.e. setpoint calculation) for each Nuclear Safety Related (NSR) and Regulatory Required (RR) setpoint. If a design basis document can not be found, an engineering task will be initiated to have the setpoint design basis established. Once a design basis document is established, the document will be referenced in the setpoint program database as a source document. This program will ensure that design basis references for setpoints are captured and easily retrievable.

C. Corrective Actions:

1. Tasks were initiated during the inspection to track completion of the update documentation related to DH-LT-0808 and DH-LT-0809 including:

   a. Drawings B-308-810 and D-308-920 were updated to show the transmitter and BWST elevations. This provides the basis for the head correction for each transmitter.

   b. The GMS2 database was updated to reference the change document, correct drawing references, and update the range for level transmitter, DH-LT-0809.

   c. The vendor manual was added as a reference to the SP as the basis for the span correction. The updated Piping and Instrumentation Diagrams (P&ID), D-308-920 and B-308-810, were also included as references in the procedure.

2. SP 1302-5.19, "Borated Water Storage Tank Level Indicator," was changed in Revision 19 on February 5, 1997 to incorporate the appropriate changes.

D. Schedule for Completion of Corrective Actions:

Except for the long term commitments in response to the NRC's 50.54(f) letter of October 9, 1996 referred to above, the other actions described above to address the BWST level transmitters span and head correction have been completed.

IFI 96-201-22 "BWST Level Instrument Drift (Section E1.3.4.2.b)"

A. Description of the Finding:

After the review of several instrument loop accuracy calculations, the team noted inconsistent treatment of drift errors in these calculations. For example: calculation C-1101-662-5350-059, "LOOP Accuracy R.G. 1.97 BWST Level Incorp. Surveillance LOOP Tolerance (MTX)," Revision 0, which established the loop accuracy of Borated Water Storage Tank (BWST) level instrumentation stated an arbitrary value for drift as one-half of the accuracy of the loop components; calculation C-1101-212-5350-051, "BWST Level Instrument Drift," Revision 0, empirically determined drift; GPU Nuclear Engineering Standard ES-002, Revision 4, "Instrument Error Calculation and Setpoint Determination," addressed drift as a variable to be considered while performing error analysis, but it did not offer any guidance on how to calculate drift; and calculation C8706-021 "R.G. 1.97 RMT Transmitter Loop Accuracy" did not address drift error in the loop analysis, but formed the initial basis for the loop accuracy of the BWST level instruments.

Licensee memo 5450-88-0023 in calculation C8706-021, stated that the BWST level instrument accuracy requirements were on hold pending completion of two other evaluations. The memo also stated that the basis for the loop accuracy requirements was included in Technical Data Report (TDR) No. 883, Revision 1. The team reviewed this TDR and did not find the loop accuracy bases.

The team noted instrument loop error concerns similar to the ones discussed in OI 96-201-11. Calculation C-1101-662-5350-59 stated that the existing surveillance tolerance for LT-808 and LT-809 was ± 2%. A memo dated August 8, 1988, attached to the calculation stated that to be consistent with the usual approach which provided margin and allowed for additional drift, the surveillance accuracy should be lowered to 1.5%. The memo stated that supporting calculations were being developed. GPU Nuclear was unable to provide these calculations. The discussion in the memo was not consistent with the present surveillance data sheets which specify a 1.0% loop accuracy which are more restrictive. Additionally, safety evaluation SE-000-214-001, Revision 2, Section 3.3.2.1(d) stated an allowance of ± 2% for level error for the low-low alarm generated by BWST level transmitters DH-LT-808 and DH-LT-809. These inconsistencies had not been resolved by the end of the inspection.

B. Discussion:

GPU Nuclear is proceeding with the actions described below.

C. Corrective Actions:

1. GPU Nuclear Engineering Standard ES-002, Revision 4, "Instrument Error Calculation and Setpoint Determination" will be used to develop a new calculation for BWST level instrument errors. Drift will be determined from the manufacturer's specifications. The "as-left" calibration tolerance limit will be established by the calculation. The following explains the disposition of documents upon issuance of the new BWST level calculation:

   a. Calculation C-1101-662-5350-059, "Loop Accuracy RG 1.97 BWST Level Incorp. Surveillance Loop Tolerance (MTX)," Revision 0, will be superceded by the new calculation.

   b. Calculation C-1101-212-5350-051, "BWST Level Instrument Drift," Revision 0, will be an input to the new calculation.

2. Those portions of the architect engineer, Gilbert Associates Inc. (GAI), calculation C8706-021, "TMI-1 New RG 1.97 RMT Transmitter Loop Accuracies," that are related to BWST level will be superceded.

Note: TDR 883, Revision 1, "TMI-1 Equipment Qualification - Performance Evaluation of Instrument Loops," remains unaffected by the issuance of the new calculation since it does not address BWST level.

D. Schedule for Completion of Corrective Actions:

1. Actions required to complete the new calculation are being tracked as task #2162 which is scheduled to be completed by July 30, 1997.

2. GAI calculation C8706-021 is scheduled to be revised by October 31, 1997.

URI 96-201-23 "Selection of BWST Low Level Alarm for Operator Action (Section E1.3.4.2.c)"

A. Description of the Finding:

The initiation of the switchover to sump recirculation was accomplished by a Borated Water Storage Tank (BWST) low level alarm from switch DH-DPS-914. This action is a critical operation. The redundant action initiator is considered computer alarm A0486 from DH-LT-0809. The NRC inspection team was concerned that critical operations were being initiated utilizing non safety-related alarms.

B. Discussion:

The preparation and review process of the Temporary Change Notices (TCNs) that initiated the switchover of Emergency Core Cooling System (ECCS) pump suction to the Reactor Building (RB) sump from the BWST utilized control room overhead annunciator alarms. Standard operator direction is to use redundant instrumentation where available which includes safety grade console indicators.

C. Corrective Actions:

1. Abnormal Transient Procedures (ATPs) 1210-6, "Small Break LOCA Cooldown," Revision 23 and 1210-7 "Large Break LOCA Cooldown," Revision 23 were completed on March 19, 1997 to specifically include the use of console indication for the initiation of ECCS suction switchover.

2. Operator training materials will be revised to reflect operator action based on the console level indication.

D. Schedule for completion of Corrective Actions:

1. Operator Training lesson plan materials will be revised and any additional training that may be necessary because of these changes will be completed by December 31, 1997.

IFI 96-201-24 "Undocumented Modification (Section E1.3.6.2.b)"

A. Description of the Finding:

On December 19, 1996 during a walkdown of the Low Pressure Injection/High Pressure Injection (LPI/HPI) Systems the inspection team observed a hose and metal tubing/fittings attached to the open end of discharge relief valve BS-V-63B. The team questioned whether this was an acceptable configuration.

B. Discussion:

GPU Nuclear immediately recognized and informed the NRC inspection team that this was not an acceptable configuration (prohibited by ASME code) and one that was probably not reviewed or approved by site engineering. At the conclusion of the walkdown, GPU Nuclear verified that the configuration was not previously reviewed by engineering and removed the attachments to the relief valve. A corrective action task was initiated to develop any required actions to prevent similar occurrences.

GPU Nuclear recognized that this attachment to the relief valve is not allowed by General Design Requirements (USAS B31.7 Nuclear Power Piping Section 1-722.6) and that this condition could cause a backpressure on the relief valve, or fluid buildup on the discharge that would affect the lift setpoint of the relief valve. There are no normal system conditions or analyzed accidents that create the condition where this valve would be expected to lift.

C. Corrective Actions:

1. An Awareness Memorandum was prepared which included the event description, consequences, root cause, and corrective actions. The memorandum became required reading for all Operating Crews and Operations Management. As part of the corrective action, Shift Supervisors were required to discuss the following items with the crews:

   a. If this type situation is known or seen in any other area of the plant the unauthorized equipment is to be removed.

   b. If there are known problems with relief valves lifting early or leaking by, this information is to be addressed by job ticket to correct the deficient condition.

   c. If there are specific valves which operators believe need a tail pipe, they are to notify the Shift Supervisor/Operations Engineering so that an appropriate solution can be obtained.

2. Although not part of the official closeout of the corrective action task, the Awareness Memorandum was routed as required reading to all TMI System Engineers who routinely perform system walkdowns.

D. Schedule for Completion of Corrective Actions:

All follow-up action is complete.

URI 96-201-25 "BWST Level Transmitter Enclosure and Heat Tracing (Section E1.3.6.2.c)"

A. Description of the Finding:

The team noted that the cover plate of the enclosure for Borated Water Storage Tank (BWST) level instrument DH-LT-808 was open and the fasteners for the cover were missing. No work was in progress that could explain the reason for the cover being open. The enclosure is a metal barrier that provides separation between redundant transmitters and process tubing. Therefore, the enclosure was installed in accordance with Section 3.3 of SP-9000-44-001, "Instrument and Control Instrument Installation," Revision 0, which specifies the use of protective barriers where separation criteria could not be met.

The team also observed that the heat tracing for BWST safety-related level instrument DH-LT-808 was left coiled within the sheet metal enclosure and not wrapped around the sensing lines or the transmitter. This configuration was not in accordance with the vendor drawing (ET-30250, Revision 2) and maintenance procedure 1420-HT1, "Heat Trace Repair and Replacement," Revision 11. Although there had been no history of freezing of the transmitter or its associated tubing, the team was concerned that the transmitter had the potential to freeze if left in the as-found condition.

B. Discussion:

The configuration was non-standard relative to applicable documents: Nelson drawing ET-30250R2 and TMI heat trace maintenance procedure 1420-HT1. The enclosure is provided for physical separation and not for thermal insulation. Recent TMI heat trace design is governed by a vendor heat trace program. This program is designed to correspond to standard configurations (trace applied to pipe/tubing and with some minimum amount of insulation) and thus may not produce correct results for the subject application.

C. Corrective Actions:

1. The cover plate was reinstalled.

2. Surveillance Procedure (SP) 1302-5.19, "Borated Water Storage Tank Level Indicator," was revised in Revision 19 on February 5, 1997 to ensure that the enclosure cover is closed after completion of the surveillance.

3. The heat trace installation has been corrected.

D. Schedule for Completion of Corrective Actions:

No additional action is required.

URI 96-201-26 "FSAR Discrepancies (Section E1.3.7)"

A. Description of the Finding:

The team reviewed the appropriate FSAR sections for the Decay Heat Removal (DHR) System and for the associated electrical and instrumentation and control systems.

The team identified the following discrepancies in the FSAR:

1. FSAR Section 6.1.2.1.b.1 stated that the maximum flow through the DHR pump bypass line is 125 gpm ± 5 gpm when operating at a shutoff head of 425 feet. Also, the system design basis document SDBD-T1-212, Revision 1, stated the same recirculation flow rate. The flow as determined in calculation C-1101-212-5360-008, Revision 0 was 150-155 gpm.

2. FSAR Table 9.5-2 contained several entries of two design temperatures and/or pressures for DHR pumps and coolers. The licensee stated that these were incorrect.

3. FSAR Section 8.2.2.10.g stated that non-segregated, metal enclosed 4160 V bus ducts were used for major circuit runs from the unit auxiliary transformers to 4160 V and 6900 V buses. The equipment bill of materials showed a voltage of 7.2 KV for the bus ducts.

4. FSAR Tables 8.2.8 and 8.2.9 (Emergency Diesel Generator loading) referred to the Borated Water Storage Tank (BWST) heat tracing load as BS-T-2B, however, this load designation was for the sodium hydroxide tank heat tracing.

B. Discussion:

The resolution of these discrepancies requires a revision to the FSAR. The GPU Nuclear process for FSAR updates requires that a Preliminary FSAR Update (PFU) be submitted. A PFU includes the recommended text revision and the required safety review. PFUs have been submitted for all of the discrepancies identified.

Section 6.1.2.1.b.1 indicated a Decay Heat (DH) Pump recirculation flow of 125 gpm vs. 150 gpm. Analyses of Low Pressure Injection (LPI) or DH System capability have all used an assumed recirculation flow of 150 gpm. Table 9.5-2 contained erroneous DH pump and cooler component data. The errors on this table can be categorized as conservative (cooler shell side design temperature inaccurately shown as 200 vs. 250 F) or obvious errors (two different values were indicated for DH pump design temperature; the table indicated that nuclear services closed cooling cooled the DH coolers). Section 8.2.2.10.g indicated that 7200V bus were run in 4160V bus ducts. This error was as an obvious documentation error. Tables 8.2.8 & 8.2.9 erroneously referred to BWST heat trace as BS-T-2B heat trace. The table was used to tabulate Emergency Diesel Generator (EDG) loads. The values used were correct. The error only affected the name associated with the load.

In September 1996, a new emphasis was placed on the quality and completeness of the FSAR. FSAR section owners were established as a step to improve the quality.

C. Corrective Action:

1. As stated in the inspection report, PFU 98-T1-126 and PFU 98-T1-129 were prepared to resolve these discrepancies. PFU 98-T1-139 and PFU 98-T1-127 have also been prepared to resolve these FSAR discrepancies. It is noteworthy that after a PFU has been submitted, the change is placed in a database for immediate use by reviewers in performing safety evaluations.

D. Schedule of Corrective Action:

1. The above listed PFUs will be included in FSAR Update 14, which is currently due in April 1998.

IFI 96-201-27 "Definition of Single Active Failure (Section E1.3.7)"

A. Description of the Finding:

Section 4.1.3.7, "Single Failure," of System Design Basis Document (SDBD) T1-212 states that the definition of active component has been interpreted to exclude self-actuating components for which there is adequate positive force to assure they function (i.e., check valves). This definition and its application in safety-related systems had not been resolved by the end of the inspection.

B. Discussion:

For the design and licensing of TMI-1, Emergency Core Cooling System (ECCS) check valves were treated as passive components. The ECCS Single Failure Analysis in FSAR Section 6.7.3 states "the ECCS mechanical design adequacy is dependent on the following assumptions: (1) flow check valves assume the proper open or closed position when required; (2) pressure relief valves assume the proper open or closed position when required. Components which are assumed to operate and are excluded from "single failure consideration are specifically required to be operable at all times without exception in the Technical Specifications.

The Design Basis Document (DBD) discussion is specific to "single failure" application for ECCS analysis. Check valves are tested as required by the TMI-1 Technical Specifications and ASME Boiler & Pressure Vessel Code, Section XI.

C. Corrective Actions:

No corrective action is required.

URI 96-201-28 "Control of Calculations (Section E1.4)"

A. Description of the Finding:

Design Control - Control of Calculations - The inspection report cited two design control concerns related to control of calculations. The inspection team identified that memoranda, Technical Data Reports (TDRs) and plant Engineering Evaluation Requests (EERs) were used to perform safety related calculations, which did not comply with GPU Nuclear procedure EP-006 and could result in the requirements for verifying and checking of design work not being met. The second concern was that several calculations and analyses reviewed by the inspection team were not the latest documents because subsequent calculations performed the same or very similar analyses. The older analyses were not identified as superseded. Additionally, changes to data from other sources that were used as input to calculations were not consistently incorporated into the completed calculations.

B. Discussion:

GPU Nuclear concurs with this finding and has initiated a series of actions to address the concerns regarding the control of calculations as presented in the inspection report.

Following NRC identification of the issue, GPU Nuclear promptly issued a Quality Deficiency Report (QDR). GPU Nuclear immediately provided supplemental training to the engineering staff that reinforced procedural requirements for the performance of calculations. Additional actions were taken to determine the root causes of this condition and to evaluate and enhance the existing process. GPU Nuclear has initiated an extensive review to identify design basis calculations for key system parameters. GPU Nuclear has also undertaken an effort to review memos, EERs and TDRs to determine the extent to which design basis calculations may have been documented outside of the established procedural requirements.

The initial root cause evaluation has been completed. The results indicate that both process and human performance issues contributed to this concern. Additional contributing factors have also been identified. The process enhancements currently underway address these items. The impact of management factors on this process is also being evaluated. The TDRs have been screened and final reviews are in process. Sample reviews of memoranda and EERs are also being performed at this time.

Previous observations have, for the most part, indicated that GPU Nuclear calculations are detailed and technically sound. Initial reviews are demonstrating that there is not a more widespread problem with the calculation procedure than that which has been identified. The implementation of process enhancements, database improvements and reinforcement of procedural compliance that are planned will further strengthen the GPU Nuclear calculation process.

C. Corrective Actions:

1. Supplemental training on the current procedure has been provided for the engineering staff.

2. GPU Nuclear has completed the initial identification of the root causes and other factors that contributed to calculation process issues.

3. An evaluation is being prepared which highlights the impact of management factors that may have contributed to the deficiencies noted in the inspection report which were related to the calculation process.

4. A new calculation process is being implemented. This will include procedural improvements along with training on the changes.

5. The extent of non-compliance with the calculation procedure is currently being assessed and the conditions of non-compliance that are identified will be corrected.

D. Schedule for Completion of Corrective Action:

1. An evaluation of management's impact on the calculation process deficiencies is scheduled for completion by August 1997.

2. Procedural changes, including training on the changes, is scheduled for completion by September 1997.

3. The Assessment of non-compliance with the calculation procedure is scheduled for completion by June 30, 1997. These will be evaluated individually and brought into compliance in a time frame commensurate with their impact on safe plant operation.