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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Response to NRC Inspection Report No. 50-382/98-201

Gentlemen:

As requested in NRC Inspection Report No. 50-382/98-201 received on August 21, 1998, attached are responses to the Unresolved Items (URIs) and Inspector Follow-up Items (IFIs). The responses provide a schedule for completion of the corrective actions. In establishing this schedule Waterford considered the safety significance of each of the items in relation to other activities at the station and prioritized the actions accordingly. In addition, the responses provide a discussion, corrective steps that have been taken, and the results achieved, where appropriate.

Waterford recognizes the team identified examples where actions to identify and correct conditions adverse to quality may not have been adequate. The generic issue of timely corrective action has been previously identified by the Waterford self-assessment process and corrective actions are ongoing. Management has communicated clear expectations and goals for timely condition report identification and resolution. Performance measures have been established and indicate an improved corrective action program.

The team also identified deficiencies in 10 CFR 50.59 evaluations. As part of the Waterford performance improvement plan, enhancements were made to the 10 CFR 50.59 process. Part of these enhancements involved refresher training for engineers. Waterford management believes that the attention to detail in the 10 CFR 50.59 process is improving. In addition, the current efforts of the NRC and the industry to clarify 10 CFR 50.59 through rule making will contribute positively to better evaluations under the regulations in the future.

A number of the questions during the Architectural/Engineering (A/E) Inspection indicated documentation of the analyses to support the design and licensing bases could not be found that explicitly considered potential single active failures. Engineering judgement, regulatory interface, and industry experience were used during original plant design and licensing to determine the limiting cases for analysis. One example is original NRC Final Safety Analysis Report (FSAR) licensing question 211.71, which specifically addresses the Refueling Water Storage Pool (RWSP) failure modes. This is one example where all single failures were not discretely analyzed in the original design or identified in the Failure Mode and Effects Analysis (FMEAs). The robustness of the design, however, is confirmed in that the examples identified by the A/E Inspection Team were evaluated and shown to be bounded by the existing plant design. As part of Waterford's heightened awareness of the design and licensing bases, we committed in response to the NRC's 10 CFR 50.54(f) letter dated October 9, 1996, to perform a Design Basis Review (DBR) program. The DBR program is being enhanced to ensure the potential for single failures receives additional attention for the remaining systems. An assessment will be performed on the Safety Injection System (SIS), Containment Spray (CS) System and Emergency Feedwater (EFW) System to identify potential single failures. Waterford also committed in the10 CFR 50.54(f) response to perform Safety System Functional Inspections (SSFIs) which will continue to test the design and licensing bases for completeness and accuracy.

The NRC team pointed out a number of design and configuration control issues. We agree there are examples provided by the inspection team that constitute design and configuration control discrepancies. The examples were primarily the result of original design oversight and inattention to detail. Ongoing design basis review and calculation upgrade programs are designed to uncover these types of discrepancies.

Waterford recognizes the NRC team identified a number of deficiencies in surveillance testing. The specific examples identified in the report have been addressed individually in the attached. Generic implications are included in the responses where appropriate. On the whole, it was determined that a unique set of circumstances was involved in each example, and a common factor for all was not indicated.

In addition, in your cover letter, Waterford was requested to evaluate the inspection findings, both specific and programmatic, against its response to the NRC's October 9, 1996, request for information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design basis information. Waterford has reviewed the results of the A/E Inspection and, in addition to potential single failure reviews described above, has identified one additional area of improvement to its response to 10 CFR 50.54(f). Electrical calculation issues identified in the inspection reinforced the need for the review of the electrical sections of the Updated Final Safety Analysis Report (UFSAR) for accuracy and consistency with the electrical calculations. The review of certain sections of the UFSAR for accuracy had previously been committed to in the Waterford response to the 10 CFR 50.54(f) letter. Waterford has reviewed the inspection findings for programmatic issues and believes the issues do not indicate any programmatic failures that would not be corrected by our normal processes.

We are pleased the inspection team determined the EFW system and SIS were capable of performing their intended safety functions, and generally adhered to the design and licensing bases. We are also pleased your team recognized our staff took corrective actions to ensure system operability. Waterford is committed to go beyond the specific problems identified, and evaluate generic, related and similar items. We appreciate the assistance your team provided in helping to validate and calibrate efforts already in progress as part of our performance improvement plan.

If you have any questions, please contact me at (504) 739-6660 or Early Ewing at (504) 739-6242.

Very truly yours,

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CMD/OPP/ssf
Attachment 1: Response to Inspection Report 98-201
Attachment 2: Acronym Listing

cc: D.P. Norkin (NRC-NRR)
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ATTACHMENT 1

ENTERGY OPERATIONS, INC. RESPONSE TO THE ITEMS IDENTIFIED IN INSPECTION REPORT 50-382/98-201

URI 98-201-01a (Single Active Failure Analysis)

Description of Item:

FSAR Section 6.3.1.4 states that the safety functions of the SIS must be accomplished assuming the failure of a single active component. This requirement is consistent with General Design Criterion 35, as defined in Appendix A to 10 CFR Part 50. To demonstrate this capability, a failure modes and effects analysis for the SIS was performed and presented in FSAR Table 6.3-1. This table did not adequately address the potential effects of the failure of a throttle valve to the full-open position since the table identified the symptoms and local effects of such a failure as "none".

The failure to consider the potential impact of a credible single active failure on SIS operation is identified as URI 50-382/98-201-01a.

Item Discussion:

UFSAR Section 6.3.3.2.1 states that assuming no active component failures is the most limiting condition for a Large Break Loss-of-Coolant Accident (LBLOCA). However, the inspection team was concerned the single failure of one Low Pressure Safety Injection (LPSI) throttle valve (SI-138A/B or SI-139A/B) to a full open position could result in more safety injection flow than assumed in the accident analysis and exceed the flow rate measured during surveillance tests. The higher LPSI flow rate could potentially adversely impact required Net Positive Suction Head (NPSH) for the pump and motor capacity. Calculated Peak Clad Temperature (PCT) during a large break Loss-of-Coolant Accident (LOCA) could also be affected if the LPSI flow were greater than assumed. The LBLOCA analysis simultaneously assumes minimum LPSI flow for safety injection to the Reactor Coolant System (RCS) (to minimize inventory makeup) and maximum LPSI flow for spillage out of the break which maximizes the PCT.

A new calculation was performed to determine the maximum LPSI flow with the failure of a LPSI throttle valve. The results showed the maximum LPSI flow used in the Net Positive Suction Head (NPSH) calculations, LBLOCA analysis, and pump/motor sizing, conservatively bound the expected flow with the throttle valve failed full open. Therefore, the analysis values used were valid.

UFSAR Table 6.3-1 "Safety Injection Failure Modes and Effects Analysis" and UFSAR Section 6.3.3.2.1 will be enhanced to reflect the potential failure mode and effect of this failure.

Corrective Steps That Have Been Taken and the Results Achieved:

A study calculation (EC-M98-027) has been prepared and approved that analyzes the effects of the failure to the full open position of one LPSI throttle valve. The analysis shows flow rates would not exceed the values used in the LOCA analysis, therefore the LPSI system can withstand this single failure.

Additional Corrective Steps Planned To Be Taken:

Calculation EC-M98-027 will be converted to an official design basis calculation. UFSAR Table 6.3-1 "Safety Injection Failure Modes and Effects Analysis" and UFSAR Section 6.3.3.2.1 will be enhanced to reflect the results of the calculation. (ER-98-1178)

Schedule of Completion of Remaining Corrective Actions:

The calculation revision and enhancements to the UFSAR will be completed by 6/30/99.

URI 98-201-01b (Single Active Failure Analysis)

Description of Item:

During the injection mode of post-accident SIS operation, each train of SIS pumps as well as the CS pumps take suction from the RWSP. The RAS secures the LPSI pumps and opens the containment Safety Injection (SI) sump isolation valves (SI-602A,B). Maximum opening time for the containment SI sump isolation valves is 35 seconds. The impact of the single failure of a LPSI pump to trip upon receipt of the RAS was not considered. As stated in FSAR Section 6.3.2.5.4, this is a credible active failure. During the 35 seconds that the containment SI sump isolation valves are opening, continued operation of a LPSI pump would withdraw additional water from the RWSP potentially causing the water level to fall below the minimum required level of five percent to prevent vortexing. This is an additional example of failing to consider the potential impact of a credible single active failure on SIS operation and is URI 50-382/98-201-01b.

Item Discussion:

The single failure of a LPSI pump to trip on a Recirculation Actuation Signal (RAS) would cause water to continue draining from the RWSP following a RAS, until the containment SI sump isolation valve (SI-602 A/B) stroked open. This issue was evaluated in original FSAR licensing in NRC Question 211.71. However, no formal analysis was found which evaluated this scenario. A calculation is being prepared which will determine the process limits for RWSP inventory. This calculation considers a SI sump isolation valve stroke time of 35 seconds. Preliminary calculation results indicate RWSP water level would not drop to the point where vortexing would occur even if a LPSI pump fails to trip.

This analysis and calculation is conservative in that the stroke time of the valves is faster than 35 seconds (approximately 23 seconds) and that approximately full flow through the butterfly valve (SI-602 A/B) is achieved at 50% open. This results in the suction for the pumps switching from the RWSP to the SIS soon after the SIS valve starts to open. Since containment pressure following a LOCA will overcome the RWSP elevation head, flow will be preferentially from the SIS in containment and not from the RWSP.

Corrective Steps That Have Been Taken and the Results Achieved:

A process calculation (EC-M98-008) "RWSP Design Requirements" is in preparation. The calculation addresses the impact of the LPSI pump failure to trip and the opening time of the containment SI sump isolation valves. This calculation will confirm the current RAS setpoint is sufficient to prevent vortexing even if a LPSI pump fails to trip on a RAS. The preparation of an RWSP process calculation was previously planned as part of the design basis reconstitution effort.

Additional Corrective Steps Planned To Be Taken:

The new RWSP process calculation (EC-M98-008) will be finalized and approved. Affected design documents will be updated.

Schedule of Completion of Remaining Corrective Actions:

The calculation will be approved and design documents updated by 06/30/99.

URI 98-201-01c (Single Active Failure Analysis)

Description of Item:

The team suggested that one potential single failure could be the failure of a LPSI pump to trip upon receipt of the RAS. The licensee subsequently determined that failure of a containment SI sump isolation valve (SI-602A,B) to open upon receipt of the RAS would allow one safety injection train to continue withdrawing water from the RWSP until the RWSP was empty.

This is an additional example of the failure to consider the potential impact of a credible single active failure on SIS operation and is URI 50-382/98-201-01c.

Item Discussion:

The single failure of a containment sump isolation valve to open upon receipt of the RAS was evaluated in original FSAR Licensing Question 211.71. However, the containment flooding aspect was not adequately covered. In 1997, Condition Report (CR 97-1287) documented a potential to flood certain Regulatory Guide (RG) 1.97 instruments (which were not qualified for submergence) and to partially flood the cooling coils of a containment fan cooler following a LOCA. The maximum post LOCA flood level calculation was revised and procedures were changed to administratively control the maximum RWSP level at <90% to prevent flooding of the above equipment. However, this single failure had not been assumed in the post-LOCA containment flooding calculation. CR 98-0750 was written to address the issue of a failure of a SI sump isolation valve (SI-602A/B) to open upon receipt of a RAS (thereby allowing one SI train to drain the entire contents of the RWSP into containment). Immediate actions were taken to restrict the RWSP level to <87%.

The post LOCA containment flood level calculation shows a portion of the coils of two containment fan coolers and several instruments (SG ILT 1115 A&B; SG ILT 1125 A&B; RC IPT 0106 A&B; and SI ILT 7145 A&B) would become submerged if the initial RWSP level was at 100%.

A previous analysis showed that even if 20% of the fan cooler coils are submerged, the impact on containment peak pressure is minimal (less than 0.15 psig increase). Since only one containment fan cooler would be partially submerged at <87% RWSP level, the acceptance criteria for containment peak pressure would still be met. Therefore, the partial submergence of a containment fan cooler does not affect the integrity of the containment barrier.

The Steam Generator (SG) wide range level measurement instruments (SG ILT 1115 A&B, SG ILT 1125 A&B) are required by RG 1.97 and no backup instruments exist for wide range. SG narrow range level instrumentation is still available. The SG wide range level is an input to the EFW system flow control logic. EFW is not required for a large break LOCA which floods containment in a relatively short time. It is required for a small break LOCA, but the time to empty the RWSP and reach the maximum flood level is much longer. Thus, automatic EFW control would have restored steam generator level to within the narrow range instrumentation band before the wide range level instruments became submerged so the operator could manually control EFW flow if necessary. The current Emergency Operating Procedures (EOPs) identify use of the narrow range SG level instruments, which are not flooded, for manual control of EFW. Furthermore, the wide range instruments would not be submerged until well after the time of peak cladding temperature for all LOCAs. Thus, potential submergence of the steam generator wide range level instruments would have no impact on the accident analysis results.

Hot leg pressure measurements (RC IPT 0106 A&B) are required by RG 1.97 (for subcooled margin) and no backup instruments exist for hot leg pressure. However, pressurizer pressure could be used as an alternate means of determining subcooled margin. Thus, loss of the hot leg pressure indication would have no impact since the information could be obtained by other means.

Safety Injection sump level measurement (SI ILT 7145 A&B) is required by RG 1.97 and no backup exists. The instruments will be available as the sump fills during a LOCA to provide positive indication of water in the sump. Loss of this instrument does not impact the accident progression or results.

Corrective Steps That Have Been Taken and the Results Achieved:

The containment flooding calculation has been revised to reflect the single active failure of a SI sump isolation valve (SI-602A(B)) to open upon receipt of an RAS. As a result, the administrative limit on maximum RWSP level has been reduced from <90% to <87% to ensure that the equipment mentioned above will not be flooded following a LOCA.

Operations procedure OP-903-001 has been revised to reflect the RWSP maximum level of <87%.

Additional Corrective Steps Planned To Be Taken:

Corrective action is complete.

Schedule of Completion of Remaining Corrective Actions:

Corrective action is complete.

URI 98-201-02 (ECCS Leakage Test Acceptance Criteria)

Description of Item:

Waterford implemented the TS requirement in OP-903-110, "Surveillance Procedure - RAB Fluid Systems Leak Test," Revision 3. This procedure stated that if a leak was identified, attempts could be made to stop or reduce the leakage before the leakage rate was measured. The team identified that the as-found leakage was not being measured, which was not consistent with the FSAR commitment. Waterford reviewed OP-903-110 and determined that Revision 1 of the procedure had required the measurement of the leak rate before any attempts to stop or reduce the leakage. However, in Revision 2 (February 1997), these steps were reversed for reasons that were not known.

The failure to measure the as-found leakage consistent with the commitment stated in FSAR Section 1.9.37, and the lack of control of procedure revisions, is identified as URI 50-382/98-201-02.

Item Discussion:

The Reactor Auxiliary Building (RAB) Fluid Systems Leak Test surveillance procedure is intended to ensure there are no radioactive fluid leaks inside the RAB that could significantly affect the accident dose rates. Conservative leakage rate assumptions (1 gpm or 3784 SCCM) are considered in the dose rate calculations. These leakage rates include packing leaks, pump seal leaks and other minor sources. The most recent system leakage rate was 306 SCCM, which is significantly below the limit in the dose rate calculations.

A condition report (CR 98-0848) analyzed this issue and determined the deletion of the requirement to measure the leakage rate prior to making adjustments to stop the leak, was the result of human error. During the revision to the procedure, steps were reordered and the requirement to identify the system leakage was omitted. The condition report determined there were no operability concerns based on the results of the most recent surveillance which quantified the "as-left" leakage rate.

Corrective Steps That Have Been Taken and the Results Achieved:

Condition report (CR 98-0848) researched the history of the procedure and reviewed recent test results to ensure there were no past operability concerns. Other surveillance leak testing procedures were reviewed to determine if there were similar issues. No similar issues were found.

Additional Corrective Steps Planned To Be Taken:

OP-903-110, "RAB Fluid Systems Leak Test" surveillance procedure will be revised to require documenting the leak rate before making any adjustments. The impact on the dose rate margins of the as found conditions will be determined.

Schedule of Completion of Remaining Corrective Actions:

Revision of the procedure will be completed by 11/30/98 (CR 98-0848 CA 2), prior to the next scheduled test.

URI 98-201-03 (Dose Consequences of RWSP Back-Leakage)

Description of Item:

As described in UFSAR Section 9.4.3, the RAB normal ventilation system is not safety related or seismically designed since its operation is not required to mitigate the consequences of an accident. The team considered it inappropriate to credit operation of this system in EC-S92-001. If the operation of the RAB normal ventilation system was not considered, the 5 gpm of ECCS back-leakage could result in control room doses that exceed the acceptance limits specified in 10 CFR Part 50, Appendix A, General Design Criterion 19, and Section 6.4 of NRC's Standard Review Plan (SRP).

The use of inappropriate design-basis assumptions for the determination of acceptable ECCS valve leakage limits is identified as URI 50-382/98-201-03.

Item Discussion:

NRC Information Notice (IN) 91-56 discusses the potential impact on post-LOCA radiological consequences as a result of valve seat leakage of liquid from the containment into other systems such as the RWSP. The Waterford LOCA radiological consequences analysis was revised to include dose contributions from a 5 gpm back-leakage to the RWSP in order to add conservatism in the dose calculation. In the Control Room dose calculation, credit was taken starting at 24 hours into the event for the non-safety RAB normal ventilation system in accordance with guidance provided in ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactor Safety Related Fluid Systems.

RG 1.4 (Rev. 2, June 1974) provides guidance for LOCA offsite dose calculations but does not discuss pathways required to be analyzed or any liquid leakage assumptions. SRP NUREG-0800, Section 15.6.5, Appendix A, provides guidance on the contributors, which should be included in design basis dose calculations. The only contributor to offsite dose identified in the SRP from liquids is by direct leakage from the Engineered Safeguard Features (ESF) system into the RAB atmosphere or to the environment. Appendix B to Section 15.6.5, Radiological Consequences of a Design Basis LOCA: Leakage from ESF Components Outside Containment, identifies this as valve stem leakage, pump seal leakage, etc. Waterford's licensing basis assumed this leakage to be released within the Controlled Ventilation Area and therefore it is processed by a safety related emergency filtration system. The licensing basis is silent on the treatment of liquid leakage past ESF boundary valves (valve seat leakage), transport through water systems, and subsequent release to the environment. If operation of the RAB normal ventilation system was not considered, the 5 gpm of Emergency Core Cooling System (ECCS) back-leakage to the RWSP could result in Control Room doses that exceed the appropriate acceptance limits.

NRC correspondence is available which evaluated similar concerns to those described in IN 91-56. NRC memorandum dated May 15, 1995 from David B. Matthews to Ellis W. Merschoff discusses the results of an NRC task group created to address IN 91-56 issues for several plants. The pertinent conclusions reached by the NRC task group documented in the memo are 1) "In general, source terms in the licensing basis for plants do not assume water leakage as a contributor to offsite doses", and 2) "Requiring licensees to assume radioactive water leakage to the atmosphere would be a compliance backfit."

Corrective Steps That Have Been Taken and the Results Achieved:

Based on the above considerations (no requirements to consider valve seat leakage of water to other systems, treatment of only direct ESF system leakage to the RAB, and NRC correspondence), the Waterford licensing basis does not require valve seat leakage as described in IN 91-56 to be included in the offsite or Control Room dose calculations. As such, the LOCA radiological consequences analysis has been revised. The 5 gpm back leakage to the RWSP and credit for the RAB normal ventilation system have been removed. The UFSAR will be updated to reflect these changes.

Additional Corrective Steps Planned To Be Taken:

Waterford will update the UFSAR as required.

Schedule of Completion of Remaining Corrective Actions:

The UFSAR will be updated by 11/30/98.

URI 98-201-04 (Containment Sump Isolation Valve Leakage Testing)

Description of Item:

The team was concerned that the as-found leakage of the valves (SI Sump Isolation Valves SI-602A/B) was not being measured, and that the Repetitive Work Tasks (RWT) did not identify the valve test conditions or relate acceptable valve leakage to the limit established in calculation EC-S91-016.

The failure to measure the as-found leakage from valves SI-602A/B, and the lack of a documented test procedure that specifies test parameters and acceptance criteria, is identified as URI 50-382/98-201-04.

Item Discussion:

Valves SI-602A(B) are listed as containment isolation valves and are exempt from local leak rate testing based on the existence of a post accident water seal (UFSAR sect . 6.2.6.3 and table 6.2-43). The associated penetrations for these valves are not required to be vented or drained for the Integrated

Leak Rate Test (ILRT) (UFSAR Table 6.2-32). The Waterford Inservice Testing Plan does not require periodic leak testing to meet ASME code requirements. The ASME IST category B classification of this valve is appropriate based upon the Entergy Operations Category A designation criteria and the calculated allowable leak rate limits.

ASME-OM Code part 10 provides that, if a leak rate is not specified, valve leakage rates are to be established by the Owner. Further guidance is provided to establishing leakage limits based upon valve diameter. EOI's Category A designation criteria is based on this Code provision. Comparatively, the limit for this valve, based the Category A designation criteria, was found to be in excess of gross leakage. For this 24 inch valve, the maximum allowable limit for prevention of ESF pump air binding is approximately 15 times greater than the Category A designation criteria. Testing in accordance with ASME Section XI testing, Generic Letter 89-10 MOV program, and requisite post-maintenance test limits provide adequate assurance of reliable valve operation in terms of the safety functions assigned.

As such, the appropriate periodic testing of these valves is component level testing based on valve maintenance frequency and post-maintenance testing following valve rework. This testing is sufficient to ensure the valves will effectively isolate the ECCS system from excessive air-inleakage during the small post-accident time span that occurs before the sump is covered by a water seal. After replacement of the valves sealing surface "T" rings, the valve is leak tested and adjustments are made as necessary.

Corrective Steps That Have Been Taken and the Results Achieved:

Waterford is in compliance with it's licensed containment leakage testing and ASME IST testing programs.

Additional Corrective Steps Planned To Be Taken:

No further corrective actions.

Schedule of Completion of Remaining Corrective Actions:

Corrective actions complete.

URI 98-201-05 (LPSI Pump Minimum Flow)

Description of Item:

Letter W3P89-2100, dated October 31, 1989, provided supplemental information regarding NRC Bulletin No. 88-04 and stated that the LPSI pumps minimum flow was still 100 gpm and did not have the imposed time restriction. Vendor Manual TD I075.0045, "Ingersoll-Rand Low Pressure Safety Injection Pumps Minimum Flow Evaluation," dated January 27, 1989 indicated that the original 100 gpm recommendation was a short-period recommendation for 3 hours or less that is required to prevent abnormal pump wear. A new minimum flow restriction of 2000 gpm was established within the manufacturer's Bulletin No. 88-04 review for continuous pump operation of 3 hours or more.

Operating procedure OP-902-002, "Loss of Coolant Accident Recovery Procedure," Revision 7, dated December 15, 1995, provides for the termination of SI (if not required); however, no specific precautions with regard to extended operation were identified. Attachment 6, "Minimum LPSI Flow Versus Pressurizer Pressure," to OP-902-ATT, "Emergency Operating Procedure Attachments," Revision 4, dated December 15, 1995, appears to indicate the expected LPSI flow versus RCS pressure, but did not identify any precautions for extended low-flow LPSI pump operation.

The proposed revision (8) was not adequate and did not address the situation where RCS pressure could be lower than the shutoff head, but high enough that the manufacturer's minimum flow recommendation could not be achieved.

The inconsistency between the manufacturer's recommendations and the response provided for Bulletin No. 88-04 and the absence of procedural guidance with regard to LPSI pump minimum flow requirements are identified as URI 50-382/98-201-05.

Item Discussion:

The vendor states that 100 gpm minimum flow is adequate for up to 3 hours. Beyond three hours, the LPSI pumps should have 2000 gpm of flow. This recommendation is based on the minimum flow rates required to prevent abnormal pump wear (mechanical seal and bearing wear). Thrust bearing loads are higher at low flows than at the design flow rates and seal life is assumed to be reduced at low flow due to increased vibration. NRC IN 88-04 issued in 1988, addressed these issues. NRC IN 93-08 also addressed the same issue with LPSI pumps only. This notice addressed a failure of a LPSI pump due to thrust bearing degradation resulting from operation at lower flows.

Waterford prepared a detailed investigation of the IN 93-08 LPSI pump low flow issue and how it affected the existing LPSI pumps. The investigation reviewed the operation of the pumps in the shutdown-cooling mode at flows below the 2000 gpm recommendation of the vendor. It concluded operation of the pumps at this low flow rate may contribute to increased wear on the motor thrust bearing. The analysis also pointed out that of the 100 similar pumps in operation, there has been just one failure (high vibration levels but the pump was still operable). The increased wear on the bearing would not result in a failure of the pump to deliver the design flow rates, but would result in higher vibration levels.

Increase in wear on the motor thrust bearing and pump seal, similar to that discussed above, would result from operation at the minimum flow rate of 100 gpm. The wear would be expected to be higher at this flow rate because the loads on the thrust bearing and pump vibrations are higher. However this wear is not expected to cause failure of the pump in a short period of time. Written correspondence from Ingersol-Dresser Pump Company states that operation at the 100 gpm minimum flow rate for periods of up to 24 hours should not result in a pump failure, only reduced bearing and seal life.

Operations of a LPSI pump on minimum recirculation for greater than three hours is not expected to occur except during accident scenarios. In accident scenarios, the time the LPSI pumps may operate is from the Safety Injection Actuation Signal (SIAS) to the time when the RAS occurs. For small break LOCAs, this can be on the order of 12 hours. Beyond 12 hours shutdown cooling would be initiated resulting in a greater LPSI pump flow. This time is not sufficient to cause bearing damage that would prevent the LPSI pumps from performing their safety functions of low-pressure injection and decay heat removal.

The recommended action of the IN 93-08 investigation was to perform vibration testing as part of the quarterly pump In Service Testing surveillance and sample lubricants for indication of bearing wear. These measures would ensure any abnormal bearing wear would be identified with adequate time to perform the necessary corrective action to ensure the LPSI pumps remain capable of performing their expected post accident requirements. These measures were already in place before the 1993 investigation. Also, during

LPSI pump surveillances, a leakage inspection is performed which would identify any LPSI pump seal leakage resulting from seal degradation or wear.

Based on the above, Waterford concludes that during accident scenarios and during some shutdown cooling operation (e.g., mid-loop operation), the LPSI pumps may be operated at flows lower than 2000 gpm. Investigation has shown this may result in bearing wear but will not result in failure of the pump to perform its safety functions. Proper monitoring of pump bearing wear will ensure any adverse bearing wear is corrected before pump operation is affected.

Corrective Steps That Have Been Taken and the Results Achieved:

A condition report (CR 98-0850) was initiated to investigate this issue. The CR gathered additional information on the issue and determined there were no operability concerns. Recent written correspondence from the vendor and a previous information notice (IN 93-08) investigation assure the pumps will remain operable. The IN 93-08 investigation was reviewed and determined to still be valid.

Additional Corrective Steps Planned To Be Taken:

Corrective action is complete.

Schedule of Completion of Remaining Corrective Actions:

Corrective action is complete.

URI 98-201-06 (CSP Level Measurement)

Description of Item:

The suction flow path associated with the EFW pumps level transmitter EFWILT-CD-9013A or B is measuring the static pressure from the EFW suction pipe of the condensate storage pool (CSP). Since the level transmitter is installed in the suction side of the pump, when the EFW pumps operate and water flows through the pipe, the pressure read by the level transmitter is reduced by friction (including entrance) losses and by the velocity head. These two factors bias the measurement and erroneously indicate a lower pool level than actual. The team determined that the licensee did not consider the dynamic induced flow effects when designing the level instrumentation for the CSP.

The failure to account for friction and velocity head losses for CSP level indication is identified as URI 50-382/98-201-06.

Item Discussion:

The Condensate Storage Pool (CSP) level sensing lines are installed on the EFW suction headers. Due to this design configuration, actual CSP indication is inaccurate due to flow related losses created by EFW flow during EFW demand conditions. Preliminary evaluations were performed to quantify the impact of EFW flow on the CSP level indication. Several EFW flow rates were analyzed. The worst case EFW flow rate (all three pumps operating at run out capacity) would result in a 28% error in the level indication. The same EFW flow rate was considered assuming only one EFW suction header was available. The inaccuracy went up to over 80%, however, in this scenario the instrument on the other leg would provide accurate indication since no flow was assumed in this suction header.

If an intermediate EFW flow rate of 1800 gpm were assumed, this flow rate would result in an error of 15%. A typical long-term flow rate that was considered is 150 gpm going to each steam generator or 300 gpm going to the non faulted steam generator. This flow results in an error of 1%.

There are two levels of interest to the operators in the CSP. The first is a Technical Specifications (TS) 3/4.7.3.1 low level limit of 91%. This limit was established to ensure there is sufficient water in the CSP at the initiation of an event. Once an EFW demand event has started, this limit becomes non-significant. There is the potential to indicate below the TS low level during normal EFW flow surveillance testing. This surveillance recirculates water back to the CSP. If the operator started makeup to the CSP based on the inaccurate level indication, the CSP could overflow. The actual CSP level can be determined after the surveillance is completed and appropriate action taken. Waterford has determined the risk of overflowing the CSP is not significant and no corrective action is needed as a result of the potential early indication of the TS low level being reached during surveillance testing.

The second level of interest is the 25% level when the Auxiliary Component Cooling (ACC) system is aligned to the CSP for makeup. This line up also supplies water to the suction of the EFW system. At the time of actual 25% CSP level, EFW will be maintaining the steam generators at a predetermined level. Depending on plant conditions, a flow of approximately 300 gpm (150 gpm to each steam generator or 300 gpm to the non-faulted steam generator) would result in a CSP level inaccuracy of 1%. Although the 25% level is when operations takes manual actions, the 1% inaccuracy is not considered significant.

Because of the piping configuration, when makeup to the CSP is provided by the ACC system, ACC water will flow back through the suction header and refill the CSP. This back flow will also affect the level instruments because of the line pressure. The level indication would be approximately 25% higher than actual. This error makes it difficult for operators to determine how much water has been added to the CSP. Although the level indication would be higher, confusion could result in transferring more water to the CSP from the Wet Cooling Tower (WCT) basin than necessary. This could potentially leave the Ultimate Heat Sink (UHS) with inadequate water inventory to meet its design requirements. As a contingency, guidance was provided to the operators to align the ACC system for a maximum of thirty minutes if makeup is required to the CSP. Analysis shows the CSP would be sufficiently filled in that time frame to allow several hours of EFW flow. Thirty minutes is sufficient time to assure makeup for EFW and maintain UHS inventory requirements.

<u>Corrective Steps That Have Been Taken and the Results Achieved:</u>

Analyses were performed to determine the CSP level indication inaccuracies due to the losses created by the EFW flow during various EFW demand events. This information was used to assess the impact on emergency operating procedures. A note was added to Appendix 10 of the EOPs to address the large uncertainties in CSP indicated level resulting from EFW or Auxiliary Component Cooling Water (ACCW) operating at high flow conditions.

The impact of early indication of a TS 3/4.7.3.1 low level limit of 91% was also analyzed. This early indication of low level could prompt the operator to start refilling the CSP and overfill it. During surveillances, this water is recirculated to the CSP and there is no actual level change. The risk of overfilling the CSP is not significant so no corrective action has been taken.

Analyses were performed to determine the appropriate amount of time that the ACC system can be aligned to the CSP without potentially transferring more water out of the WCT basin than the inventory requirements allow. The thirty-minute recommendation was made to operations based on the analyses. A step to close the appropriate valves when 30 minutes has elapsed, was added to Appendix 10 of the EOPs to give this guidance. An analysis to determine the time to refill the CSP during the described scenario, which would then provide a basis of operating procedure revision, was previously planned as part of the design basis reconstitution effort as open item EFW-130.

To determine if there were other instruments that would be affected by flow, an evaluation of the other tanks and pools in the plant was performed. There were no other instances identified where level instruments would be negatively affected by expected flow.

Additional Corrective Steps Planned To Be Taken:

An Engineering Request (ER-98-0876) was written to relocate the level instrument sensing lines to stagnant headers. Relocation of the sensing points to piping not affected by EFW flow will correct these issues. Once the sensing points are relocated, the notes and cautions in the operations procedures will be removed.

Schedule of Completion of Remaining Corrective Actions:

The relocation of the level instrument sensing points is expected to be completed during Cycle 10 but not later than Refuel 10.

IFI 98-201-07 (Steam Generator Level Transmitter Failure)

Description of Item:

Failure of one steam generator's level transmitter (SG-ILT-1115A or 1125B) could cause the EFW secondary control valves to open fully and admit more water than required. The excess water may go unnoticed in the control room until the water level in the affected steam generator would reach 85 percent. This failure mode and effect had never been analyzed. Qualitatively, it was stated that the high-level alarm was chosen because it is lower than the reactor trip setpoint (87 percent), but the quantitative design basis for the alarm at the 85 percent level could not be identified.

This issue is identified as IFI 50-382/98-201-07.

Item Discussion:

Failure of one steam generator's wide range (WR) level transmitter could cause one EFW regulating valve to open fully and, on an Emergency Feedwater Actuation Signal (EFAS), allow more flow than required. This excess water may go unnoticed in the control room until the water level in the steam generator reaches the high level alarm at 85.4% WR. The operators would be required to take manual control of EFW to limit or secure the emergency feedwater flow prior to overfilling the steam generator. This scenario and potential single failure were not documented in the basis for the 85.4% WR high level alarm.

An ER was performed to demonstrate the operators have sufficient time upon receipt of the high level alarm to secure EFW prior to overfilling the affected steam generator. It was conservatively determined that the operators have more than 12 minutes from EFW actuation to prevent overfill and over 8 minutes from receipt of the high level alarm to prevent overfill. This time is based on conservative assumptions such as taking no credit for steaming and calculating the EFW flowrate based on flow through both EFW lines. Also, the high EFW flow alarm that actuates at a flow rate greater than 1100 gpm provides another indication of excess EFW flow prior to the high level alarm.

Corrective Steps That Have Been Taken and the Results Achieved:

ER-98-0764 demonstrates the operators would have sufficient time upon receipt of the high level alarm to secure EFW prior to overfilling the affected steam generator.

Calculation EC-I92-019 has been revised to document the single failure in the basis of the steam generator high level alarm.

Additional Corrective Steps Planned To Be Taken:

Corrective Action is complete.

Schedule of Completion of Remaining Corrective Actions:

Corrective Action is complete.

URI 98-201-08 (EFW Discharge Check Valve Testing)

Description of Item:

The IST tests of the EFW discharge check valves EFW-207A,B and AB, did not verify that the valves actually fully (reverse flow) close. If the check valves do not close, some of the pump's flow may be recirculated through the EFW pumps (which are not operating) to the CSP rather than reaching the steam generators. This may prevent the EFW from fulfilling its safety function of maintaining the steam generators water level.

The failure to ascertain that the pumps discharge check valves fully close in accordance with ASME Section XI, Subsection IWV-3522 and TS Section 4.05 is identified as URI 50-382/98-201-08.

Item Discussion:

In responding to a question from the NRC inspection team about leakage limits for the EFW pump discharge check valves, Waterford personnel identified that the test method for IST testing of the valves did not verify check valve closure adequately. CR 98-0822 was initiated to document and correct this problem.

The EFW pump discharge check valves are listed in the IST Plan as requiring full-stroke exercise testing in the open and closed directions on a cold shutdown frequency. Procedure OP-903-014, Emergency Feedwater Flow Verification, is used to verify EFW flowpaths and to satisfy the cold shutdown frequency testing for certain EFW System valves. The procedure explicitly addresses full-stroke open testing of the discharge check valves but does not explicitly include steps and documentation for closure testing.

The quarterly EFW pump testing procedure, OP-903-046, was intended to provide closure verification of the EFW pump discharge check valves on a quarterly basis. This procedure tests each pump individually and directs that appropriate data be collected while the pump is operating in the recirculation mode. This mode of operation is necessary since the only practical flowpath available to use for the EFW pumps during normal plant operations is a minimum recirculation flowpath. Satisfactory performance of a particular pump was considered to be an indication that the discharge check valves for the two pumps not being tested are closed since the discharge lines from all three pumps are cross-connected. It was expected that if either or both of the discharge check valves for the idle pumps were not adequately closed, the performance of the tested pump would appear degraded. In this case, the pump differential pressure recorded would be less than the reference value of differential pressure corresponding to the nominal recirculation flowrate.

Apparently, during the development of the quarterly pump test procedure, the EFW pump performance curves were not examined to determine how much flow diversion would be required to cause differential pressure of the tested pump to be lower than acceptance criteria. The Total Dynamic Head (TDH) of each EFW pump is essentially constant in the low flow range (first few hundred GPM). Since each pump is operated in its recirculation mode when tested, it operates in this low flow range. If a flow diversion pathway exists when the pump is tested, such as through one or both of the discharge check valves of the other pumps, a noticeable change in differential pressure may not be observed since the pump curve is "flat" in the range of flow where the test is performed. Therefore, the test method implemented in the quarterly pump test procedure is not an acceptable means of verifying check valve closure.

Corrective Steps That Have Been Taken and the Results Achieved:

Data obtained from the performance of OP-903-014 during REFUEL 8 was analyzed. Test data was conservatively adjusted to account for instrument inaccuracies and the performance of each pump was evaluated. It was determined that all EFW pumps performed better than the performance expected when maximum allowed flow diversion exists. This evaluation concluded that the REFUEL 8 test confirmed all three discharge check valves were closed.

A review of the closure verification test methods for pump discharge check valves having closed safety functions was conducted for other safety-related pumps in the IST Program. It was determined the test methods used were not the same as for the EFW pump discharge check valves (closure inferred by other pump's performance).

Based upon the test data from Refuel 8 which confirmed operability of the discharge check valves this issue is not safety significant.

Additional Corrective Steps Planned To Be Taken:

Test procedures will be revised, as appropriate, to explicitly include steps and documentation to verify closure of EFW pump discharge check valves as required by the Waterford Inservice Test (IST) Plan.

Schedule of Completion of Remaining Corrective Actions:

Procedure revisions will be completed by January 15, 1999.

IFI 98-201-09 (EFW Tornado Missile Protection)

Description of Item:

The turbine stack and steam piping could be damaged by a tornado generated missile and adversely affect the operability of the EFW turbine-driven pump. The team noted that assuming the turbine-driven pump is damaged because of lack of adequate tornado-missile protection, and a single failure of a diesel generator, there would be only one motor-driven EFW pump available for plant cooldown. The design basis requires flows from either two motor-driven pumps or a turbine-driven pump to safely cooldown the plant.

Waterford contends that the vent stack and steam lines are not to be protected on the basis of low probability. Thus, this issue has been referred to the NRR staff for further review. (IFI 50-382/98-201-09).

Item Discussion:

Waterford evaluated potential vulnerabilities associated with tornado missile protection at the plant. Evaluation results support Waterford's position that physical tornado missile protection is not required for the Emergency Feedwater Pump Turbine stack and steam piping.

A review of the evolution of the Waterford licensing design basis shows tornado missile protection received detailed evaluations and walkdowns by the owner, the regulator, the designer, and the manufacturer. Prior to obtaining the operating license, Waterford realized an inadvertent error was made in the original FSAR Table 3.5-3 regarding EFW pipe thickness requirements to withstand the tornado generated missile impact. FSAR Table 3.5-3 was revised via Amendment 33 in 1983 to reflect that missile protection of the exposed EFW system piping and exposed Main Steam (MS) system piping were attained by very low tornado missile strike probability. This calculation was based on a conservative manual calculation methodology. NRC review and approval of Amendment No. 33 was obtained prior to issuing the operating license.

On October 21, 1996, Waterford provided (letter W3F1-96-0188) an evaluation of the vulnerability of some conduits for the UHS routed above the RAB walls to tornado missiles. Subsequently Waterford (letter dated June 4; 1997; W3F1-97-0132) updated the NRC regarding the resolution of the conduit issue and to apprise the NRC about work performed on the design and licensing basis for tornado missile protection. As discussed in the June 4, 1997 letter, the design and licensing basis for tornado missile protection for particular systems and components was predicated on two important concepts, (1) the concept of low probability of damage by virtue of the diversity, redundancy, robustness of design features, and (2) the concept of a low probability of offsite hazards incorporated in Standard Review Plan (SRP) 2.2.3 and 3.5.1.4.

Waterford was evaluated for offsite hazards in addition to tornado missiles from natural phenomena, on the basis of a commonly applied qualitative probability standard of 10^{-6} documented in SRP 2.2.3. The qualitative probability standard in SRP 2.2.3 specifically states, "...the expected rate of occurrence of potential exposures in excess of 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower."

The probabilistic approach to evaluate vulnerable targets has been used at other nuclear power plants and has received NRC approvals. Calvert Cliffs Nuclear Power Plant, Units 1 and 2 obtained NRC approval (May, 1995) for eliminating the licensing requirement for tornado missile barriers based on associated low risk as shown by probabilistic risk assessment techniques consistent with the guidance provided in NUREG-0800 (SRP). Perry Nuclear Power Plant, Unit 1 was approved for the use of the TORMIS methodology when evaluating the need for positive missile protection of unprotected components or portions of systems. This methodology is consistent with the guidance provided in SRP Section 2.2.3, SRP Section 3.5.1.4, and RG 1.117. These approvals used an acceptable probability criterion of 10^{-6}

The latest probability risk assessment confirmatory analysis summarized in the June 4, 1997, letter to NRC was based on the "TORMIS" methodology as proposed in Electric Power Research Institute (EPRI) Report NP-2005. This "TORMIS" methodology was earlier reviewed by the NRC in 1983 and accepted as a state-of-the-art probabilistic Monte Carlo technique that can be used in PRA evaluation for tornado generated missiles. Table 3, attached to the June 4, 1997, letter to the NRC, documents the probability calculated for tornado vulnerability for the Terry Turbine Stack as 3.96×10^{-12} and 3.00×10^{-11} for the steam line. These probabilities fall well within regulatory guidelines.

The exposed steam piping vulnerability was identified and documented as an amendment to the FSAR prior to obtaining the operating license. The EFW turbine stack had been self identified and evaluated prior to the A/E Inspection in resolution of CR 96-1563. Development of a tornado missile design criteria document was also committed to in Waterford's 50.54f response. Based on evaluation results, this item is not safety significant.

Corrective Steps That Have Been Taken and the Results Achieved:

Corrective Action is complete subject to further review by the NRC.

Additional Corrective Steps Planned To Be Taken:

Waterford will update the UFSAR and other design documents after final resolution of June 4, 1997, letter.

Schedule of Completion of Remaining Corrective Actions:

Schedule to be determined based on resolution of June 4, 1997 letter.

URI 98-201-10 (10 CFR Part 21 Reviews)

Description of Item:

The manufacturer of Agastat relays issued a 10 CFR Part 21 notification, dated September 29, 1994, concerning the inability of the E7000 series relays to switch a 1-ampere load at rated voltage. Waterford had not performed an evaluation to determine if the condition was adverse to quality at Waterford. The initial review determined that 13 other 10 CFR Part 21 notifications cited between July, 1994, and June, 1996, were not evaluated.

The team identified this item as URI 50-382/98-201-10.

Item Discussion:

The initially identified 13 10CFR 21 notifications were reviewed in detail during and following the A/E Inspection. In the final analysis, the actual number of overlooked 10 CFR 21 notifications was eleven. The remaining items were either not 10 CFR 21 notifications or were duplicate notifications. The 10 CFR Part 21 evaluation process was changed in 1996 and assigned to a different organizational group, resulting in a failure to follow up and document closure of these eleven issues.

Corrective Steps That Have Been Taken and the Results Achieved:

CR 98-0819 was generated to document this adverse condition, to complete an "Operability Determination" for all components/systems impacted by each of the items and to track completion of corrective actions. Eleven of the items were valid 10 CFR Part 21 notifications and were assigned to be screened and evaluated. All items have been closed except the one applicable to the E7000 relay contacts.

ER-98-0750 was initiated to address the Agastat relay concern where Agastat derated the E7000 timing contacts from 1 Amps Direct Current (ADC) to 0.5ADC. Waterford conducted a review of (318) the plant's class 1E application of E7000/7000 series timers. There are no applications where the relay is required to break current in excess of 0.5ADC. The relay contacts have been tested to "make" up to 4.6ADC successfully for 10,000 operations without having a failure. Sixteen applications were identified where the relay contacts "make" Direct Current (DC) circuits with current values in excess of 4.6ADC. There has been no failure at Waterford of an E7000/7000 series relay attributed to the failure of a DC timing contact. Also, the applications where the contacts makes >4.6ADC are not considered to be of immediate concern due to this infrequent operation.

Additional Corrective Steps Planned To Be Taken:

Perform an additional review/analysis of the remaining 16 relays.

Schedule of Completion of Remaining Corrective Actions:

Waterford will complete the review/analysis by 10/30/99.

URI 98-201-11 (USQ Issue and EDG Dynamic Analysis)

Description of Item:

FSAR Section 8.3.1.2.4.c was changed to reflect the deviation from the RG (not meeting the 95 percent frequency requirement of RG 1.9). The Licensing Document Change Request (LDCR) 93-0094, dated August 29, 1994, and 10

CFR 50.59 safety evaluation dated January 19, 1994, documented this change. The safety evaluation did not identify this item as an unreviewed safety question (USQ) even though the safety margin as stated in the FSAR and the TS Bases was reduced and the possibility of a malfunction of equipment important to safety of a different type other than any evaluated in the FSAR was created because of the potential overlapping of diesel loads and the frequency response was less than that specified in RG 1.9. Waterford could not demonstrate that this scenario would be the worst case for frequency and voltage excursions for the EDG. The team also identified other inconsistencies concerning electrical loading data that was not current with the FSAR loading tables and accident scenarios.

The team identified the adequacy of EDG dynamic load study and the USQ issue as URI 50-382/98-201-011.

Item Discussion:

The design basis for the Waterford EDGs is found in UFSAR Section 8.3.1.2.4 which documents Waterford's compliance with RG 1.9. As stated in RG 1.9, the design basis requires the following:

1) Section B - "A diesel generator unit selected for use in an onsite electric power system should have the capability to (1) start and accelerate a number of large motor loads in rapid succession and be able to sustain the loss of all or any part of such loads and maintain voltage and frequency within acceptable limits and (2) supply power continuously to the equipment needed to maintain the plant in a safety condition if an extended loss of offsite power occurs;

2) Section C.4 - "...each diesel-generator unit should be capable of starting and accelerating to rated speed, in the required sequence, all the needed engineered safety feature and emergency shutdown loads. The diesel-generator unit design should be such that at no time during the loading sequence should the frequency and voltage decrease to less than 95 percent and 75 percent of nominal, respectively."

The acceptance criteria of RG 1.9 and the original design basis for the EDGs, and thus the basis for the Waterford TS surveillance assumed a step-load philosophy that recognizes the loads to be picked up by the EDGs will occur in a designated sequence.

Information Notice (IN) 92-53, "Potential Failure of Emergency Diesel Generators Due to Excessive Rate of Loading," dated July 29, 1992, was issued to alert licensees of the possibility EDGs could fail during a Loss-of-Offsite Power (LOOP) if certain electrical loads automatically started at the same time as other loads were sequenced onto the emergency buses. The particular concern was that simultaneous addition of significant loads onto the Emergency Diesel Generator (EDG) could cause it to fail. The IN noted the ability of the EDG to supply the required voltage and current could be degraded or lost if the rate of loading exceeded its design capacity.

In response to IN 92-53, Waterford initiated a transient analyses study to evaluate the dynamic performance of the EDGs under varying load conditions. This study used "state of the art", industry recognized software (Electromagnetic Transient Program (EMTP) Version 2) to simulate the Waterford EDGs and the large motor loads. The results of this study are documented in manual #460000011, "Emergency Diesel Generator Units Dynamic Loading Study," Revision 0.

The initial approach for the study was to establish the limiting case transient for the EDG and its connected loads. It was determined that simultaneous actuation of process loads on the EDG would have the longest recovery time. The bounding case was determined to be a combination of all permissive actuated loads as one load block at the end of the diesel loading sequence. It must be recognized this worse case scenario was beyond the RG 1.9 design basis of the EDGs which called for sequential loading of the EDGs rather than the simultaneous addition of the process loads.

As a result of this worse case scenario, Waterford determined the diesel frequency momentarily decreased below the requirements (95% of nominal) of RG 1.9. The dynamic study showed that the frequency dipped to 56Hz (93.3 percent), but the EDG was capable of recovering to its design frequency (60 hertz). The study further concluded that the duration of frequency deviation below 95% would be less than 3 seconds, running loads as well as accelerating loads would not be adversely impacted and the safety performance capabilities of the EDG would not be degraded.

Hence for the scenario postulated in the IN, Waterford analyses yielded a minor frequency deviation from the RG 1.9 recommendations. However, Waterford continued to comply with its design basis and the intent of RG 1.9 in that the study demonstrated that even with the simultaneous addition of the process loads, all started motors would accelerate to full speed and running loads as well as accelerating loads would not be adversely impacted.

Further evaluation to remove conservatism from the analyses, so as to meet the 95% requirement, was not performed as:

- The probability of all the process related loads actuating simultaneously towards the end of the sequencing cycle is extremely remote, and

- The impact on all the loads and the EDGs is insignificant.

The study concluded the Waterford diesel generators can adequately accelerate the required 1E loads during postulated events even in the remote event of simultaneous actuation of several large loads. The running loads as well as accelerating loads are not adversely impacted and the safety performance capabilities of the EDGs are not degraded. Hence, Waterford concluded the actions taken regarding IN 92-53 did not constitute a Unreviewed Safety Question (USQ) and Waterford remained within the design and licensing basis for the EDGs.

Waterford chose to document the results of these analyses in the FSAR even though this was clearly beyond the requirements of the Waterford licensing basis.

<u>Corrective Steps That Have Been Taken and the Results Achieved:</u>

At the time of inspection, the selection of the "worst case" scenario analyzed in the study was questioned. Preliminary computer studies using the database developed for the original study were performed to simulate lumped loads starting at different stages of the sequencing process. In particular, a case study of staggering process loads at the point of lowest voltage after the start of the first load (i.e. staggered loads with overlapping starts) was performed. The results of these preliminary studies indicated the originally documented study was the limiting case for frequency and voltage excursions.

<u>Additional Corrective Steps Planned To Be Taken:</u>

Waterford will enhance the formal calculation to confirm the "worst case" loading criteria for issues identified in IN 92-53. In addition, new case studies will be performed using current plant configuration to update the existing analyses.

Schedule of Completion of Remaining Corrective Actions:

The new case studies will be completed by 06/30/99.

URI 98-201-12 (10 CFR 50.59 Evaluation – Battery Modification)

Description of Item:

The team determined that the Waterford's 10 CFR 50.59 safety evaluation for modification package DC-3362 was inadequate since it did not identify the need for revising the TS to reflect the intercell connection resistance specified by the manufacturer. Further, the safety evaluation did not discuss the effect of the decrease in the safety-related battery duty cycle from 1 hour to 17.3 seconds for batteries 3A-S and 3B-S, and from 8 hours to 31 minutes for battery 3AB-S and the increased duration of battery recharge time. The calculations for battery and battery charger needed to be revised to show the loading, sizing, duty cycle and battery charger charging times.

The team identified this item as URI 50-382/98-201-12.

Item Discussion:

There are three different issues discussed under this item:

a) Battery intercell connection resistance.
b) Battery sizing calculations and UFSAR figures.
c) Battery charger sizing criteria.

a) Battery intercell connection resistance

The station batteries installed under modification package DC-3362 had their discharge characteristic curves calculated on the basis of a manufacturer supplied intercell connection resistance of 20-50 x 10^{-6} ohms. The battery voltage profile curve developed by the manufacturer was specific for the Waterford battery cells.

The Waterford TS 4.8.2.1.b.2 and 4.8.2.1.c.3 state the connection resistance between battery cells and terminals must be less than 150 x 10^{-6} ohms. Waterford disagrees that the 10 CFR 50.59 safety evaluation for modification package DC-3362 was inadequate since it did not identify the need for revising the TS to reflect the intercell connection specified by the manufacturer. This value is still commonly specified in

TS of most operating plants and the bases for this value is industry experience of battery systems. The industry application of this requirement is that a cell or two may approach the 150 micro ohm resistance value. The entire string of cells is expected to be maintained close to the manufacturer supplied values. Based on this understanding, Waterford concluded the TS to be adequate and did not submit a TS change to revise the parameters as part of DC-3362.

At Waterford, the intercell resistance measurement is part of the 18 month surveillance and the values are compared to baseline. A review of several sets of measurements did not reveal any upward trend on any of the connections. A quarterly surveillance is performed to check for any visible signs of corrosion (TS 4.8.2.1). It should be noted that if the intercell resistance values trend upwards, then the voltage parameters surveyed under TS Table 4.8-2 will be adversely impacted and corrective action will be initiated. Furthermore, the voltage profile for the discharge tests will indicate sharp degradation in battery performance.

b) Battery sizing calculations and UFSAR figures.

The battery profiles depicted in the UFSAR are specific to the installed batteries and are based on the current safety analysis and Station Blackout (SBO) requirements. The battery profiles in the UFSAR prior to installation of the new batteries were not specific to plant loads and operating conditions nor were they consistent with the safety analysis. There is no formal documentation for the exact bases for the old battery profiles. The profiles for the new batteries interjected a basis that reflects the current safety analyses.

The battery sizing calculations were based on an LOOP/LOCA or LOOP/MSLB scenario where the battery chargers are load shed from the AC system and the batteries are required to provide support for all the safety related loads. Once the diesels have started, two battery chargers will be automatically reenergized during sequencer step 2b. This will occur at the time of 17.3 seconds as depicted in the UFSAR Table 8.3-1. (Note, only one charger is required to carry shutdown loads and recharge the batteries. This is consistent with TS as only one charger is required to maintain the DC system operable). UFSAR Figures 8.3-2 and 8.3-3 depict the load profile for the duration the batteries are supplying load. Any failure of the battery charger(s) to sequence onto the diesel generator is considered a single failure and the redundant train will be available for safe shutdown. There is no design scenario that requires the batteries to supply accident loads for a period of 1 hour (A&B batteries) or 8 hours (AB battery), without charger support. This is consistent with licensing bases and industry standards (IEEE Std. 308 endorsed by RG 1.32 and IEEE Std. 946 1985) which require the design basis for each of the 125V DC batteries to have adequate capacity to supply the vital loads without charger support until AC power is restored.

Although the batteries are only required to supply accident loads for less than one minute without charger support, the battery sizing calculations consider that the peak load current lasts for one minute. This is consistent with the requirements of IEEE Std 485 1983 and is delineated in FSAR Section 8.3.2.1.1. The Waterford batteries are adequately sized to supply 4 hour SBO loads. As noted in the Inspection Report, Waterford considers TS surveillance testing for the SBO profiles (4 hours) envelopes the 1 minute Design Basis Accident (DBA) battery profiles depicted in the current UFSAR .

During the inspection, it was identified that the charger output current ramps up (a few seconds to deliver full load DC) after the AC power is restored. The battery will continue to be discharged due to spring charging current for the circuit breakers that are closed in the EDG sequencer step (2b) coincident with battery charger. Preliminary calculations demonstrated the total battery discharge current was below the peak load current used for battery sizing criteria. These calculations will be formally documented.

Waterford's position on sizing batteries for 1 minute for the DBA profile is consistent with design bases of some other utilities. This position was verbally discussed with NRR personnel at the time of corrective action for the Electrical Distribution Systems Functional Inspection (EDSFI) findings. At that time, it was concluded there was no documented bases for the 1 hour or 8 hour profile requirement as the SBO rule was enforced. Utilities needed to demonstrate that surveillance testing performed for the battery service test enveloped the requirements of the DBA and SBO profiles. If the 1 hour or 8 hour DBA load profile in the Waterford FSAR was to be enforced, then the battery surveillance tests for SBO profile could not be performed.

The safety evaluation for DC-3362 considered the design basis summarized above. It concluded the AB battery discharge profiles in the UFSAR (for LOCA/LOOP) were applicable for 17.3 seconds as compared to the 1 hour profile in the original FSAR. The AB battery profile was changed to 31 minutes instead of 8 hours to allow for bus transfer time.

c) Battery charger sizing criteria.

The Waterford design of the 125V DC system does not require batteries to supply normal power to Static Uninterruptible Power Supplies (SUPS) through the battery chargers. When AC power is restored to the 480V 1E buses, the SUPS have an independent AC rectifier supplying the inverter and do not need power from the DC system. Hence SUPS loads are not considered part of the battery load when AC power is restored to the 480V AC system. There was a concern expressed by the inspection team that the TS 3.8.3.1 LCOs allow indefinite plant operation with SUPS inverter connected to the DC source and the associated rectifier of the SUPS disabled. CR 98-0792 was initiated to document the discrepancy in the interpretation of TS 3.8.3.1. Loss of a SUPS rectifier at the onset of an event would be considered a single failure of that train and redundant equipment is available for safe shutdown.

The battery charger sizing calculations were revised subsequent to installation of DC 3362. The time to recharge a fully discharged battery increased from 11 hours to 22.19 hours for a single charger operation. This is acceptable as the recharge time is still greater than the minimum preferred recharge time of 8 hours (Ref. IEEE Std. 946). A fast recharge time is detrimental for the battery and an extremely slow recharge time may restrict plant operation. Note, Waterford batteries do not require significant recharge time post LOCA/LOOP event as AC power is restored within a minute for A&B batteries and within 31 minutes for the AB battery.

Corrective Steps That Have Been Taken and the Results Achieved:

a) Battery intercell connection resistance

As indicated in the inspection report, an evaluation of critical loads such as the EDG and inverter breaker closing coils was performed to determine whether they would have adequate voltages to operate with higher intercell resistance. It was established that there were adequate voltage at the terminals of the coils for breaker operation.

b) Battery sizing calculations and UFSAR figures.

Preliminary calculations were performed to demonstrate that DC spring charging current for breakers when the battery charger output current is ramping up, does not exceed the peak load considered for battery sizing criteria. It was determined that the batteries were adequately sized.

c) Battery charger sizing criteria.

CR 98-0792 was initiated for clarification of TS 3.8.3.1 for SUPS operation.

Additional Corrective Steps Planned To Be Taken:

a) Battery intercell connection resistance.

The maintenance procedures will be revised to incorporate the maximum intercell resistance that is acceptable without degrading battery performance capabilities.

Waterford will complete an evaluation of the basis for the intercell resistance value in the current TS.

b) Battery sizing calculations and UFSAR figures.

Waterford will update the battery sizing calculations to include the loads that are discharging the battery while the charger output current is ramping up.

c) Battery charger sizing criteria.

Waterford will submit a change to TS 3.8.3.1 to clarify the requirements for SUPS operation.

Schedule of Completion of Remaining Corrective Actions:

Administrative controls will be established for the intercell resistance values by 01/30/99. This will support battery tests to be performed in the next refueling outage.

All other items will be completed by 06/30/99.

IFI 98-201-13 (EDG Loading)

Description of Item:

Calculation EC-E90-006 did not evaluate the worst-case loading of the EDG caused by having only a single diesel generator operating, nor did it consider the maximum brake horsepower (bhp) requirements for major loads such as the EFW, LPSI, and containment spray (CS) motors.

The team determined that Waterford had not established or documented the maximum loading of the EDGs. The team identified this as IFI 50-382/98-201-13.

Item Discussion:

The inspectors reviewed calculation EC-E90-006. This calculation considers the STEADY STATE electrical loading of both emergency diesel generators for limiting accident conditions during a LOCA/LOOP, main steam line break (MSLB) with a LOOP, and shutdown with a LOOP. The calculation did not evaluate the worst-case loading of the EDG caused by having only a single diesel generator operating.

The focus of calculation EC-E90-006 is to analyze the worst case EDG loading from fuel oil consumption perspective. The tank capacity to store fuel requirements as delineated in RG 1.9 has been previously cited as not meeting regulatory commitments (Inspection Report 97-10). The limiting case for fuel oil consumption is with both EDGs operating to mitigate the consequences of an event. For a single failure of one EDG, the fuel oil in the tank associated with the failed unit would be available for the operating unit as no other failures are postulated and the cross connections in the fuel oil delivery system would be available. Thus, a single failure of an EDG to start is not a limiting case for the EDG system, which was the subject of calculation EC-E90-006.

The calculation further demonstrates that the maximum load on each EDG is less than 4150kW with large motor loads operating at steady state conditions. This load is significantly reduced for long term shutdown conditions, post accident, as decay heat and containment temperatures drop in magnitude. The EDGs are rated at 4400kW with a two-hour overload capability of 4840kW. Engineering judgement indicates that the electrical capability of each EDG would not be exceeded if some of the large pumps (e.g. EFW, LPSI, CS) operate at run out conditions for a short duration. Hence a 'worst case' electrical loading calculation for abnormal pump operation was not formally documented.

Corrective Steps That Have Been Taken and the Results Achieved:

As indicated in the inspection report, a preliminary evaluation with safety related large motors operating under maximum load conditions, determined that the maximum loading (approx.4360kW) was still within the nominal rating of the EDG.

Additional Corrective Steps Planned To Be Taken:

Waterford will formally document the worst case electrical loading of each EDG for postulated accident conditions.

Schedule of Completion of Remaining Corrective Actions:

The calculation for maximum load on each EDG will be completed by 6/30/99.

URI-98-201-14 (Non Safety Load Sequencing)

Description of Item:

The plant's normal computer supply, powered from 480 VAC bus 3A31-S cubicle 8C, tripped on loss of offsite power but automatically sequenced to the safety-related bus after 2 minutes. This automatic sequencing of nonsafety loads after load shedding during a loss of voltage is not in conformance with the FSAR Section 8.3.1.2.15(e)(7) which states that reconnection of non-essential loads can only be done manually under administrative control.

The team identified that non-Class 1E loads Diesel Generator 3A-S Air Compressor #1 and #2 are powered from the Class 1E bus portion of MCC 3A312-S even though a non-Class 1E bus is provided.

The team identified this item as URI 50-382/98-201-14.

Item Discussion:

The NRC identified some non-safety loads as being powered from the safety related bus, tripped on loss of offsite power and automatically sequenced back to the safety related bus.

The plant loads impacted by this NRC concern are non-safety loads that are important to the operation of Waterford. These loads typically consist of the Plant Computer Static Uninterruptible Power Supply system, the AB Static Uninterruptible Power Supply system, the Plant Security Static Uninterruptible Power Supply system, Emergency Diesel Generator Air Compressors, and other small PDP panels and emergency lighting panels. These loads have been analyzed for their impact on the Emergency Diesel Generator by the incorporation of their loads in the EDG loading calculation and utilizing isolation as described in our UFSAR and RG 1.75. This isolation minimizes the impact on the safety bus of a fault on the identified load. Although these loads have proper separation and are within the rating of the diesel generators, they do not meet the UFSAR notation with respect to automatically loading non-safety loads on the safety bus. Non-safety loads are either tripped on loss of voltage or are installed with two protective devices.

Corrective Steps That Have Been Taken and the Results Achieved:

Condition report (CR 98-0763) was written to address the issue. These loads have been evaluated as not required for safe shutdown but important to plant operations; however, this is inconsistent with the UFSAR. Waterford is preparing an UFSAR revision to describe which non-safety loads are to be automatically reloaded on the safety bus.

Additional Corrective Steps Planned To Be Taken:

Waterford will complete the analysis and revision of the UFSAR under the 10 CFR 50.59 process to describe automatic loading of non-safety loads on the safety bus.

Schedule of Completion of Remaining Corrective Actions:

Resolution of engineering issues will be completed by 10/30/99.

URI 98-201-15 (Station Battery Charger and Inverter Operation at Degraded Grid Voltage Conditions)

Description of Item:

The team reviewed calculation EC-E91-050 and noted that the voltages at safety-related MCCs powering the safety-related inverters and battery chargers were less than the required minimum voltage of 432 Vac during a plant degraded grid voltage situation. Insufficient ac voltage was available to operate the safety-related battery chargers and inverters and no analysis existed to verify that the safety-related dc system would be capable of powering the 125 Vdc and 120 Vac safety-related loads, which were normally powered by the battery charger and inverters.

The team identified this item as URI 50-382/98-201-15.

Item Discussion:

The degraded voltage calculation shows the expected voltages at the safety-related Motor Control Centers (MCCs) powering the safety-related inverters and battery chargers are less than the required minimum voltage of 432 Volts Alternating Current (VAC) during plant degraded voltage conditions. The calculation determines that during degraded voltage conditions, the voltage available at the safety-related MCCs powering the class 1E battery chargers and inverters ranges from 420 to 424 VAC for Train A and B and 417 to 421 VAC for Train AB. This condition was identified in 1992, during the calculation's preparation. The vendor for the instrument SUPS stated there would be no adverse effect on the operation of the safety related instrument SUPS. The documentation was not inserted into the calculation or vendor manual. Waterford has since received further documentation indicating the safety-related battery chargers and instrument SUPS units will function during the degraded voltage conditions described above. The information requires verification prior to insertion into the calculation and vendor manual.

Corrective Steps That Have Been Taken and the Results Achieved:

CR 98-0844 was written to address this concern. The immediate actions of the CR included a briefing of Operations to heighten their awareness of this condition.

Waterford has received preliminary vendor information and test reports that show that the chargers and the instrument SUPS will operate sufficiently during degraded voltage conditions.

Additional Corrective Steps Planned To Be Taken:

Additional corrective actions include the verification of vendor information and the update of affected Design Basis Documents. Licensing Basis Documents will be updated if necessary.

Schedule of Completion of Remaining Corrective Actions:

All actions will be completed by 09/30/99.

IFI 98-201-16 (EDG Load Sequencing Test Procedure)

Description of Item:

The team reviewed EDG sequencing test procedure OP-903-115, "Train A Integrated Emergency Diesel Generator/Engineering Safety Features Test", Revision 4, and determined that load shedding and sequencing of loads associated with load blocks 2a, 6b, and 6d are verified but the actual timing of the load blocks as stated in FSAR Table 8.3-1 was not documented. The team identified this item as IFI50-382/98-201-16.

NOTE: This discussion applies to 'A' train, 'B' train discussion is similar.

Item Discussion:

The loads specified in Load Blocks 2a, 6b and 6d of FSAR Table 8.3-1 require timing devices separate from the automatic load sequencer. Battery Charger 3AB1-S, Battery Charger 3AB2-S and the SUPS 3AB 30KVA Normal Supply are automatically loaded by individual timing relays mounted within the motor starters for these components. Essential Chiller compressors and oil pumps are automatically loaded via the program timer within the chiller control panel. Contacts of timing relays are verified to change state, but the actual time delay is not recorded (i.e. the timing function of these components is not verified).

A preliminary investigation has indicated timing of these components is not critical, since the loading of these components out of sequence (i.e. timing function not operating properly) would not cause adverse affects on the EDG. Procedure OP-903-115 is an integrated test where electrical busses are actually de-energized to simulate Loss of Offsite Power conditions concurrent with an SIAS actuation. This is a functional test where shutdown and emergency loads are sequenced on as required. Past functional testing per OP-903-115 has revealed no EDG anomalies.

We have determined timing of these components is important to ensure individual loads are functioning within their design parameters. A review of the affected components has revealed the timing testing for these components can be performed outside of OP-903-115. Repetitive tasks will be initiated to periodically verify proper timing of subject devices.

Corrective Steps That Have Been Taken and the Results Achieved:

ER-98-0773 was initiated to document the recommended method of verifying the timing function for these components.

Additional Corrective Steps Planned to be Taken:

Initiation of Repetitive Tasks to verify proper timing of subject components.

Schedule of Completion of Remaining Corrective Actions:

Repetitive Work Tasks to verify proper timing of components will be performed by 9/30/99.

The response to ER-98-0773 will be completed by 04/30/99.

IFI 98-201-17 (Battery Surveillance Test)

Description of Item:

The team reviewed WA 01156296, Task 001734, surveillance procedure ME-003-230, Revision 11, that was performed on May 22, 1997 to verify TS 4.8.2.1.d for battery 3AB-S. The DBA profile is defined as minute 0 to 1 (363 amps), minutes 1 to 30 (279 amps), and minutes 30 to 31 (342 amps). This SBO load profile testing did not envelop the last remaining one minute (30 to 31 minutes) load of the DBA profile.

Therefore, at minutes 30 to 31, the SBO load profile (279 amps) did not envelop the accident profile (342 amps).

The team identified this item as IFI 50-382/98-201-17.

Item Discussion:

TS 4.8.2.1.d requires that each battery bank (3A-S, 3B-S, and 3AB-S) be demonstrated operable at least once per 18 months, during shutdown, by verifying the battery capacity is adequate to supply and maintain in operable status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

The UFSAR specifies two design duty cycles for each of the Waterford safety-related batteries; one for a DBA and another for a Station Blackout (SBO). Waterford uses the SBO duty cycle as a basis for periodic Service Tests.

For battery 3AB-S, the duration of the DBA duty cycle specified in the UFSAR is 31 minutes and is defined as minute 0 to 1 (363 amps), minutes 1 to 30 (279 amps), and minute 30 to 31 (342 amps). The duration of the SBO duty cycle is 240 minutes and is defined as minute 0 to 1 (363 amps), minutes 1 to 224 (279 amps), minutes 224 to 239 (285 amps), and minute 239 to 240 (342 amps).

Waterford believes that basing the 3AB-S Service Test load profile on the SBO duty cycle results in a more limiting test and is therefore conservative. If the battery is capable of supplying the 342-amp load at minute 239-240 as required by the SBO duty cycle, then it is capable of supplying 342 amps at minute 30-31 of the DBA.

Therefore, Waterford believes the methodology of ME-003-230 is conservative and provides assurance battery 3AB-S is capable of supplying and maintaining the actual or simulated emergency loads for the design duty cycle.

Corrective Steps That Have Been Taken and the Results Achieved:

An analytical approach was used to determine the battery 3AB-S performance during a DBA. Specifically, Battpro 2.1 was used to show that battery terminal voltage would be 116.1 Volts Direct Current (VDC) at minute 31. Additionally, a Battpro 2.1 profile with a 342-amp load at minute 30-31 was added to the SBO duty cycle specified by the UFSAR. The results show that the lowest voltage obtained is 111.56 volts at minute 240. This is in excess of the required 108.6 VDC specified by Waterford calculations and the UFSAR.

Additional Corrective Steps Planned To Be Taken:

Corrective action is complete.

Schedule of Completion of Remaining Corrective Actions:

Corrective action is complete.

URI 98-201-18 (Instrument Accuracy)

Description of Item:

The team found that the flow and pressure indicators, ACC IFI7074A/B, CC IFI7070A1/B1, SI IFI1306, SI IFI1307, SI IFI0390A/B used for the ACCW, CCW, LPSI, HPSI performance testing and the charging pump discharge pressure test did not meet the ± 2 percent full-scale total loop accuracy requirements of OM-6. The total loop accuracy error ranged between 0.95 percent and 2.34 percent above the OM-6 requirement of ± 2 percent.

Instrumentation used for IST that had accuracies inconsistent with ASME Section XI Code requirements is identified as URI 50-382/98-201-18.

Item Discussion:

In accordance with 10 CFR 50.55a, Waterford was required to update its IST Program for the Second Ten-Year IST interval to meet the requirements of the 1989 Edition of ASME Section XI by December 1, 1997. Subsection IWP of this edition of the Code directs that pump testing be performed in accordance with the requirements stated in ASME/ANSI OM (Part 6)

In 1997, Waterford developed or modified plans and procedures as necessary to implement the new Code requirements beginning on December 1, 1997. During this process, it was decided that compliance with instrumentation accuracy requirements should be confirmed since the new Code provided more specific guidance about instrumentation accuracy than the Code used in the First Ten-Year IST interval (1980 Edition, through Winter 1981 Addenda). No instrument accuracy discrepancies were expected. A review of instrumentation accuracy during a major program update was thought to be a prudent action and not necessarily required for implementation of the update. Hence, ER-97-0390 was initiated on August 14, 1997, to evaluate permanent plant instrumentation used in IST to confirm that it met the applicable accuracy requirements as given in OM-6. As required, on December 1, 1997, Waterford implemented its second ten-year IST interval. Although the response to ER-97-0390 was not complete at that time, testing in accordance with the new program began.

On May 13, 1998, the NRC requested evidence Waterford performed Section XI pump testing with 2% accurate instrumentation. Prior to this request, work had been in progress to complete the ER response. On May 21, 1998, the ER response was essentially complete such that only final reviews and approvals remained. The results of the ER evaluation showed that some of the instruments used to perform pump IST did not meet the applicable accuracy requirements. Specifically, the ACCW, CCW, LPSI, and High Pressure Safety Injection (HPSI) pump flow, and the charging pump discharge pressure indications were not accurate to within ± 2%. The nonconforming instruments were documented in the Waterford corrective action process (CR 98-0734) on May 21, 1998. The minor deviations in required instrument accuracy presented in the ER evaluation resulted in minimal impact on pump test data when corrected and there were no operability concerns. On May 22, 1998, subsequent to completion of the CR operability assessment, a response to the question on instrumentation accuracy was provided to the NRC inspection team which included details of the ER evaluation and actions taken (condition report and operability information). The final approved response to the ER was completed on 6/4/98.

In summary, testing of the ACCW, CCW, LPSI, HPSI, and charging pumps was not in compliance with requirements as set forth in Part 6 of ASME/ANSI OM during the period from December 1, 1997 until discovery of the instrument accuracy non-conformance on May 21, 1998. An assessment showed that no operability concerns existed.

Corrective Steps That Have Been Taken and the Results Achieved:

- The nonconforming instruments were documented by CR 98-0734 on 5/21/98. Test data from the most recent surveillances of the affected pumps was adjusted to account for inaccuracies greater than allowed values. Pump performance remains acceptable. The effect of not complying with the specified accuracy requirements was evaluated and determined not to be safety significant. The causal determination and corrective action plan was approved on 7/17/98.

- The engineering review needed to confirm compliance with ASME/ANSI OM (Part 6) was documented by the ER-97-0390 response on 6/4/98.

- IST surveillance procedures for the ACCW, CCW, LPSI, HPSI, and charging pumps have been revised to specify the use of instrumentation meeting the applicable accuracy requirements. No revisions were required for the remaining IST pump test procedures since the instrumentation specified is acceptable.

Additional Corrective Steps Planned To Be Taken:

Corrective actions are complete.

Schedule of Completion of Remaining Corrective Actions:

Corrective actions are complete.

IFI 97-201-19 (UHS Basin Capacity)

Description of Item:

The water inventory requirements for the long-term cooling event described in Section 6.3.3.4 of the FSAR "Post-LOCA Long Term Cooling," identifies an EFW supply of 344,000 gallons, which equates to 170,000 gallons from the CSP and 174,000 gallons from the UHS basin. This volume exceeded Section 9.2.5.2 of the FSAR's large-break LOCA UHS water requirements and current TS bases (164,389 gallons). The FSAR scenario did not address water consumption associated with UHS heat dissipation of containment heat loads, and subsequent cooldown from hot standby conditions and SDC operation. Calculation MN(Q)-9-9 also identified that Section 6.3.3.4 of the FSAR requires 344,000 gallons for EFW, and states that this more conservative long-term cooling plan is still in effect, but did not use this as a basis for water requirements within this calculation. Although the calculation concluded that the large-break LOCA consumes 164,389 gallons for UHS heat dissipation, the calculation did not provide justification for exclusion of the small-break LOCA analysis, with 174,000 gallons as the required WCT Basin inventory.

The recognition of additional functional requirements for the WCT basin to support EFW system water storage needs, in addition to the UHS heat dissipation needs is identified as IFI 50-382/98-201-19.

Item Discussion:

The connections between the evaluations for plant water storage needs during accident conditions were not clearly made, although these evaluations did demonstrate the water storage needs were met. A comprehensive analysis of volume requirements for the WCT for the Branch Technical Position (BTP) 5-1 Natural Circulation Cooldown, tornado and other design basis events were not documented adequately for Waterford. There were several separate analyses that could be combined to make the necessary determinations. For example, the water required from the wet cooling tower basin for use of the UHS was not included in the BTP 5-1 analysis requirements, but it is included in other analyses. This issue had been identified in the design basis review efforts and resolution was in process prior to the A/E inspection. An analysis of the EFW requirements on the WCT basin volume considering the water consumption needs for shutting down the plant from accident initiation to cooling the RCS down to 200°F using the shutdown cooling system is under final review. The preliminary results confirm the current wet cooling tower basin volume and CSP volume given in TS 3/4.7.4 bound the water needs of the plant for all events and give clear documentation of this conclusion.

Corrective Steps That Have Been Taken and the Results Achieved:

The EFW requirements for the BTP 5-1 analysis and Long Term Cooling analysis have been completed. The results show that the water requirements for EFW used in the past were bounding.

Additional Corrective Steps Planned To Be Taken:

Clearly document the water consumption for the UHS combined with the EFW water requirements for all events.

The UFSAR and Tech Spec bases will be evaluated and revised as necessary to clearly reflect the inventory requirements of the WCT basins for EFW.

Schedule of Completion of Remaining Corrective Actions:

The analysis of the total water needs of the plant will be finalized and the evaluation of the UFSAR and TS bases will be completed by 06/30/99.

URI 97-201-20 (Procedure For SFP Cooling)

Description of Item:

Waterford indicated that SFP heat rejection would be controlled as part of the Emergency Plan (E-Plan) responsibilities by providing guidance to operations based on assessments of the meteorological conditions at the time of the event. Plant operating procedures did not address these restrictions explicitly. Waterford cited procedure EP-002-100, "Technical Support Center (TSC) Activation, Operation and Deactivation," Revision 26, as the applicable guidance procedure.

The guidance provided by procedure EP-002-100, Attachment 7.12, "Post Accident Contingencies and Concerns" indicated that Fuel Pool cooling should be restored before the Fuel Pool Temperature exceeds 180°F and that CCW flow through the standby fuel pool heat exchanger should be secured. EP-002-100 provided no procedural guidance or cautions to restrict SFP heat loading on the UHS to 11.4 million Btu/hr as used within the WCT and DCT capacity analysis and development of inputs for determination of UHS heat rejection water consumption. To maintain operation consistent with the FSAR, TS Bases and supporting analysis, SFP cooling operation restrictions must be applied.

The lack of procedural guidance regarding SFP cooling operation restrictions is identified as URI 50-382/98-201-20.

Item Discussion:

Waterford treats the spent fuel cooling needs after an accident as an Emergency Plan function. The procedure requires the technical support center and the emergency offsite facility to determine the necessary action based on the conditions that exist at the time of the accident. The inspection team identified that cooling of the Spent Fuel Pool could impact the UHS heat removal capacity. In addition, the inventory of water necessary for the cool down of the plant concurrent with the operation of the Spent Fuel Pool Cooling system might not preserve the assumption given in UHS design basis. Specific guidance is required to ensure the spent fuel cooling is controlled so the assumptions used in the design basis of the UHS are not violated.

Waterford will develop the necessary guidance in procedures to limit the Component Cooling Water (CCW) flow to the Spent Fuel Pool Cooling system to a constant rate that does not impact the assumptions used in the UHS design basis water consumption calculation. This guidance will be added to the appropriate procedures.

Corrective Steps That Have Been Taken and the Results Achieved:

The review of existing design information determined the UHS has the capacity to maintain the spent fuel pool temperature below the required limits, and could cool down the spent fuel pool.

Additional Corrective Steps Planned To Be Taken:

The necessary guidance will be developed and added to the appropriate procedures.

Schedule of Completion of Remaining Corrective Actions:

The procedure revision will be complete 02/01/99.

URI 97-201- 21 (SFP Makeup Requirements)

Description of Item:

The team concluded that the UHS volume requirements as described within the FSAR, TS bases and supporting design calculations were inconsistent with the facility design and licensing bases.

The failure to account for additional functional requirements for the WCT basin to support SFP system makeup water storage needs, in addition the UHS heat dissipation needs, is identified as URI 50-382/98-201-21.

Item Discussion:

Waterford meets the requirements of SRP 9.1.3 and RG 1.13 in that it has a Seismic Category I makeup system and an onsite Seismic Category I water storage facility as a backup source. Section 9.1.3.3 of the UFSAR identifies that makeup sources to the Spent Fuel Pool (SFP) are from the RWSP and the Seismic Category I CSP. The Waterford analysis, calculation EC-M97-006, has determined the WCT basins can be available for use in supplying makeup to the spent fuel pool during a DBA while still satisfying the UHS and EFW requirements. The WCT basins as makeup sources to the SFP are over and above those required by SRP 9.1.3 and RG 1.13. Although additional water is available from the WCT basins, it would not be appropriate to modify the UFSAR to include a description of this capability in the licensing basis simply because this design margin is available.

Corrective Actions that have been taken and Results Achieved:

Calculation EC-M97-006 was reviewed and it was determined it did not clearly define the design requirements for makeup to the SFP.

Additional Corrective Steps Planned to be Taken:

Calculation EC-M97-006 will be revised to clarify the design requirements for SFP makeup.

Schedule of Completion of Remaining Corrective Actions:

Calculation revisions will be approved by 05/01/99.

URI 97-201-22 (Potential Overpressure of Nitrogen System)

Description of Item:

"Emergency Feedwater control valves EFW-223A(B), EFW-224A(B), and EFW isolation valves, EFW-228A(B), EFW-229A(B), are controlled by instrument air and backed up by a nitrogen supply system.

The team identified that failure of a nonsafety-related pressure regulator in the nitrogen system could affect the safety functions of both trains of EFW system flow control and isolation valves since the relief valve to protect the safety system was also nonsafety, noncode, and undersized. According to the pressure rating of safety-related Class 3 components downstream (800 psig), this over-pressurization could jeopardize the capability of the isolation and the control valves of the EFW system to open or close during a design-basis event. Waterford determined that over-pressurization could render the nitrogen system inoperable and subsequently cause connected safety-related systems to become inoperable.

The potential over-pressurization of the nitrogen system that could affect several safety-related systems is identified as URI 50-382/98-201-22."

Item Discussion:

During the A/E Inspection, the potential for over-pressurization of the safety-related portion of the plant's nitrogen system, due to a single failure in the non-safety portion, was discovered. The failure of a single, non-safety pressure regulating valve could cause nitrogen system pressure to exceed its design pressure rating and potentially render multiple safety systems of separate trains inoperable. The regulating valve reduces nitrogen pressure in the non-safety system from over 2000 psig to approximately 750 psig, for delivery to multiple loads, including safety related air operated valve accumulators for both A and B trains. A relief valve existed downstream of the regulator for protection of the downstream systems. However, this relief valve was found to be undersized, non ASME and its relief setpoint was above that required for system protection. LER 98-010 was submitted to the NRC.

In order to immediately correct this deficiency, an appropriately sized ASME Section VIII relief valve in a connecting line was placed into service. However, it was later discovered the isolation valve upstream of the relief valve would restrict flow to the relief valve, potentially causing pressure to rise above design pressure of the nitrogen system, if the regulating valve failed. A second condition report was written and the immediate action taken was to swap nitrogen supply from the liquid supply system to the bulk nitrogen truck stationed on site. In this line up the ASME Section VIII relief valve was then adequate to protect the system when supplied from the bulk nitrogen truck since the flow restriction was not between the high pressure source and the relief valve.

On May 21, 1998 the original, undersized, non-ASME relief valve downstream of the pressure regulating valve was replaced with an appropriately sized, ASME Section VIII relief valve and the liquid supply system was returned to service.

Past operation was also reviewed and the safety related components supplied by the nitrogen system were analyzed to determine the impact of the potential over-pressurization. The capability of performing safety functions of the applicable components under the postulated over pressure condition was evaluated. Design limits would have been exceeded but the components would have remained operable and components necessary for safe shutdown of the plant would have operated as required. This includes the low-pressure nitrogen regulating and isolation valves in the safety-related portion of the nitrogen system, which regulate pressure from the individual accumulators to the air operated valve actuators.

To fully restore the system to code compliance, a new properly sized and set ASME Safety Class 3 relief valve has been added to the supply line to the safety-related equipment.

Corrective Steps That Have Been Taken and the Results Achieved:

The undersized, non-ASME relief valve was promptly replaced with an ASME Section VIII valve of the proper set pressure and flow capacity. This allowed the nitrogen supply system to be returned to service.

The ASME Safety Class 3 relief valve was originally scheduled to be installed during the upcoming refueling outage (scheduled for February of 1999). However, during a forced outage in September of 1998, Waterford installed the new valve. This valve provides safety related relief protection for the safety-related components supplied by the nitrogen system.

CR 97-2551 identified potential problems with failures affecting interfacing systems. One of the corrective actions from this CR was to perform an evaluation of the impact of failures on the interfacing risk significant mechanical systems. This evaluation is in progress.

Additional Corrective Steps Planned To Be Taken:

Review and approve the evaluation on the impact of failures in systems that interface with risk significant mechanical systems. This evaluation will look for single failures that may not have been previously evaluated.

Schedule of Completion of Remaining Corrective Actions:

This evaluation will be completed by 6/30/99.

URI 97-201-23 (EFW Pump Room HVAC)

Description of Item:

The EFW motor-driven pump rooms are cooled by chilled water. Waterford has documented in calculations MN(Q)- 9-3, "Ultimate Heat Sink Study," Revision 2, and MN(Q)-9-17, "Tornado Multiple Missile Protection of Cooling Towers," Revision 2, that during post LOCA or following events such as a tornado, the chilled water temperature may rise above its maximum design temperature of 42°F (Design Specification 1564.747,"Water Chillers," Revision

15) and may reach temperatures as high as 52 °F . But when calculating (Calculation 5I, "Emergency Feedwater Pump Rooms," Revision 2) the maximum temperature in the EFW motor-driven pump rooms, a nonconservative chilled water temperature of 42 °F was used.

The team identified this issue as URI 50-382/98-201-23

Item Discussion:

Existing analysis indicates post-tornado shutdown conditions with high ACCW water temperature to the Essential Chilled Water (CHW) condenser will cause chilled water to increase 10°F above its normal operating temperature of 42°F. This scenario could cause several areas containing safety-related equipment to exceed the 104°F normal room temperature. Calculation 5-I does not determine the maximum temperature in the EFW motor-driven pump room as stated in NRC Inspection Report No. 50-382/98-201, but rather the maximum heat load. The 42°F chilled water temperature is not an input to calculation 5-I. However, calculation 5-W does determine the maximum temperatures of the CCW Pump Room, the EFW Pump Rooms, and the Shutdown Heat Exchanger Room during post-tornado shutdown conditions. Calculation 5-W determines the post tornado temperatures using 52°F chiller water will be 116°F, 117 °F and 108 °F respectively. However, the evaluation focused on equipment qualification and did not address the maximum temperature limits at which the equipment could perform its safety function. CR 98-0852 was initiated, and the operability evaluation revealed the affected equipment would satisfy its safety function. Memo #EQP-82-9-131, dated September 29, 1982, which is referenced in 5-W used an Arrhenius methodology analysis (or relative aging analysis) to determine the equipment in the subject rooms is analyzed for the temporary excursions above design basis temperature.

Corrective Steps That Have Been Taken and the Results Achieved:

An initial analysis was performed which showed the equipment would not exceed design temperature limitations.

Additional Corrective Steps Planned To Be Taken:

The corrective actions resulting from the CR 98-0852 require a revision of calculation 5-W, and an evaluation to determine the elevated temperature effect on equipment important to safety in the CCW Pump Room, the EFW Pump Room, and the Shutdown Heat Exchanger Room.

Schedule of Completion of Remaining Corrective Action:

The revision to calculation 5W will be completed by 6/30/99.

The documentation of the effect on the equipment in these rooms will be completed by 6/30/99.

IFI 97-201-24 (CSP Water Level)

Description of Item:

When EFW flow is established using water from ACCW, the CSP is getting filled at the same time, but the operating procedure (OP-902-005, "Loss of Offsite Power / Station Blackout Recovery Procedure," Revision 9, dated December 1, 1995) directing the operators to establish this flow, did not provide any instruction to the operators to monitor water level in the CSP. This condition is aggravated by the fact that level monitoring is not possible during this operation because the level instrument would be biased with a large error indicating that the CSP water level is high off scale. If the operators do not monitor the CSP's water level, there could be a potential for overfilling and damaging the CSP (refer Section E1.2.2.2.a for details).

The adequacy of procedures to monitor CSP level when filled by ACCW is identified as IFI 50-382/98-201-24.

Item Discussion:

Because of the piping configuration, when makeup to the CSP is provided by the ACC system, ACC water will flow back through the suction header and refill the CSP. This back flow will affect the level instruments because of the line pressure. The level indication could be approximately 25% higher than actual. This error makes it difficult for operators to determine how much water has been added to the CSP. Although the level indication would be higher, confusion could result in transferring more water to the CSP from the Wet Cooling Tower (WCT) basin than necessary. This could potentially leave the UHS with inadequate water inventory to meet its design requirements. As a contingency, guidance was provided to the operators to align the ACC system for a maximum of 30 minutes if makeup is required to the CSP. Analysis shows the CSP would be sufficiently filled in that time frame to allow several hours of EFW flow. Thirty minutes is sufficient time to assure makeup for EFW and maintain UHS inventory requirements.

Corrective Steps That Have Been Taken and the Results Achieved:

Analyses were performed to determine the appropriate amount of time that the ACC system can be aligned to the CSP without potentially transferring more water out of the WCT basin than the inventory requirements allow. This resulted in the thirty-minute recommendation to Operations. A step to close the appropriate valves when 30 minutes has elapsed was added to Appendix 10 of the EOPs to give this guidance. An analysis to determine the time to refill the CSP during the described scenario, which would then provide a basis of operating procedure revision, was previously planned as part of the design basis reconstitution effort as open item EFW-130.

Additional Corrective Steps Planned To Be Taken:

An engineering request (ER-98-0876) was written to relocate the level instrument sensing lines to stagnant headers. Relocation of the sensing points to piping not affected by EFW flow will correct these issues. Once the sensing points are relocated, the notes and steps in the operations procedures will be removed.

Schedule of Completion of Remaining Corrective Actions:

The relocation of the level instrument sensing points is expected to be completed by Cycle 10 but not later than Refuel 10.

IFI 97-201-25 (ACCW – EFW Suction Path Testing)

Description of Item:

When water in the CSP is depleted (at 25 percent level), Waterford relies on establishing EFW flow using the WCT basins in accordance with operating procedure, OP-902-005, "Loss of Offsite Power- Station Blackout," Revision 9. This flow is established by running the ACCW pumps and opening isolation valve ACC-116A/B. When opened, this valve exposes the suction piping of EFW to the discharge piping of ACCW. Excess flow would then refill the CSP through the EFW suction pipes at the same time feeding the steam generators.

The piping arrangement connecting the ACCW to EFW has never been flow tested to ensure that the full design flow would pass through the lines. The team determined that Waterford's existing testing was not adequate to ensure that the full design flow would pass through the lines.

The team identified this issue as IFI 50-382/98-201-25.

Item Discussion:

Full flow testing of the EFW suction flow path from ACCW is not practical. Testing the system as described would unnecessarily expose the steam generators to biological contamination, which could have a long-term affect on equipment performance.

In lieu of full flow testing, the following items are performed to ensure the ability of the ACCW to supply the required EFW suction flow:

- Periodically, the ACCW suction supply to the EFW pumps is back flushed through drain valves ACC-115A(B). Although this flush is at low velocity, it affords a means to remove solids accumulated in the normally stagnant piping.

- ACC-114A(B) and ACC-116A(B) are periodically manually exercised to verify operability.

The ACCW system at Waterford is a recirculation system with an open basin as opposed to an open service water system. A side stream re-circulating filtration system was installed at the Wet Cooling Tower Basins (the ACCW pump intake) to remove particulates, including biologic and debris. The ACCW is not subject to silting and macrofouling. The incidence of suspended solids is low (as indicated from chemistry department trends). Therefore, degradation or blockage of the piping that could affect flow would be minimal.

Corrective Action Steps That Have Been Taken:

The above actions were in place prior to the A/E inspection.

Additional Corrective Action Steps Planned To Be Taken:

A formalized calculation to document the ability of the ACCW to supply the required EFW flow rate will be performed (Design Basis Review Open Item No. EFW-130). The calculation will use friction loss and diameter reduction values to model bounding maximum flow resistance due to potential MIC corrosion byproducts (sessile) in the piping. Similar modeling techniques have been used successfully in designing open service water systems for safety-related applications at other nuclear facilities where the use of biocide was not anticipated.

The above actions ensure the EFW suction supply from ACCW will meet its safety-related design basis, while precluding the potential detrimental effects resulting from full flow testing the ACCW supply through to the EFW system.

Schedule of Completion of Remaining Corrective Actions:

The calculation (EC-M98-013) to document the ability of the ACCW to supply the required EFW flow will be completed by 6/30/99.

IFI 97-201-26 (WCT Basin/ACCW Pump Vortex)

Description of Item:

Calculation MN(Q)-9-38, "Capacity of Wet Cooling Towers," Revision 3, established a TS capacity for the WCT at 174,002 gallons considering a full useable volume from the TS minimum water level (-9.86 ft), to the ACCW pump suction curb height. Assumption 5.1 and 5.2 in the above calculation indicated the ACCW pump suction is at the same elevation as the suction curb and the WCT is assumed to be empty at this elevation. Therefore, it appears that the TS WCT volume basis considers the water down to the suction lip as the useable volume and has no identified allowance for vortex formation.

The team identified this issue as IFI 50-382/98-201-26.

Item Discussion:

The analysis for the ACC basin vortexing assumed that since a vortex breaker was installed, an unusable volume to account for vortexing was not required. The existing vortex calculation (EC-M95-012) was revised to determine the minimum level in the wet cooling tower basin to prevent vortexing. The unusable volume of water to account for vortexing is bounded by the unusable volume assumed for suspended solids.

Corrective Steps That Have Been Taken and the Results Achieved:

The revised calculation determined that the level at which vortexing occurs does not impact the useable WCT water volume. The required water inventory is available between the TS minimum level and the level at which vortexing occurs.

Additional Corrective Steps Planned To Be Taken:

All corrective actions are complete.

Schedule of Completion of Remaining Corrective Actions:

All corrective actions are complete.

URI 98-201-27 (ACCW Transfer/Cross-connect)

Description of Item:

Water inventory management and cross-connect operation are addressed in the Emergency Plan. The licensee cited procedure EP-002-100, "Technical Support Center (TSC) Activation, Operation, and Deactivation," Revision 26, dated October 13, 1997, as the applicable procedure for guidance. However, the team noted that the guidance provided by procedure EP-002-100, Attachment 7.12, "Post Accident Contingencies and Concerns," indicated that makeup to the operable WCT basin via the unavailable WCT basin (cross-connect operation) should be established if the operating basin level is less than 5 percent and the Essential Chiller is using ACCW for cooling. The team did not agree with the licensee's position. These instructions did not address the concerns for the transfer of WCT water to the CSP because of the potential for rapid pump down of the basin and limited makeup capability by the WCT cross-connect. Further, the procedure did not address how the cross-connect operation is to be performed.

The lack of procedural guidance regarding ACCW Transfer to the CSP and Basin Cross-connect Operation is identified as URI 50-382/98-201-27.

Item Discussion:

The transfer (when and how) of water from one wet tower basin to the other basin is determined by the technical support center. No specific guidance is provided in the procedures.

Procedural guidance will be developed and added to the appropriate procedures. This guidance will indicate when the cross connecting of the two basins should be considered, the valve lineup to effect the cross connection, and the expected results. An analysis will determine possible usage rates from the basin and predict the basin level where there will be enough surge capacity to allow the cross connect to supply the necessary water.

The total water consumption analysis will confirm that the high usage rate of water from the wet tower basin will not be applicable at the time the basin will reach the 5% level.

Corrective Steps That Have Been Taken and the Results Achieved:

This issue was evaluated to determine the timing of the flow demands on the WCT basin and the proper actions to meet those demands.

Additional Corrective Steps Planned To Be Taken:

The procedural guidance will be developed and added to the appropriate procedures.

Schedule of Completion of Remaining Corrective Actions:

The revisions to the procedures will be complete 6/30/99.

URI 97-201-28 (UHS Basin Temperature)

Description of Item:

Calculation MN(Q)-9-17, "Tornado Multiple Missile Protection of UHS", Revision 2, dated 9-26-95 determined the performance of the UHS during shutdown following a tornado scenario. Within the first 2 hours of the event analysis, the DCT is not available, the WCT in the natural draft mode may not be capable of rejecting the entire heat load, and the remaining heat load is absorbed by the water volume of a WCT basin, which may result in a temperature rise above 105°F. However, the calculation preserved the assumption of 105°F by considering the entire heat load to be absorbed by the WCT basin volume resulting in a 25°F temperature rise from an initial temperature of 80°F to 105°F. The team noted that the current TS 3/4 7.4 (Ultimate Heat Sink) requires that the average basin water temperature be maintained less than or equal to 89°F. An initial basin water temperature of 89°F with the projected first 2-hour post-tornado heat load would result in cooling water temperature of 114°F to the Essential Chillers. The use of non-conservative temperatures in lieu of TS limiting values is identified as URI 50-382/98-201-28.

Item Discussion:

The tornado calculation documenting the UHS response shows the basin water temperature may rise to a temperature of 114°F with the tornado heat loads on the system. The affects of 120°F (bounding) inlet water to the Essential Chillers have been justified. Information in the calculation confirms that adding the total heat load on the ACCW system in the first 2 hours to the WCT basin assuming an initial basin temperature of 80°F results in a final temperature of 105°F. The 80°F was used as an additional data point and was not used in any design basis calculation. It was not intended as an input to determine ACCW inlet temperature to the essential chiller temperature from the UHS. This additional information was also not intended to justify the 114°F temperature from the UHS test.

Corrective Steps That Have Been Taken and the Results Achieved:

Calculation (MN(Q)-9-17) was reviewed. The use of the appropriate TS temperature assumption will have no effect on the calculation results.

Additional Corrective Steps Planned To Be Taken:

Corrective action is complete.

Schedule of Completion of Remaining Corrective Actions:

Corrective action is complete.

ATTACHMENT 2

ACRONYMS

ac, AC	Alternating Current
ACC	Auxiliary Component Cooling
ACCW	Auxiliary Component Cooling Water
ADC	Amps Direct Current
A/E	Architectural/Engineering
ANSI/ANS	American Nuclear Standards Institute/American Nuclear Society
ASME	American Society of Mechanical Engineers
bhp	Brake Horse Power
BTP	Branch Technical Position
CA	Corrective Action
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CHW	Essential Chilled Water
CR	Condition Report
CS	Containment Spray
CSP	Condensate Storage Pool
DBA	Design Basis Accident
DBR	Design Basis Review
dc, DC	Direct Current
DCT	Dry Cooling Tower
E-Plan	Emergency Plan
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDSFI	Electrical Distribution Systems Functional Inspection
EFAS	Emergency Feedwater Actuation System
EFW	Emergency Feedwater
EOP	Emergency Operating Procedure
ER	Engineering Request
ESF	Engineered Safety Features
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
gpm	Gallons per Minute
HPSI	High Pressure Safety Injection
HVAC	Heating,Ventilation,Airconditioning
IEEE	Institute of Electrical and Electronic Engineers
IFI	Inspection Followup Item

ILRT	Integrated Leak Rate Test
ILT	Instrument – Level Transmitter
IN	Information Notice
IST	Inservice Test
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Condition of Operation
LDCR	Licensing Document Change Request
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LPSI	Low Pressure Safety Injection
MCC	Motor Control Center
MOV	Motor Operated Valve
MS	Main Steam
MSLB	Main Steam Line Break
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUREG	NRC Technical Report Designation
PDP	Power Distribution Panels
PCT	Peak Clad Temperature
PRA	Probabilistic Risk Assessment
psig	Pounds per Square inch Gage
RAB	Reactor Auxiliary Building
RAS	Recirculation Actuation Signal
RCS	Reactor Coolant System
RG	Regulatory Guide
RWSP	Refueling Water Storage Pool
RWT	Repetitive Work Task
SAR	Safety Analysis Report
SBO	Station Blackout
SCCM	Standard Cubic Centimeters per Minute
SDC	Shutdown Cooling
SFP	Spent Fuel Pool
SG	Steam Generator
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SRP	Standard Review Plan
SSFI	Safety System Functional Inspection
SUPS	Static Uninterruptible Power System
TDH	Total Dynamic Head
TS	Technical Specifications

TSC	Technical Support Center
UHS	Ultimate Heat Sink
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USQ	Unreviewed Safety Question
VAC	Volts Alternating Current
VDC	Volts Direct Current
WA	Work Authorization
WCT	Wet Cooling Tower
WR	Wide Range