U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Licensee:	GPU Nuclear Corporation
Facility:	Three Mile Island Station, Unit 1
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EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/98-06
August 10, 1998 - August 27, 1998

The inspection examined engineering activities associated with findings from the NRC Architect Engineering (AE) Team Inspection that was completed in January of 1997 and documented in NRC Inspection Report 50-289/96-201. The inspection also examined the effectiveness of the corrective actions taken to address the violations stemming from the issues identified by the AE team and documented in the Notice of Violation and Proposed Imposition of Civil Penalties dated October 8, 1997.

- Although several iterations were required, engineering personnel addressed the issue of a letdown line break outside containment and verified that safety related equipment in the auxiliary building, would remain operable following a letdown line break. (E8.1)

- The concerns with the high pressure injection system involving coordination of the opening of the suction and discharge valves with pump MU-P-1C operation; potential for gas accumulation in the pump suction piping; and the stroke times for system valves specified in surveillance procedure 1300-3H, "IST of Makeup Pumps and Valves," were resolved. (E8.2, E8.3, E8.5)

- A violation of NRC requirements was identified with the changes made to the position of the cross connect valves. The safety evaluation failed to identify that the change involved an unreviewed safety question, as the change introduced a malfunction of a different type than any evaluated in the UFSAR; i.e., a failure of the "C" HPI pump due to gas entrainment. (E8.4)

- Engineering personnel adequately demonstrated that during a seismic event, the auxiliary steam piping will not fail. No physical changes were required to arrive at this conclusion and the issue did not represent any past operational concern. (E8.6)

- The issues associated with makeup pumps' net positive suction head (NPSH) were resolved. If the makeup tank is depressurized, or if there is a low-low level in the borated water storage tank, adequate NPSH will still be available for the pumps. (E8.7, E8.8)

- Voltage drop analysis and calculations showed that safety related equipment, including MU-V-18, would have sufficient voltage to perform their design function. (E8.9)

- Operation of makeup system valve MU-V-18 prior to January 8, 1997, was not appropriately controlled by procedures. Because it did not represent a more broad concern with respect to establishment of procedures for safety activities, this minor violation was not subject to formal enforcement action. (E8.10)

- Inconsistencies in data sheets of calibration surveillance procedures, such as instrument uncertainties, static head corrections and instrument loop errors were adequately resolved. (E8.11, E8.21, and E8.22)

- FSAR and technical specification discrepancies identified by the AE team were properly resolved and the ongoing efforts by the licensee to improve the overall quality of the FSAR appeared appropriate. (E8.12, E8.15, E8.16, and E8.26)

- Adequate corrective actions were taken to address the issues associated with potential air entrainment into the decay heat removal system pump suction during the switch from the Borated Water Storage Tank to the Reactor Building sump. (E8.13)

- The corrective actions taken to address the violation stemming from improperly taking credit for containment overpressure in low pressure injection (LPI) pumps NPSH calculations were appropriate. New calculations reflected that adequate NPSH would be available for the LPI pumps without taking credit for containment overpressure. (E8.14)

- Programmatic issues such as inservice testing program discrepancies involving DHRS valves, and Corrective Action Program deficiencies with tracking open items associated with safety system functional inspection (SSFI) and service water system operational performance inspection (SWSOPI) activities were resolved. (E8.17a, 17b, 17c, and E.18)

- Engineering evaluation confirmed that there was no safety concern associated with reverse flow through the reactor building (RB) sump screens. (E8.19)

- The discrepancy associated with the use of non safety-related annunciation to accomplish safety related actions by operators was appropriately resolved. Good procedural guidance and training were provided. (E8.23)

- The actions taken to correct the improper attachment of a hose and metal tubing to the discharge of relief valve BS-V-63B were adequate. This minor violation was not subject to formal enforcement action. (E8.24)

- Appropriate actions were implemented to ensure that the BWST level transmitter enclosure and heat tracing were maintained properly. (E8.25)

- The actions taken and those ongoing to identify and control calculations/analysis were adequate. Enhanced procedure, EP-006, "Calculations," was comprehensive and contained good guidance for controlling calculations. (E8.28)

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III. Engineering

E8 Miscellaneous Engineering Issues (92903)

Between November 11, 1996 and January 10, 1997, an NRC Architect Engineering (AE) Team conducted a design inspection at TMI-1. The results were documented in NRC Inspection Report 50-289/96-201. The team identified twenty eight items as either Unresolved or Inspector Follow up Items (URIs or IFIs). Some of the items were later identified by the NRC as violations of NRC requirements and transmitted to GPU Nuclear in a letter dated October 8, 1997. This inspection was conducted to review the resolution of the remaining open items and the corrective actions taken to address those already identified by the NRC to be violations. The results of these reviews are documented below.

E8.1 (CLOSED) IFI 50-289/96-201-01, Letdown line Break in the Auxiliary Building

a. Inspection Scope

The AE team identified a discrepancy involving the licensee's consideration of the effects of a letdown line break in the auxiliary building. It was not clear to what extent the licensee was required to consider the effects of a letdown line break in the auxiliary building.

b. Observations and Findings

At TMI-1, high energy lines are defined as piping containing fluids that exceed both 200°F and 275 psig. The reactor coolant system (RCS) letdown line in the auxiliary building normally operates below 200°F and therefore, based on the existing licensing basis definition, it appeared that full diameter breaks did not have to be postulated, although the pressure exceeds 275 psig. However, the licensing basis for TMI-1 included the licensee's response to the NRC's request for additional information (RAI) involving the licensee's evaluation of six full diameter letdown line break locations described in the FSAR, Appendix 14A. Therefore the AE Team determined that the TMI-1 licensing basis for pipe breaks included the postulation of full diameter letdown line breaks between the containment penetration and the breakdown orifice as described in Appendix 14A of the FSAR. Consequently, the design of safety related equipment in the affected areas should consider the conditions resulting from these breaks.

Following further reviews, GPU Nuclear agreed that high energy line break (HELB) consideration for the letdown line outside containment was part of the licensing basis for TMI-1 and therefore initiated actions to complete a break analysis for the line.

The inspector reviewed the licensee's operability determination (PRG No. 97-072 and 98-22 meeting minutes) asserting that equipment in the pipe area would not be adversely affected by a line break. The inspector identified weaknesses and errors in the determination. For example, the bases for equipment operability were unclear. The licensee entered this issue into their corrective action program (CAP No. T1 98-0666) to address the inspector's concerns. The issue was reviewed further by the licensee and a new operability determination was documented in PRG Meeting No. 98-76 (August 24, 1998). Technical Data Report (TDR) No.1230 documented the evaluations and engineering judgement made regarding the capability of TMI-1 auxiliary building safety-related equipment to remain functional when subjected to a postulated letdown line break environment. Since the letdown line break was not part of the design bases, the licensee did not include the affect safety related equipment in their Environmental Qualification (EQ) list. The inspector did not identify any discrepancy with this assertion.

The inspector reviewed operability determination documented in PRG Meeting Minutes No. 98-76 and TDR No. 1230. The licensee stated that the design basis of electrical equipment at TMI-1 is IEEE Report NSG/TCS/SC4-1 of 1969. Based on the expected service condition from a letdown line break, electrical equipment would remain operable in accordance with the IEEE report. Valves and cables were determined to be qualified to more severe conditions than would be expected from a letdown break. Since the licensee could not locate an original analysis of a break in the letdown line of the auxiliary building, the analysis contained in the TDR represented GPU Nuclear's analysis of record for letdown line break in the auxiliary building. This analysis therefore served as the basis for the statements in Amendment 41 to the operating license and now contained in the FSAR. The inspector found no discrepancy and determined that the TDR properly addressed the issue. The inspector conducted a walkdown of the area around the letdown line in the auxiliary building and verified that the safety-related equipment in the area were addressed in the TDR.

c. <u>Conclusions</u>

Although several iterations were required, engineering personnel addressed the issue of a letdown line break outside containment and verified that safety related equipment in the auxiliary building, would remain operable following a postulated letdown line break.

E8.2 <u>(CLOSED) IFI 50-289/96-201-02, Evaluation of Simultaneous Start of MU&P Pump MU-P-1C and Suction and Discharge Valves</u>

a. <u>Inspection Scope</u>

The AE team identified a discrepancy involving make up pump MU-P-1C and the opening stroke times of the suction and injection valves. The specific concern was over the effect on the pump of a slow opening suction valve and a fast opening high pressure injection valve.

b. Observations and Findings

The suction line of pump MU-P-1C was normally isolated from makeup tank MU-T-1 by valves MU-V69A and MU-V69B. It is also isolated from the BWST by motor operated valve MU-V-14B. In the event of an accident the pump and the motor operated valve receive an actuation signal at the same time. No interlocks or time delays were provided to start the pump until after the suction valve had opened. MU-P-1C would not be aligned to any water source until MU-V-14B automatically opened in response to an actuation signal.

The licensee evaluated this issue and determined that pump MU-P-1C was operable. This determination was based on a 1974 preoperational test results that included the simultaneous start of pump MU-P-1C and opening of valve MU-V-14B, and on a review of the test records for valve MU-V-14B. During the preoperational test, MU-V-14B stroke time was nine seconds. The licensee stated that the opening stroke time of valve MU-V-14B had remained at nine seconds or less, based on a review of test records. Since the pump achieved full speed within six seconds, the opening time for the suction valve was adequate to ensure flow to the pump. The high pressure injection valve in the pump discharge line was estimated to open fully in ten seconds. However, the maximum design basis stroke time for MU-V-14B opening is 22 seconds in Surveillance Procedure (SP) 1300-3H, "IST of MU Pumps and Valves."

The licensee determined that the reference stroke time of 13 seconds in SP 1300-3H would be adequate to ensure the required makeup pump net positive suction head (NPSH). Therefore, SP 1300-3H was revised to reduce the design basis opening stroke time for both MU-V-14A and MU-V-14B from 22 seconds to 13 seconds. In addition, the normal positions of makeup pump suction cross connect manual valves MU-V69A and MU-V69B were changed from closed to open. With these valves open makeup pump MU-P-1C suction would no longer be isolated from the makeup tank MU-T-1 during pump startup. However, a separate concern created by the changing of these valves' position is addressed in conjunction with Unresolved Item 50-289/96-201-04, "Adequacy of Makeup Tank Pressure/Level Curves."

The inspector determined that the specified stroke time of 13 seconds was appropriate for the suction valves. The inspector reviewed revision 46 of surveillance procedure (SP) 1300-3H, "IST of MU Pumps and Valves," and verified that the design basis stroke times for MU-V-14A and MU-V-14B opening was 13 seconds. Based on the licensee's evaluation, this change would ensure that the required makeup pump NPSH would be available under accident conditions. The inspector also reviewed drawing 302-661, "Makeup & Purification System Flow Diagram," Revision 50 and Operating Procedure 1104-2, "Makeup and Purification System," Revision 113 to verify that makeup pump suction cross connect manual valves MU-V69A and MU-V69B were normally in the open position.

c. Conclusions

The licensee ensured that upon automatic actuation of the HPI System, the opening of the suction and discharge valves would be in sufficient time to not create any adverse effect on pump MU-P-1C operation.

E8.3 (CLOSED) IFI 50-289/96-201-03, Evaluation of Gas Accumulation in Suction Piping for Makeup Pump MU-P-1C

a. Inspection Scope

The AE team identified a discrepancy involving the potential for accumulation of non-condensibles, such as hydrogen, released from the stagnant water in the suction line of makeup pump MU-P-1C because of the physical configuration of the pump's suction piping.

b. Observations and Findings

The pump's suction was normally isolated from makeup tank MU-T-1 by valves MU-V69A and MU-V69B. It is also isolated from the BWST by motor operated valve MU-V-14B. The isolated piping section was approximately 140 feet long with a long horizontal section approximately 10 feet above the makeup pump suction elevation. Other than quarterly cycling of motor operated valve MU-V-14B and ensuring that the piping was filled and vented after any maintenance work that required draining of the piping, no monitoring for gas accumulation was performed.

The licensee instituted a daily check of Pump MU-P-1C suction pressure as an interim measure during the design inspection. The licensee found that if the pressure was maintained at greater than 30 psig, there was no mechanism that could have caused the accumulation of non-condensibles in the piping. In addition to this issue, the licensee identified a condition where the isolated suction piping for the pump may have been overpressurized during testing and could similarly have been during the system response to a design basis accident. This condition was documented in Licensee Event Report (LER) 97-003-1, "Potential Over pressurization of Makeup Pump Suction Piping due to Inadequate Test and Operating Procedures," dated September 29, 1997. The LER concluded that there were no adverse safety consequences or safety implications that resulted from the condition, and that the event did not affect the health and safety of the public.

In response to these issues, the licensee changed the position of the makeup pump suction cross connect valves MU-V-69A and MU-V-69B from normally closed to open. Safety evaluation number 000211-015, "Operation with MU Suction X-connect Valves Open," was issued to support this change. The revised valves positions were shown on drawing 302-661, "Makeup & Purification Flow Diagram," Revision 50. As a result of this change, the suction of all three makeup pumps are normally aligned to the makeup tank.

The inspector reviewed the licensee's evaluations and discussed the issue with engineering personnel. The inspector determined that opening the pump suction valves resolved the gas accumulation concern. The inspector reviewed drawing 302-661, "Makeup & Purification System Flow Diagram," Revision 50 and Operating Procedure 1104-2, "Makeup and Purification System," Revision 113 and verified that makeup pump suction cross connect valves MU-V69A and MU-V69B were normally in the open position. However, a separate concern created by the changing of these valves' position is addressed in conjunction with Unresolved Item 50-289/96-201-04, "Adequacy of Makeup Tank Pressure/Level Curves."

c. Conclusions

The concern witn gas accumulation in makeup pump MU-P-1C suction piping was resolved. The monitoring of the makeup pump suction pressure when the concern was raised was appropriate and changing the position of the pumps' suction cross connect valves resolved the problem.

E8.4 (CLOSED URI 96-201-04, and EEI 97-070-01023, Adequacy of Makeup Tank Pressure/Level Curves

a. Inspection Scope

The AE team identified a concern associated with the pressure and level operating limits of makeup tank MU-T-1 in calculation C-1101-211-5310-0047, "Makeup Tank Drawdown during LOCA." Specifically, there were inconsistencies in the calculation involving the makeup tank volume/height relationship, omission of some piping and fittings, use of non-conservative pump flowrates, failure to account for vortexing in the makeup tank, and failure to consider the conservative minimum BWST level. Subsequent NRC reviews determined that the discrepancies constituted a violation (EEI 97-070-01023) of 10 CFR 50, Appendix B, Criterion III, "Design Control," as documented in a Notice of Violation and Proposed Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

To correct this problem, the licensee performed new calculations, and revised applicable procedures and the FSAR. Calculation C-1101-211-E610-066, "Makeup Tank Level and Pressure Limits," was performed to account for a lower back pressure, procedural limitations to flow, and to correct the other non-conservative assumptions identified in the violation. Calculation C-1101-211-E540-085, "TMI-1 Makeup Tank Gas Entrainment Determination," addressed potential vortex formation in the makeup tank. Operating Procedures 1101-1, "Plant Limits and Precautions," and 1104-2, "Makeup and Purification System," and Abnormal Transient Procedure 1210-01, "Reactor Trip," were revised to be consistent with the new analyses. Section 6.1.3.1 of the FSAR was revised to reflect the results of the analyses. Safety Evaluation Number 000211-019, "MU Pump NPSH and Gas Binding Concerns Evaluation," was issued to support this FSAR change.

The inspector reviewed calculations C-1101-211-E610-066, "Makeup Tank Level and Pressure Limits," Revision 1; C-1101-211-E610-068, "Makeup System Model for MU Pump NPSH Calculations," Revision 0; and C-1101-211-E540-085, "TMI-1 Makeup Tank Gas Entrainment Determination," Revision 0 and verified that the issues identified during the design inspection had been resolved. The inspector also verified that the makeup tank pressure/level limits included in Figure 1 of Operating Procedure 1104-2, "Makeup and Purification System," Revision 113 were consistent with the results of the calculations, and that makeup and purification system information had been deleted from Operating Procedure 1101-1, "Plant Limits and Precautions," Revision 64. In addition, the inspector verified that Operating Procedure 1104-2, "Makeup and Purification System," required normal makeup isolation valve MU-V-222 to be locked in the throttled position, consistent with the results of calculation C-1101-211-E610-066and the results of Special Test Procedure 1-97-0060, "Special Temporary Procedure for Makeup System Changes Prior to Cycle 12," performed on October 18, 1997.

In response to the issues discussed in Sections E.8.2 and E.8.3, the licensee revised the position of the makeup pump suction cross connect valves MU-V-69A and MU-V-69B from normally closed to open, aligning the suction of all three makeup pumps to the makeup tank. Previously with the valves normally closed, only makeup pump 1C was isolated from the makeup tank while pumps 1A and 1B were aligned to the tank. One of pumps 1A or 1B functioned for normal reactor coolant system makeup. Safety Evaluation Number 000211-015, "Operation with MU Suction X-connect Valves Open," was approved on August 25, 1997, to support this change. The safety evaluation concluded that the system configuration change was within the bounds of the existing calculation and current licensing basis and could be made without prior NRC approval. However, the inspectors found that the safety evaluation failed to consider the effects of a failure of valve MU-V-14A or MU-V-14B to open. In calculation C-1101-211-E610-066, "Makeup Tank Level and Pressure Limits," Revision 1, developed in support of the safety evaluation, the licensee did not address the potential single active failure of either BWST supply valves MU-V-14A or MU-V-14B to open under a design basis accident scenario with offsite power available. The scenario would result in three makeup pumps operating with a single flow path from the BWST to the pump suction creating a faster draw down from makeup tank MUT-1. This condition became more limiting than that previously analyzed, affecting the basis for the pressure and level operating limits of the makeup tank.

After questions were raised by the inspectors, the licensee performed further analysis and determined that this single failure condition was more limiting than the previously analyzed condition, and that the pressure and level operating limits of the makeup tank would not bound all design basis conditions. Upon determining this, the licensee notified the NRC on August 20, 1998, in accordance with 10 CFR 50.72(b)(2)(iii) and later issued LER No. 98-09 on September 18, 1998. The licensee stated that if all three high pressure injection pumps were operating in the high pressure injection mode, and a single failure to open occurred to one of the two normally closed suction valves, makeup pump suction pressure would be lowered to the extent that there could be gas entrainment in the pumps' suction in the event of a large break LOCA. The licensee's preliminary review of makeup tank data determined that, for about one percent of the time during the preceding year, pressure and level were in the region where gas entrainment could occur. This potential for gas entrainment in the "C" makeup pump from the makeup tank would not have existed with the suction cross connect valves closed. Therefore, the change created the possibility for a malfunction of a different type than any previously evaluated in the UFSAR.

Later on, a problem related to this issue was identified by the licensee with the makeup tank level instrumentation. During calibration, the "as found" level instrument drift was more than expected, causing the indicated level to be higher than the actual level in the tank. The compounding effect of the instrument drift was that the potential for gas entrainment to occur increased to about three percent of the time over the preceding year. The instrumentation issue is discussed in detail in NRC inspection report 50-289/98-05.

The licensee immediately revised the makeup tank pressure and level operating limits in Operating Procedure 1104-2, "Makeup and Purification System," Revision 113, to restrict the acceptable range in an operational area that restored the original design conclusions. The licensee also stated that upon completion of the revised calculation, additional corrective actions would be taken as required.

The inspectors independently observed that the effect of the error associated with the makeup tank level and pressure limits (Revision 1 of Calculation E610-066) was to reduce design margin, resulting in less conservatism associated with prediction for gas entrainment. Calculation E610-66 represented a conservative analysis of gas entrainment from the makeup tank which did not specifically account for (1) different break sizes and RCS back pressure, (2) EOP steps to secure the HPI pumps, or (3) actual pump configuration and flow rates. The analysis was meant to bound all design basis cases, thereby minimizing the time at which BWST level is reduced to where ECCS switchover occurs. The analysis also did not credit operator actions taken in accordance with EOP 1210-7 (Steps 2.11 and 2.15) to secure makeup pumps if "throttling criteria" are met.

For the smallest large break LOCA (0.5 square foot break) of concern, the RCS depressurizes below the DHR pump shutoff head within approximately four minutes. The EOP throttling criterion of greater than 1000 gpm LPI (DHR) flow per train is met at this point. Therefore, operators following EOP's will likely secure at least one or more makeup pumps in the scenario of interest, and gas entrainment would be precluded. Further, even if operators failed to follow established EOP's, the HPI function is no longer needed for ECCS at any time after four minutes in this scenario. In either case, operators have a 20-25 minute period of time (after this EOP decision point) to secure or otherwise throttle any one makeup pump before BWST level would reach the point where gas would be predicted at the makeup tank outlet nozzle. The inspectors also considered that the configuration wherein one of the BWST MU-V14 valves fails to open would not necessarily lead to a common mode failure of all three pumps due to gas entrainment. This can be seen by a careful evaluation of the piping isometric and flow paths between the makeup tank and each pump, as well as the volumetric flow "splits" between the BWST (95%) and the makeup tank (5%) in a design basis condition. Therefore, the inspectors judged the safety significance to be limited.

The inspectors assessed the potential for gas entrainment from a risk perspective and concluded that the change in risk was low. The scenario is only of interest in certain medium sized LOCAs (and in steam generator tube rupture) where offsite power remains available, the makeup tank is outside of its operational pressure/level limits (which for the case in question represented 1-3% of the period October 1997 - August 1998), a BWST supply valve fails to open, and certain proceduralized operator actions (e.g., secure HPI, isolate tank) are not credited.

However, notwithstanding the risk and technical significance of tha change made, the approval of SE 211-15 on August 25, 1997, to cross-tie the HPI header introduced an unreviewed safety question, when the potential to fail the "C" makeup pump due to gas entrainment was created. Although the makeup tank preserve/level operational limits were revised, the unreviewed safety question still existed representing a violation of 10 CFR 50.59 (a)(2)(I). **(VIO 98-06-01)**

c. Conclusions

The corrective actions associated with the adequacy of makeup tank pressure/level curves adequately addressed the original AE team findings. However, a violation of 10 CFR 50.59 was identified with the opening of the HPI suction header cross connect valves without NRC approval. The safety evaluation failed to identify that the change involved an unreviewed safety question, as the change introduced a malfunction of a different type than any evaluated in the UFSAR; i.e., a failure of the "C" HPI pump due to gas entrainment.

E8.5 (CLOSED) URI 50-289/96-201-05,Design Basis Valve Stroke Times in Surveillance Procedure

a. Inspection Scope

The AE team identified a concern involving the stroke times associated with Makeup pump recirculation isolation valves MU-V-36&37, and injection isolation valves MU-V-16A through D. The referenced design basis stroke times for the valves in surveillance procedure 1300-3H, "IST of MU Pumps and Valves," of 36 seconds was higher than the actual design basis ECCS delay time of 35 seconds assumed in the LOCA analysis.

b. Observations and Findings

FSAR section 14.2-14 (Accident Analysis) states that the ECCS delay time (LOCA) was 35 seconds based on 1 second Calculation E610-66 represented a conservative analysis of gas entrainment from the for instrumentation lag, 10 seconds for start of the emergency power source if offsite power is not available, and 24 seconds for system response time. However, FSAR 6.1.3.1, (ECCS) states that systems respond within 25 seconds, and procedure 1300-3H had 25 seconds as criterion for the valves' stroke times consistent with FSAR 6.1.3.1.

In a recent analysis to support a request for power up rate, the licensee had concluded that 25 seconds was an acceptable system response time. The NRC issued amendment No. 203 which approved the LOCA analysis for the impending power up rate to 2620 megawatts thermal, TMI-1 2772 MWT ECCS Analysis with RELAP5/MOD2-B&W, (Doc. No. 43-10222P), Revision 0, dated March 1997. Section 4.2.1.9 (Vol. I, Large Break, Vol. II, Small Break), ECCS, of the analysis assumes a delay time of 36 seconds for HPI (and LPI) injection based on: signal generation, electrical supply startup, and initiation of the pumped injection flows. Since the approved design bases was then 36 seconds, the inspector determined that the original discrepancy was resolved and no longer a concern

The inspector reviewed surveillance procedure 1300-3H, "IST of MU Pumps and Valves," revision 49, and verified that the stroke times specified for the valves were less than 25 seconds (18 open and 19 closed) and the design basis stroke time referenced was 25 seconds. The inspector reviewed past test data and verified that the valves all stroked within 15 seconds eliminating any past operability concern.

c. Conclusions

The licensee demonstrated that the stroke times for makeup system valves specified in surveillance procedure 1300-3H, "IST of Makeup Pumps and Valves," were appropriate.

E8.6 (CLOSED) IFI 50-289/96-201-06, Consequences of Failure of Auxiliary Steam Piping

a. Inspection Scope

The AE team identified a discrepancy involving the non safety-related auxiliary steam piping in the auxiliary building. The licensee had no analysis to show that a failure of the piping due to a seismic event would not result in degradation of safety-related equipment classified for mild environment in the area.

b. Observations and Findings

To resolve this item, the licensee performed a seismic analysis of the auxiliary steam pipes. The inspector reviewed report 240046-R-001, "Seismic Verification Review of Auxiliary Steam System Piping Three Mile Island Unit 1," Revision 1, dated June 5, 1998. The report, prepared for GPU Nuclear by EQE Engineering Consultants, concluded that based on walkdowns and confirmatory analysis, the auxiliary steam piping has sufficient seismic margin to maintain pressure integrity during and following a seismic event equal to the plant safe shutdown earthquake (SSE). The seismic verification was based on a methodology analogous to that used for seismic verification of mechanical equipment under the USI A-46 program. Since no physical changes were required, the issue had not represented any operational concern. The inspectors also concluded that the methodology used provided a reasonable framework to assess the effects of a seismic event on the nonsafety-related auxiliary steam piping.

c. Conclusions

The licensee adequately demonstrated that during a seismic event, the auxiliary steam piping will not fail and since no physical changes were required to arrive at this conclusion, the issue did not represent any past operational concern.

E8.7 (CLOSED) IFI 50-289/96-201-07,Loss of Pressure in MU&P Tank due to Letdown Line Break

a. Inspection Scope

The AE team identified a discrepancy involving the available net positive suction head (NPSH) for the makeup pump. The impact of loss of pressure in makeup tank MU-T-1 on the available NPSH for the makeup pumps due to a postulated letdown line break or crack combined with the failure of the check valve in the line to the makeup tank was not clearly addressed.

b. Observations and Findings

The licensee performed an evaluation of the available NPSH for the makeup pumps if the gas pressure in the makeup tank were lost due to a pipe break This evaluation concluded that adequate NPSH would be available to operate one makeup pump, based on zero psig tank pressure and normal tank level.

The inspector reviewed the analysis that supported the licensee's conclusion and found it appropriate. The licensee stated that only one makeup pump would be required to operate during this event since the pipe break would have rendered one safety train inoperable. The inspector reviewed the possible scenarios and concurred with the licensee's determination. The inspector also reviewed the licensee's evaluation and verified that adequate NPSH would be available for a makeup pump with no gas pressure in the makeup tank.

c. Conclusions

The licensee adequately demonstrated that with the makeup tank depressurized, or with a low-low level in the BWST, there was no safety concern with the makeup pumps' NPSH.

E8.8 (CLOSED) IFI 50-289/96-201-08, M&UP Pump NPSH When Taking Suction From BWST or Makeup Tank

a. Inspection Scope

The AE team identified a discrepancy with the available net positive suction head (NPSH) for the makeup pumps when taking suction from the BWST at low-low level or during potential vacuum conditions in the makeup tank.

b. Observations and Findings

The licensee completed calculation C-1101-211-E610-066, "Makeup Tank Level and Pressure Limits," Revision 1, to address the available NPSH for the makeup pumps under accident conditions. This calculation established the required pressure and level operating limits for the makeup tank, and considered the minimum indicated BWST level of 9'-6" prior to transferring the makeup pumps to "piggy-back" mode. The licensee stated that with the makeup tank maintained in the normal operating band there was no specific mechanism or sequence of events that could result in a vacuum in the makeup tank during draw down. In addition, operator was required to open the suction valves from the BWST if the makeup tank reached 55 inches. The licensee concluded that the potential for vacuum conditions in the makeup tank was precluded by system design and operational controls.

The inspector reviewed calculation C-1101-211-E610-066, "Makeup Tank Level and Pressure Limits," and verified that adequate NPSH would be available for the makeup pumps based on the required pressure and level operating limits for the makeup tank being maintained. The inspector reviewed Abnormal Transient Procedure (ATP) 1210-6, "Small Break LOCA Cooldown," Revision 25; ATP 1210-7, "Large Break LOCA Cooldown," Revision 25; and 1210-10, "Abnormal Transients Rules, Guides and Graphs," Revision 34 and verified that "piggy-back" mode would be established prior to the indicated BWST level reaching 9'-6". The inspector also reviewed ATP 1210-1, "Reactor Trip," Revision 37 and Alarm Response Procedure D-3-2, "Main Annunciator Panel D," Revision 13 and verified that the suction valves from the BWST, MU-V-14A and MU-V-14B, would be opened if the makeup tank reached 55 inches.

c. Conclusions

The licensee adequately demonstrated that with the makeup tank depressurized, or with a low-low level in the BWST, there was no safety concern with the makeup pumps' NPSH when taking suction from the BWST.

E8.9 (CLOSED) IFI 50-289/96-201-09, Incomplete DC System Voltage Drop Calculations

a. Inspection Scope

The AE team identified a discrepancy with the voltage drop analysis for makeup valve MU-V-18. Specifically, the voltage drop analysis was not performed due to lack of design data for the valve and for other components listed in the existing 125 Vdc system calculation, C-111-734-5350-004.

b. Observations and Findings

To address this issue, the licensee re-evaluated the minimum available voltage for all the components in revision 2 of 125 Vdc system calculation, C-111-734-5350-004. The inspector noted that the licensee had obtained the required input data of some components, from manufacturers (specification data sheets) and in some cases utilized plant test data. The cable impedance data was used from design documents. The inspector reviewed the voltage drop analysis and found no problems. For example, a calculated minimum of 102 volts was available at the terminal of the makeup isolation valve MU-V-18 compared to a minimum required voltage of 90 volts specified by the vendor, under the worst case loading condition, and assuming a worst case voltage of 105 V at the battery. The inspector verified that other components such as solenoid operated relief valve RC-RV2 and main steam motor operated valves, MS-V-10A and B, had sufficient voltage to perform their intended design functions.

c. Conclusions

Voltage drop analysis and calculations showed that safety related equipment, including valve MU-V-18 would have sufficient voltage to perform their design function.

E8.10 (CLOSED) URI 50-289/96-201-10, Alignment of MU-V-18 DC Power Supply

a. Inspection Scope

The A/E team identified that, the licensee had not established any instruction or procedures to assure that the makeup system discharge cross-connect valves (MU-V-76-A,B and MU-V-77A and B) and the dc power source of makeup system valve MU-V-18 were properly aligned.

b. Observations and Findings

The team identified that on January 8, 1998, the dc power source of makeup system valve MU-V-18 was not properly aligned. With the valve mechanically aligned as a loop "A" valve, it was electrically powered from the "B" 125 VDC battery source. Under normal alignment of HPI system, MU-V-18 supply is powered from "A" dc distribution panel (1M). The licensee corrected the DC power alignment immediately, and issued a temporary order change that required periodic verification of system alignment by control room staff to assure its proper operation. Proper mechanical/electrical alignment of the valve is required because the loss of a 125 VDC power train, which would render a HPI train inoperable, could result in failure of MU-V-18 to close, thus, degrading the other HPI train. Therefore, the selection of the "A" or the "B" 125 VDC power source from the 1M, 125 Vdc distribution panel must be based on the makeup pump discharge lineup.

At that time of the AE team inspection, FSAR section 6.1.3.1, stated that in the event of a small break LOCA (SBLOCA) a 70/30 HPI flow split (70% to the reactor core, 30% out of the break) will assure adequate core cooling. This 70/30 flow split is ensured by closing air operated valve MU-V-18, which is a safety-related valve that automatically isolates on an engineered safeguards (ES) actuation signal, thus isolating the makeup line flow to HPI leg "B" downstream of the cavitating venturi. Valve MU-V-18 is powered from 125 VDC distribution panel 1M, which in turn is supplied from either of the redundant 125 VDC main distribution panels 1A or 1B via an automatic transfer switch. The transfer switch is designed to be locked out when an ES signal is received. The design of the control scheme for valve MU-V-18, is such that a loss of 125 VDC power to the valve would not close the valve if control air was not available. The valve can be remote closed if both power and control air were available. FSAR analysis also took credit for partial isolation of the makeup flow if the operator closes the non safety-related makeup control valve MU-V-17 (upstream of MU-V-18). During the A/E team inspection, the licensee explained that even though MU-V-17 was not a safety-related component it was included in the inservice test program, and its operational status was monitored continuously because the valve was used for controlling the makeup flow during normal operation.

The licensee further determined that, assuming an ES actuation concurrent with, prior to, and after the dc power loss, there was no adverse impact on nuclear safety. In the first and second cases, both trains of HPI will be initiated and then MU-V-18 fails "open" when dc is lost. The running HPI pump does not trip when dc 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 third case, the normal dc distribution panel power supply (IM) would be transferred to the alternate dc source when dc is lost and MU-V-18 will close upon ES actuation, although only one HPI pump will start in this case. Since one HPI pump is capable of providing the minimum HPI flow assumed in the ECCS LOCA analysis, therefore, there was no adverse impact on nuclear safety. The inspector reviewed this analysis and identified no concern.

Based on the above review, the inspector concluded, that prior to January 8, 1997, GPUN did not establish adequate instructions or procedure for the proper alignment of the dc power source of makeup system valve MU-V-18. As a result, the alignment of the HPI system and the dc power supply of distribution panel (1M) was not satisfactorily accomplished on January 8, 1997. This was contrary to 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

Although operation of makeup system valve MU-V-18 prior to January 8, 1997, was not appropriately controlled by procedures, it did not represent a more broad concern with respect to establishment of procedures for safety activities. This minor violation was not subject to formal enforcement action.

E8.11 (CLOSED) URI 50-289/96-201-11, Makeup Tank Level Instrument Loop Tolerances

a. Inspection Scope

The A/E team identified, several instances in which the licensee had used non-conservative instrument loop errors in data sheets of calibration surveillance procedures. The calculated allowable loop errors were more restrictive than those in the instrument loop data sheets.

b. Observations and Findings

To address this concern, the licensee revised applicable plant calculations and surveillance procedure data sheets. The inspector reviewed the updated instrument data sheets and found that the loop errors were equal to or more than the estimates in the revised calculations. For example, a specified loop error of ±1% shown in data sheet of level recorder MU-14-LR for make-up tank level instrumentation (MU-14-LT) in surveillance procedure was found conservative, compared to ±2.74% estimated in revised loop error calculation (C-1101-662-5350-049, Revision 5). Similarly, the review of other data sheet for the control room indication instrument loop indicator, MU-LI-778A, and computer point A0498 associated with instrument loop, MU14-LT, were found conservative and consistent with the revised calculation. The inspector concluded that the licensee's use of a conservative loop error considered in data sheets of established surveillance procedures in above cases was appropriate. The inspector also verified that none of the discrepancies previously identified had any effect on equipment functionality.

c. Conclusions

Inconsistencies identified in data sheets of calibration surveillance procedures were resolved. The instrument loop errors in data sheets of calibration procedures reviewed were found to be acceptable.

E8.12 (CLOSED) URI 50-289/96-201-12, FSAR Discrepancies

a. Inspection Scope

The AE team identified discrepancies in various sections of the FSAR regarding the makeup system. While the discrepancies were minor in nature, the team was concerned with the fact that they 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. Observations and Findings

In subsequent follow up, the NRC determined that, although a violation of 10 CFR 50.71(e), these discrepancies were of minor significance and the issue was therefore identified as a non-cited violation in a letter dated October 8, 1997 to the Licensee.

During this inspection, the inspector reviewed the licensee's actions taken to correct the identified discrepancies as well as those taken to prevent similar FSAR discrepancies. All the discrepancies identified in the AE team report were corrected in the April 1998 FSAR update. In a broader sense, the licensee established FSAR section owners to improve the quality of the FSAR and also transferred the ownership of the FSAR to the engineering department. The inspector did not identify any discrepancies in the portions of the FSAR reviewed during this inspection.

c. Conclusions

The licensee adequately resolved the FSAR discrepancies identified by the AE team. In addition, the ongoing efforts by the licensee to improve the overall quality of the FSAR appeared appropriate.

E8.13 (CLOSED) URI 50-289/96-201-13, and EEI 97-070-01013, Adequacy of BWST Setpoint for DHRS Pump Switch over to RB Sump

a. Inspection Scope

The AE team identified concerns associated with calculation C-1101-212-5310-050, "TMI-1 BWST Vortexing Determination," Revision 0. This calculation evaluated the DHRS pump suction transition from the BWST to the reactor building (RB) sump under accident conditions. Following further NRC review, this issue was identified as a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

The concerns included (1) failure to consider instrument uncertainty in the low-low level alarm setpoint; (2) the use of a non-conservative reactor building pressure assumption; (3) the validity of the assumed time for operator response; (4) disagreement between the valve stroke times used in the calculation and the design basis times stated in the valve test procedure; and (5) incorrect methodology used for vortex determination. Based on these concerns the team concluded that there was a potential for air entrainment in the DHRS that could have rendered the pumps inoperable.

The licensee performed an operability evaluation of the DHRS on December 20-21, 1996, concluded that both trains of the DHRS and the RB spray system were inoperable, and notified the NRC in accordance with 10 CFR 50.72. The licensee issued temporary changes to several procedures to require the operators to begin transferring the DHRS pump suction from the BWST to the reactor sump when the BWST reaches the low level alarm setpoint of 9'-6", and to complete the transfer when the BWST reaches the low-low level alarm setpoint of 6'-4". Based on the changes the DHRS and the RB spray system were considered operable.

The licensee issued calculation C-1101-212-5310-050, "TMI-1 BWST Vortexing Determination," to address the errors identified during the design inspection. In addition, the licensee revised Abnormal Transient Procedure (ATP) 1210-6, "Small Break LOCA Cooldown," Revision 25; ATP 1210-7, "Large Break LOCA Cooldown," Revision 25; ATP 1210-8, "RCS Superheated," Revision 22; and 1210-10, "Abnormal Transients Rules, Guides and Graphs," Revision 34 to require the operators to begin transferring the DHRS pump suction from the BWST to the reactor sump when the BWST reaches the low level alarm setpoint of 9'-6", and to complete the transfer when the BWST reaches the low-low level alarm setpoint of 6'-4".

The inspector reviewed Calculation C-1101-212-5310-050, "TMI-1 BWST Vortexing Determination," Revision 1 and verified that the errors identified during the design inspection had been corrected, and that the calculation inputs, assumptions, and method were consistent with the design bases.

The inspector reviewed Abnormal Transient Procedure (ATP) 1210-6, "Small Break LOCA Cooldown," Revision 25; ATP 1210-7, "Large Break LOCA Cooldown," Revision 25; ATP 1210-8, "RCS Superheated," Revision 22; and 1210-10, "Abnormal Transients Rules, Guides and Graphs," Revision 34 to verify that these procedures were consistent with the calculation. The inspector also reviewed Surveillance Procedure 1300-3B, "IST of DH-P1A/B and Valves," Revision 50 and Surveillance Procedure 1303-11.16, "DH Leak Test," Revision 35 and verified that the critical valve stroke times used in the calculation were consistent with the design basis stroke times.

c. Conclusions

The issues associated with potential air entrainment into the DHRS pump suction during switch over from the BWST to the RB sump have been appropriately resolved. The corrective actions taken to address the violations stemming from this issue were adequate.

E8.14 (CLOSED) URI 50-289/96-201-14, and EEI 97-070-01093, Adequacy of Safety Evaluation for an FSAR Change Crediting Containment Overpressure for LPI Pumps NPSH

a. Inspection Scope

The AE team identified concerns associated with calculation C-1101-212-5360-027, "NPSH Available for LPI and BS Pumps Following Large Break LOCA," Revision 0 and Safety Evaluation SE-115403-004, Revision 0. Following further NRC review, this issue was identified as a violation of 10 CFR 50.59, Changes, Test and Experiments, as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

Calculation C-1101-212-5360-027determined that the NPSH requirements for the DHRS pumps would not have been met for a LPI flow of 3300 gpm as indicated by flow instrumentation under post-LOCA conditions unless credit was taken for sub-cooling of the reactor building sump water (i.e., containment overpressure).

The NRC safety evaluation report (SER) for TMI-1, dated July 11,1973, stated that the licensee's NPSH evaluation was based on the assumption that reactor building pressure was equal to the vapor pressure of the sump water and was acceptable. In this evaluation containment overpressure was not considered.

A statement was added to FSAR Section 6.4.2, Update 9, dated July 1990, to address sump sub-cooling. It stated, "an additional calculation indicated that sufficient NPSH would be available for the maximum flows, as limited by procedure, including instrument errors, recirculation flow and taking credit for sump sub-cooling (containment overpressure)." The licensee's safety evaluation, SE-115403-004, Revision 0, failed to identify that, because the required NPSH for these pumps could not be met without taking credit for containment overpressure, a potential unreviewed safety question was involved.

The licensee issued calculation C-1101-210-E610-011, "LPI and BS Pump NPSH Available from the RB Sump Following a LBLOCA," Revision 1 to determine the maximum allowable indicated LPI and BS pump flows, considering instrument uncertainty, to ensure that adequate pump NPSH is available, with no credit taken for containment overpressure. In addition, the licensee revised Abnormal Transient Procedure (ATP) 1210-6, "Small Break LOCA Cooldown," Revision 25 and ATP 1210-7, "Large Break LOCA Cooldown," Revision 25 to limit the LPI and BS flow rates.

The inspector reviewed calculation C-1101-210-E610-011, "LPI and BS Pump NPSH Available from the RB Sump Following a LBLOCA," Revision 1 and verified that no credit had been taken for containment overpressure, and that the calculation inputs, assumptions, and method were consistent with the design bases. The inspector also reviewed related calculations C-1101-210-E610-010, "Reactor Building

Minimum Level During Recirculation Following a LBLOCA," Revision 1 and C-1101-212-5450-040, "BWST Minimum Usable Volume," Revision 2 to verify that these calculations were consistent with calculation C-1101-210-E610-011. The inspector reviewed Abnormal Transient Procedure (ATP) 1210-6, "Small Break LOCA Cooldown," Revision 25 and ATP 1210-7, "Large Break LOCA Cooldown," Revision 25 to verify that these procedures included flow limits that were consistent with the calculation C-1101-210-E610-011.

c. Conclusions

The issues associated with improperly taking credit for containment overpressure in LPI pumps NPSH calculations were resolved. The corrective actions taken to address the violation stemming from this issue were appropriate. Calculations properly reflected that adequate NPSH would be available for the LPI pumps without taking credit for containment overpressure.

E8.15 (CLOSED) IFI 50-289/96-201-15, Technical Specification Discrepancy

a. Inspection Scope

The AE team identified a discrepancy in the technical specification involving the DHRS. Specifically, TS 3.3.1.1.f stated that the two reactor building sump isolation valves (DH-V-6A&B) shall be manually or remotely operable. However, only remote operation of the valves would support the design basis assumptions for the transition time of the DH pump suction switch over from the BWST to the RB sump during post-accident conditions.

b. Observations and Findings

The licensee initiated technical specification change request (TSCR) No. 263 to delete the "manual" operation from the technical specifications (TS). In TS Amendment No. 203, the "manual" was deleted. The valve surveillance tests required that the valves be tested for remote operation and the licensee stated that TMI-1 had not been operated with these valves in a condition where they were only manually operable. The inspector verified these changes and found the licensee's actions appropriate.

c. Conclusions

The inspector found this discrepancy to be of minor significance and determined that the licensee had taken appropriate actions to correct it. This item is closed.

E8.16 (CLOSED) URI 50-289/96-201-16, Discrepancy Between FSAR and Technical Specifications Regarding DHRS Leakage

a. Inspection Scope

The AE team identified a discrepancy between Technical Specification 4.5.4, DHRS Leakage, and FSAR Section 6.4.4 and Table 6.4-3. The discrepancy was that the technical specification allowed 6 gal/hr leakage from the DHRS while the FSAR specified the leakage for the DHRS and Building Spray system as 2255 cm^3/hr (0.6 gal/hr).

b. Observations and Findings

To resolve the discrepancy, the license revised the technical specification (Amendment No. 205) and applicable plant procedures to reflect the design bases leakage number (0.6 gal/hr) specified in the FSAR. In an initial review, the licensee determined that the discrepancy was not safety significant because the dose of 0.39 rem (corresponding to the TS 6 gal/hr leakage) still represented a negligible portion of the total two-hour thyroid dose limit of 189 rem.

The inspector reviewed the changes in the technical specification and in the following procedures: SP 1303-11.16, "DH Leak Test," Revision 35; SP 1303-11.27, "MU Pump Suction Leak Check," Revision 9; SP 1303-11.32, "MU Pump Discharge and Letdown Leak Check," Revision 0; and SP 1303-11.50, "RB Spray Leak Check," Revision 6. The inspector found a discrepancy in procedure SP 1303-11.16, which still reflected an acceptance criterion of less than or equal to ($\leq$) 6 gal/hr for DHRS leakage. The licensee indicated that since the procedure would only be used during outages, the appropriate change was not incorporated yet. The inspector found this inappropriate since an approved plant procedure that was obviously in conflict with the technical specification was being maintained. The procedure was then revised by the licensee such that no specific acceptance criterion was specified but instead reference was made to verify that the leakage was in compliance with the tech specs. With this change, the procedure was in compliance with the technical specification, and further changes to the technical specification would not require any changes to the procedure. The inspector reviewed the revised procedure and identified no other concerns.

c. Conclusions

The licensee was initially untimely at correcting the discrepancy in procedure SP-1303-11.16, "DH Leak Test." Nevertheless, the discrepancy between the FSAR and the technical specification was adequately resolved.

E8.17a (CLOSED) URI 50-289/96-201-17A, and EEI 97-070-10143, Leak Testing of DHRS Pump Suction Check Valves

a. Inspection Scope

The AE team identified a concern with the licensee's IST Program, dated March 1,1994. The leak testing of decay heat (DH) removal system (DHRS) check valves DH-V-14A and DH-V-14B was not included in the program. Following further NRC review, this issue was identified as a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

These valves are in the suction lines from the Borated Water Storage Tank (BWST) to the DHRS pumps. Leakage of post-accident reactor building sump water through these valves in conjunction with leakage through the isolation valve or the failure of the isolation valve to close could have resulted in increased offsite dose.

In a licensee memorandum (5310-92-024, "DH-V-14A/B, DH-V-5A/B, BS-V-52A/B Valves), a calculation incorrectly concluded that check valves DH-V-14A and DH-V-14B did not perform a safety function in the closed position, because an incorrect value for the reactor building pressure was used. Consequently, leak testing of check valves DH-V-14A and DH-V-14B was not included in the IST program. The licensee attributed the cause of the violation to lack of familiarity and inadequate training on the requirements of procedures, as well as lack of management enforcement to fully implement procedural requirements. The licensee established interim operating restrictions on reactor building pressure, BWST temperature, and reactor building fan cooler availability on December 20, 1996, to assure that leakage through these check valves could not reach the environment. The licensee issued Surveillance Procedure 1300-3Z.5, "IST of ECCS Bypass Valves - DH Suction Valves," Revision 0 to test the leakage through check valves DH-V-14A and DH-V-14B.

The inspector reviewed Surveillance Procedure 1300-3Z.5, "IST of ECCS Bypass Valves - DH Suction Valves," Revision 0 to verify that check valves DH-V-14A and DH-V-14B were being tested in accordance with the requirements of 10 CFR 50.55a. The inspector selected some other safety related valves during the course of this inspection and verified that the valves were included in the IST program and tested appropriately for their safety functions.

c. Conclusions

Appropriate actions were taken to address IST program discrepancies involving DHRS valves. The valves in question were properly included in the IST program.

E8.17b (CLOSED) URI 50-289/96-201-17B, and EEI 97-070-08014, Leak Testing of DHRS Pump Discharge Check Valves

a. Inspection Scope

The AE team identified a concern associated with testing of the DHRS pump discharge check valves, DH-V-16A and DH-V-16B. The IST program did not include the closed function of DH-V-16A and DH-V-16B. Following further NRC review, this issue was identified as a violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," and documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

If one of the DHRS pumps failed after an accident, discharge cross connect valves DH-V-38A and DH-V-38B would be opened to provide flow through both low pressure injection paths. In this condition the discharge check valve associated with the failed pump, DH-V-16A or DH-V-16B, would have to close to ensure adequate low pressure injection flow.

The licensee determined that the check valves (DH-V-16A and B) were not included in the IST program because the determination of the safety functions for the valves was inadequate. The licensee performed a test on December 19, 1996, during the design inspection, to assure that check valves DH-V-16A and DH-V-16B seated in the closed position. Both valves performed acceptably during the test. The licensee revised Surveillance Procedure 1300-3P, "IST of Check Valves During Shutdown," Revision 20 to include testing of check valves DH-V-16A and DH-V-16B in the closed position. The licensee also performed a review of IST program scope and implementation. The results of the review did not reveal any programmatic concern with the IST program.

The inspector reviewed Surveillance Procedure 1300-3P, "IST of Check Valves During Shutdown," Revision 20 to verify that check valves DH-V-16A and DH-V-16B were being tested in the closed position. In addition the inspector reviewed Special Temporary Procedure 1-96-0060, "Closure Test of DH-V-16A/B," dated December 19, 1996 to verify that DH-V-16A and DH-V-16B seated in the closed position on December 19, 1996. The inspector also reviewed and discussed the results of the IST program reviews conducted by the licensee. No programmatic issues were identified.

c. Conclusions

Appropriate actions were taken to address IST program discrepancies involving DHRS valves. The valves in question were properly included in the IST program.

E8.17c (CLOSED) URI 50-289/96-201-17C, and EEI 97-070-08014, Inspection of DHRS Pump Vault Floor Drain Check Valves

a. Inspection Scope

The AE team identified concerns associated with testing of the DH vault floor drain check valves. The IST program and the inspection procedures for check valves did not include the ball float type check valves installed in the DH vault floor drains. Following further NRC review, this issue was identified as a violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

FSAR section 6.4.5 stated that a leak in a DH vault would 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 were two floor drains from each DH vault to the auxiliary building sump.

The licensee inspected these check valves during the AE team inspection and determined that the valves were operable. The licensee revised preventive maintenance procedure U-17, "Zurn Floor Drain Inspection," Revision 5 to include inspection of ball float type check valves. In addition the licensee stated that these check valves would be added to the IST program as augmented IST, and that Drawing 302-719, "Sump Pump & Drainage System (Reactor & Auxiliary Building) Flow Diagram" would be revised to identify these drains.

The inspector reviewed preventive maintenance procedure U-17, "Zurn Floor Drain Inspection," Revision 5; Surveillance Procedure 1300-4G, "BS & DH Floor Drain Inspection," Revision 0; and administrative procedure 1041, "IST Program Requirements," Revision 31 to verify that the DH vault floor drain check valves were included in the IST program as augmented IST. In addition the inspector reviewed Drawing 302-719, "Sump Pump & Drainage System (Reactor & Auxiliary Building) Flow Diagram," Revision 56 to verify that the DH vault drains were identified.

c. Conclusions

Appropriate actions were taken to address IST program discrepancies involving DHRS valves. The valves in question were properly included in the IST program.

E8.18 (CLOSED) URI 50-289/96-201-18, and EEI 97-070-03013, Timeliness of Action on SSFI Open Items

a. Inspection Scope

The AE team identified 15 items from a safety system functional inspection performed by the licensee in 1992 that were still open. Following further NRC review, the issue was identified as a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, and documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

The licensee determined that the main reason for the violation was management's tolerance of allowing longstanding technical issues to remain open. When the SSFI was conducted in 1992, procedural controls had not been established to ensure that such adverse conditions would be resolved in a timely manner. The licensee entered the SSFI and other open items into their corrective action program (CAP).

The inspector reviewed the licensee's tracking system and verified that all 15 SSFI open items (212-1 through 13, and 212-42 and 61) identified by the AE team had been appropriately resolved. The inspector reviewed some of the resolutions and identified no additional concerns. In addition, the inspector reviewed the licensee's actions taken to address some open items from a Service Water System Operation Performance Inspection (SWSOPI) performed in 1995. All but five of the thirty three SWSOPI open items reviewed were closed. The inspector reviewed the status of the five remaining items and determined that they were being addressed appropriately and did not present any safety concern. For example, a TDR (1190) to address potential single failure vulnerability with the Decay River/Decay Closed Cooling Water had been completed and verified by an independent group (Contractor), but had not been verified and signed off yet by management due to existing workloads and other priorities. The results showed that no single failure vulnerability existed.

c. Conclusions

Open items associated with SSFI and SWSOPI activities were appropriately entered into the corrective action program and being adequately resolved.

E8.19 (CLOSED) IFI 50-289/96-201-19, Evaluation of Reactor Building Sump Screen for Reverse Flow

a. Inspection Scope

The AE team identified a concern with the capability of the reactor building sump screens to withstand pressures due to reverse flow if the RCS drop line was opened during post-accident conditions to control boron precipitation in the reactor core.

b. Observations and Findings

At the time of the AE team inspection, the licensee stated that an analysis to determine the acceptable RCS pressure below which the drop line could be opened without damaging the reactor building sumps had not been performed and that an evaluation would be performed. The licensee performed a preliminary operability determination of this issue, documented in Plant Review Group (PRG) Meeting Number 97-046, held on June 26,1997. They concluded that there was a reasonable expectation that the potentially affected systems would remain operable.

After further investigation the licensee revised the system operation to eliminate the use of a reverse flow path through the reactor building sump screens during post-accident conditions. This change was documented by CAP Number T1997-0439, dated June 27,1997.

The inspector found the licensee's actions taken to address this issue appropriate. The inspector reviewed the initial operability determination and found that it provided reasonable basis for concluding that the affected systems remained operable. The inspector reviewed Operating Procedure 1104-4, "Decay Heat Removal System," Revision 107 to verify that the revised system operation would limit the flow through the drop line and eliminate the potential of reverse flow through the reactor building sump screens during post-accident conditions. The inspector also reviewed safety evaluation PCR 1-05-97-0519, dated October 17, 1997, and verified that these changes did not require a technical specification change or result in an unreviewed safety question.

c. Conclusions

The licensee properly documented that there was no safety concern associated with reverse flow through the RB sump screens.

E8.20 (CLOSED) URI 50-289/96-201-20, and EEI 97-070-07014, Testing of Molded Case Circuit Breakers

The AE team identified that appropriate procedures had not been followed for testing frequency of the breakers in MCCs 1A ES ESF VENT and 1B ES ESF VENT. Following further reviews, the NRC identified this issue as a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997. The licensee's corrective actions to address this issue were reviewed, found appropriate and documented in section M8.1 of NRC Inspection Report 50-289/98-02. These items were closed in that report.

E8.21 (CLOSED) URI 50-289/96-201-21, and EEI 97-070-01033, Static Head Correction for BWST Level Transmitter

a. Inspection Scope

The AE team identified a concern involving the static head correction for BWST level instrumentation. Specifically, the team questioned the bases for the static head corrections for differences in elevation between the bottom of the BWST and the level transmitters in data sheets E1-1 through 4 for level transmitters DH-CT-808 and DH-LT-209 in surveillance procedure 1302-5.19, "Borated Water Storage Tank Level Indicator," Revision 18. Subsequent NRC reviews determined that this inadequacy constituted a violation of 10 CFR 50, Appendix B, Criterion III, Design Control, as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

In response to this violation, the licensee stated that they had used a proper head correction that was measured for these level transmitters (DH-LT-0808 and -0809) The field sketches in question, included as part of the engineering evaluations (EER 88-070-E and EER 88-060-E), indicated the actual measured head correction. Also, the head correction information used in surveillance procedure 1302-5.19, was obtained from these engineering evaluations. While the EERs were reviewed and approved, the field sketches prepared as part of the EER were not incorporated into design documents which receive a separate design verification or design check.

The licensee took the following corrective actions:

- Updated borated water storage tank level transmitters drawings 308810 and 308920 on February 17, 1997 to show the transmitters and BWST elevations that form the basis for the head correction of each transmitter.

- Updated the GMS2 database to reference the change document for easy reference change document retrieval.

- Updated BWST surveillance procedure, SP 1302-5.19, on February 5, 1997, to reflect the conservative values and reference drawings.

- Issued a new loop error calculation for BWST level instrument loops (C-1101-212-E510-057), that referenced the revised P&ID drawings for the appropriate head corrections.

The inspector reviewed the revised documents and found them acceptable. The level transmitters drawings, 308810 and 308920 indicated the proper instrument level elevations. To preclude similar concern, the licensee's long term corrective action (as documented in response to the NRC's 50.54(f) letter dated October 9, 1997), stated that they plan to update the setpoint basis program and, in that program, ensure that EERs which involve setpoints would receive a design basis verification that will check design inputs. Through discussion with the licensee staff, the inspector determined that this program was progressing as established and is scheduled to be completed by February 1999.

c. Conclusions

The licensee adequately addressed the concern with static head correction for BWST level instrumentation. Appropriate corrective actions were taken to address the resulting violation and long term actions were being properly tracked.

E8.22 (CLOSED) IFI 50-289/96-201-22,BWST Level Instrument Drift

a. Inspection Scope

The A/E team inspection identified that the licensee was inconsistent in treating the instrument drift values in determining the loop errors in their BWST level instrument calculations. In addition, the licensee was unable to provide adequate bases to support the instrument drift differences used in these documents.

b. Observations and Findings

To resolve these inconsistencies involving the assumed instrument drift values in BWST level calculations and other documents, the licensee developed a new calculation C-1101-212-E510-057,"TMI BWST Level Loop Accuracy-DH-LT-0808, DH-LT-0809, DH-DPS-0914, on August 19, 1997. The licensee consolidated all the different bases applicable to BWST instruments information fragmented in other documents into this calculation to eliminate the inconsistencies.

The inspector's review of the new calculation indicated that the licensee had used the guidance of the existing GPUN Nuclear Engineering Standard ES-002, Revision 4," Instrument Error Calculation and Setpoint Determination" as bases to develop the new calculation. The inspector noted that the drift values used for the BWST devices (Rosemont transmitter, Foxboro modules, Bailey indicator, Dixon indicator and computer input) were consistent with vendor specifications. In cases where drift specification was not available from the vendor, such as Westinghouse indicator and Barton differential pressure indicating switches, historical performance data were used. The inspector's review of the as left tolerances (errors) of BWST level transmitters DH-LT-0808 and 0809 indicated no concern. All supporting bases were clearly defined, and appropriately shown in this calculation.

c. Conclusions

Inconsistencies identified in BWST level instrument documents were resolved and appropriate actions were taken to eliminate uncertainties in use of instrument drift values.

E8.23 (Closed) URI 50-289/96-201-23, Selection of BWST Low Level Alarm for Operator Action

a. Inspection Scope

The AE team was concerned that critical operations were being initiated based on non safety-related alarms. Specifically, the team found discrepancies with operator actions required in response to a low BWST level alarm from level switch DH-DPS-914. The switch was listed as a category 1 device while the control room annunciator panel was non safety-related.

b. Observations and Findings

The licensee had implemented temporary procedure changes in December, 1996, requiring operators to open valves DH-V-6A/B on a low BWST level alarm from level switch DH-DPS-914. Per GMS-2 database, this level switch was listed as a Category 1 device, but it interfaced with the non safety-related main control room annunciator panel. The panel was powered from a vital power source. To provide redundancy for the DH-DPS-914 alarm, the licensee proposed the use of plant computer alarm A0486 which had an input from DH-LT-0809 instrument loop. This loop was safety-related, however, the plant computer was not.

Also, operators initiated the switch over of emergency core cooling system (ECCS) pump suction to the reactor building sump from the BWST based on control room overhead annunciator alarms. The inspector noted that the licensee's standard operator direction is to use redundant instrumentation where available which includes safety grade console indicators. To assure adherence to the above standard direction, the licensee revised the abnormal transient procedures (ATP 1210-6 and 10), to include the specific use of console indicators for the initiation of ECCS suction switch over. In addition, training was provided to the operators as documented in the training material and established operator re-qualification schedule of cycle 97.

The inspector verified that appropriate instructions to accomplish the switch over to reactor building suction in the above procedures were incorporated, The inspector also verified that operators were verifying the BWST level indicator and low level alarms for this process as indicated on the revised procedure.

c. Conclusions

The discrepancy associated with the use of non safety related annunciation to accomplish safety related actions by operators was appropriately resolved. Good procedural guidance and training were provided.

E8.24 (CLOSED) IFI 50-289/96-201-24, Undocumented Modification

a. Inspection Scope

The AE team observed a hose and metal tubing attached to the open end of discharge relief BS-V-63B. The concern was that the attachment could adversely affect the design function of the relief because it could create a back pressure on the relief valve or fluid build up on the discharge that could change the lift setpoint.

b. Observations and Findings

The licensee removed the attachment from the relief valve and initiated actions to determine the reason for the inadequacy. It was identified that personnel had not properly adhered to the controls of administrative procedure, 1000-ADM-7350.05, "Configuration Changes." The procedure required that configuration changes be documented and reviewed by engineering. The procedure also states that the activity of temporary modifications are to be controlled by AP 1013, "Temporary Modifications and Bypass of Safety Functions." Step 2.5.3.2 of AP 1013 requires the restoration of component to its normal configuration prior to return to service. The installation or failure to properly remove the temporary equipment was contrary to the requirements of the procedures.

As part of the licensee's actions taken to correct this discrepancy, system engineers, who routinely perform system walkdowns, were informed of the circumstances surrounding this event, including the consequences and corrective actions, and management expectations. Supervisors were required to review the event with their shift to heighten personnel awareness to the requirements associated with relief valves and attachments to valves in general.

The inspector verified that this review had been accomplished with all five crews as documented in operations form, Capture No. 96-357. The inspector determined that the consequence of this event was minimal. Although the condition was prohibited by the ASME code, there was no normal or accident condition that would cause the 75 psig relief valve to lift. Therefore, this minor procedure violation was not subject to formal enforcement action.

c. Conclusions

The actions taken to correct the improper attachment of a hose and metal tubing to the discharge of relief valve BS-V-63B were adequate. This minor violation was not subject to formal enforcement action.

E8.25 <u>(CLOSED) URI 50-289/96-201-25, and EEI 97-070-060[illegible], BWST Level Transmitter Enclosure and Heat Tracing</u>

a. <u>Inspection Scope</u>

The AE team identified a concern involving BWST level transmitter DH-LT-808. The team found that the cover plate of the enclosure for the instrument was missing, thereby raising the question of adequacy of the required instrument protective barrier. Subsequent NRC reviews determined that this was a violation of 10 CFR 50, Appendix B, Criterion V, as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. <u>Observations and Findings</u>

The licensee attributed the cause of BWST level transmitter DH-LT-808 not being found intact to the lack of adequate instructions in surveillance procedure 1302-5.19, "Borated Water Storage Tank Level Indicator." The procedure did not specifically require that the enclosure cover be replaced. Also, in the case of the level transmitter as-found heat tracing configuration not being consistent with applicable procedure and vendor instructions, the licensee determined that the cause of this was due to incorrect installation in 1988, and possibly due to a lack of experience by the installers.

The licensee's corrective actions taken to address these issues included: (1) reinstallation of the cover plate as required; (2) updating the surveillance procedure, 1302-5.19, to include additional instructions to ensure that the enclosure cover is closed after completion of the surveillance; and (3) correcting the level transmitter heat trace.

The inspector conducted a walkdown of the accessible areas of the BWST and noted that the cover plate of level transmitter DH-LT-808 was reinstalled properly. The inspector reviewed the surveillance procedure 1302-5.19, and verified that the licensee had added the appropriate instructions in the revised procedure to ensure that the enclosure cover would be closed after completion of the surveillance. The inspector also verified by reviewing Work Request No. 785993, completed on June 4, 1997, that the heat trace issue was appropriately corrected.

c. <u>Conclusions</u>

Appropriate actions were implemented to ensure that the BWST level transmitter enclosure and heat tracing were maintained properly.

E8.26 (CLOSED) URI 50-289/96-201-26, FSAR Discrepancies

a. Inspection Scope

The AE team identified some discrepancies in sections of the FSAR regarding the decay heat removal system (DHRS). While the discrepancies were minor in nature, the team was concerned with the fact that they 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. Observations and Findings

In subsequent follow up, the NRC determined that although a violation of 10 CFR 50.71(e), these discrepancies were of minor significance and the issue was therefore identified as a non-cited violation in a letter dated October 8, 1997 to the Licensee.

During this inspection, the inspector reviewed the licensee's actions taken to correct the identified discrepancies as well as those taken to prevent further FSAR discrepancies. The inspector verified that all the discrepancies were corrected in the April 1998 FSAR update. In a broader sense, the licensee established FSAR section owners to improve the quality of the FSAR and transferred the ownership of the FSAR to the engineering department.

c. Conclusions

The licensee adequately resolved the FSAR discrepancies identified by the AE team. In addition, the ongoing efforts by the licensee to improve the overall quality of the FSAR appeared appropriate.

E8.27 (CLOSED) IFI 50-289/96-201-27, Definition of Single Active Failure

The AE team raised a question involving the licensee's definition of "single active failure" as it involves check valves. Section 4.1.3.7 of TMI-1's decay heat removal system design basis document, SBDB-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). Therefore, the licensee's position was that check valves were not considered to be "active" components.

The NRC reviewed the licensee's position and found it to be consistent with NRC practice and with the licensing basis at TMI-1. Check valves, except for those in containment isolation systems, have been treated as passive components. Therefore, this item is closed.

E8.28 (CLOSED) URI 50-289/96-201-28, and EEI 97-070-10153, 10163, 10173, and 0183, Control of Calculations

a. Inspection Scope

The AE team identified situations where safety-related calculations where not controlled in accordance with the licensee's procedures. Following further NRC review, this issue was identified as a violation of 10 CFR 50, Appendix B, Criterion III, Design Control, as documented in a Notice of Violation and Imposition of Civil Penalties, dated October 8, 1997.

b. Observations and Findings

The licensee initiated QDR 97-2001 to document the concerns associated with the control of calculations. The licensee conducted a root cause evaluation and identified the causes of the problem to be lack of familiarity/inadequate training on the requirement of procedures and lack of management enforcement to fully implement procedural requirements. To correct this problem, the licensee initiated several actions including correcting the specific problems identified by the AE team and making programmatic enhancements to prevent recurrence.

The inspector verified that the issues specified in the AE inspection report (96-201) had been corrected as follows:

- The calculation buried in memorandum 5310-92-366(12/22/96), "Evaluation of TMI-1 HPI SSFI Observation No. 211-10" was incorporated into calculation No. C-1101-211-E630-080.

- The analysis for not leak testing check valves DH-V 14A and B as part of the IST program was nullified and the valves, including DH-V-5A and B, were incorporated into the IST Program and tested satisfactorily during the cycle 12 refueling outage.

- TDR No. 836, "Evaluation of Loading for the Emergency Diesel Generator and Engineered Safeguards (ES) Buses, was incorporated into design calculation C-1101-741-E510-005.

- TDR No. 995, "Voltage Drop Study for Degraded Grid Condition," was incorporated into design calculation C-1101-700-E510-010.

- EER 88-060-E, "BWST Level Alarm Setpoint Change," and EER 88-070-E, "Calibration for DH-DPS-914," were converted into calculation C-1101-212-E510-057.

As further corrective actions, the calculation process was enhanced with the implementation of procedure E-006, "Calculations." In addition, the licensee did the following: reviewed design basis calculations for key design basis parameters and implemented the calculations into the EP-006 process; reviewed a sample of TDRs, Topical Reports (TRs) for buried calculations that needed to be incorporated into the EP-006 process; and reviewed design calculations for FSAR Chapter 14 and other selected systems. The inspector reviewed EP-006, Calculations, Revision 7, and found it detailed. It is applicable to both TMI and Oyster Creek, and establishes the requirements, methods and responsibilities for the documentation reviews, and control of calculations generated by engineering. It contained information that would ensure that calculations are performed, reviewed, and approved properly. The inspector found the licensee's actions appropriate.

c. Conclusions

The inspector determined that the actions taken and those ongoing to identify and control analysis/calculations were adequate. The enhanced procedure, EP-006, "Calculations," was comprehensive and contained good guidance for controlling calculations.

V. Management Meetings

X1 Exit Meeting Summary

On August 27, 1998, the inspectors presented a summary of the inspection findings to the licensee at an exit meeting held at the site. Discussions with GPU Nuclear staff were held subsequent to the exit meeting to confirm certain details associated with the issues described in section E8.4. The list of persons contacted, below, reflects the participants at the meeting, including those who participated via telephone from GPU Nuclear's corporate office.

LIST OF PERSONS CONTACTED

Licensee

A. Rone,VP - Engineering
J. Langenbach, VP - TMI
R. McGoey, Director - Engineering Support (Acting)
P. Walsh, Director - Equipment Reliability
D. Skillman, Director - Configuration Control
S. Wilkerson, Manager - Systems Engineering
C. Hartman, Manager - Systems Engineering
A. Asarpota, Manager - Modifications
J. Schork, Manager - Nuclear Safety
M. Kapil, Manager - Electric Power & Instrumentation
N. Trikouros, Manager - Safety & Risk Analysis
E. Fuhrer, Nuclear Safety and Compliance Committee
B. Knight, Licensing Engineer
A. Miller, Licensing Engineer
B. McSorley, Systems Engineer
S. Maingi, PA - BRP

INSPECTION PROCEDURES USED

IP 92903 Follow up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened 50-289

98-06-01	VIO Makeup System Pumps Suction Cross Connect (E8.4)

Closed 50-289

96-201-01	IFI Letdown line Break in the Auxiliary Building (E8.1)
96-201-02	IFI Evaluation of Simultaneous Start of MU&P Pump MU-P-1C and Suction and Discharge Valves (E8.2)
96-201-03	IFI Evaluation of Gas Accumulation in Suction Piping for Makeup Pump MU-P-1C (E8.3)
96-201-04	URI Adequacy of Makeup Tank Pressure/Level Curves (E8.4)
96-201-05	URI Design Basis Valve Stroke Times in Surveillance Procedure (E8.5)
96-201-06	IFI Consequences of Failure of Auxiliary Steam Piping (E8.6)
96-201-07	IFI Loss of Pressure in MU&P Tank due to Letdown Line Break (E8.7)
96-201-08	IFI M&UP Pump NPSH When Taking Suction From BWST or Makeup Tank (E8.8)

96-201-09	IFI	Incomplete DC System Voltage Drop Calculations (E8.9)
96-201-10	URI	Alignment of MU-V-18 DC Power Supply (E8.10)
96-201-11	UR	Makeup Tank Level Instrument Loop Tolerances (E8.11)
96-201-12	URI	FSAR Discrepancies (E8.12)
96-201-13	URI	Adequacy of BWST Setpoint for DHRS Pump Switch over to RB Sump (E8.13)
96-201-14	URI	EEI 97-070-01093, Adequacy of Safety Evaluation for an FSAR Change Crediting Containment Over pressure for LPI Pumps NPSH (E8.14)
96-201-15	IFI	Technical Specification Discrepancy (E8.15)
96-201-16	URI	Discrepancy Between FSAR and Technical Specifications Regarding DHRS Leakage (E8.16)
96-201-17a	URI	Leak Testing of DHRS Pump Suction Check Valves (E8.17a)
96-201-17b	URI	Leak Testing of DHRS Pump Discharge Check Valves (E8.17b)
96-201-17c	URI	Inspection of DHRS Pump Vault Floor Drain Check Valves (E8.17c)
96-201-18	URI	Timeliness of Action on SSFI Open Items (E8.18)
96-201-19	IFI	Evaluation of Reactor Building Sump Screen for Reverse Flow (E8.19)
96-201-20	URI	Testing of Molded Case Circuit Breakers (E8.20)
96-201-21	URI	Static Head Correction for BWST Level Transmitter (E8.21)
96-201-22	IFI	BWST Level Instrument Drift (E8.22)
96-201-23	URI	Selection of BWST Low Level Alarm for Operator Action (E8.23)
96-201-24	IFI	Undocumented Modification (E8.24)
96-201-25	URI	BWST Level Transmitter Enclosure and Heat Tracing (E8.25)
96-201-26	URI	FSAR Discrepancies (E8.26)
96-201-27	IFI	Definition of Single Active Failure (E8.27)
96-201-28	URI	Control of Calculations (E8.28)

97-070-01023 EEI Adequacy of Makeup Tank Pressure/Level Curves (E8.4)

97-070-01013 EEI Adequacy of BWST Setpoint for DHRS Pump Switch over to RB Sump (E8.13)

97-070-01033 EEI Static Head Correction for BWST Level Transmitter (E8.21)

97-070-01093 EEI Adequacy of Safety Evaluation for an FSAR Change Crediting Containment Over pressure for LPI Pumps NPSH (E8.14)

97-070-03013 EEI Timeliness of Action on SSFI Open Items (E8.18)

97-070-06014 EEI BWST Level Transmitter Enclosure and Heat Tracing (E8.25)

97-070-07014 EEI Testing of Molded Case Circuit Breakers (E8.20)

97-070-08014 EEI Leak Testing of DHRS Pump Discharge Check Valves (E8.17b)

97-070-08014 EEI Inspection of DHRS Pump Vault Floor Drain Check Valves (E8.17c)

97-070-10143 EEI Leak Testing of DHRS Pump Suction Check Valves (E8.17a)

LIST OF ACRONYMS USED

AE	Architect Engineering
BS	Reactor Building Spray (ECCS)
BWST	Borated Water Storage Tank
CAP	Corrective Action Process
CFR	Code of Federal Regulations
DBD	Design Basis Documents
DC	Decay Heat Closed
DH	Decay Heat Removal (ECCS)
DHRS	Decay Heat Removal System
DR	Decay Heat River (ECCS)
EDG	Emergency Diesel Generator
EEI	Escalated Enforcement Item
ESF	Engineered Safety Feature
EQ	Environmental Qualification
FSAR	Final Safety Analysis Report
GL	Generic Letter
GPUN	GPU Nuclear
HELB	High Energy Line Break
HPI	High Pressure Injection
IFI	Inspection Follow up Item
LER	Licensee Event Report
LOCA	Loss of Coolant accident
LPI	Low Pressure Injection
MU	Makeup
MUT	Makeup Tank
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PRG	Plant Review Group
SSFI	Safety System Functional Inspection
SWSOPI	Service Water System Operational Performance Inspection
TMI	Three Mile Island
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USQ	Unreviewed Safety Question
VIO	Violation