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EXECUTIVE SUMMARY

From August 4, through September 12, 1997, the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (NRR), Events Assessment, Generic Communications, and Special Inspection Branch, conducted a design inspection at D. C. Cook Nuclear Power Plant, Units 1 & 2. The inspection team consisted of a team leader from NRR and five contractor engineers from the Stone & Webster Engineering Corporation (SWEC).

The primary objective of the design inspection was to evaluate the capability of the selected systems to perform the safety functions required by their design bases, the adherence of the systems to their design and licensing bases, and the consistency of the as-built configuration and system operations with the Updated Final Safety Analysis Report (UFSAR). For the purpose of this inspection, the team selected the Residual Heat Removal (RHR) System and the associated Emergency Core Cooling System (ECCS) functions, and the Component Cooling Water (CCW) System, based on their risk significance and importance in accident mitigation. In addition, the team evaluated the interfacing safety functions and support systems associated with the RHR and CCW systems.

The team's findings included various issues in the areas of design and procedural control, safety evaluations, use of engineering judgement and operability determinations, temporary modifications, and consistency between the UFSAR and Technical Specifications (TS). During the course of the inspection, the licensee initiated approximately fourteen 1 and 4-hour non-emergency event notifications to the NRC for operation outside the design basis, or operation in an unanalyzed condition, in accordance with the requirements of 10 CFR Part 50.72. The licensee made many of these event notifications as a result of their review of team concerns and questions.

Some of the issues indicate that the ECCS system may not have performed its safety function under all design basis accident scenarios. Examples of these findings are listed below.

(1) The licensee failed to account for ECCS instrumentation uncertainties and bias in the plant-specific EOP decision criteria, and to establish the proper refueling water storage tank (RWST) and containment level setpoints necessary to preclude vortexing in the containment sump. These deficiencies could allow the operators to prematurely perform the ECCS switchover to the LOCA recirculation mode of operation without adequate water level in the containment, and also increase the probability that the RHR pumps could be damaged due to vortex-induced failure.

(2) Since 1992, there was a single failure vulnerability that could render both trains of safety injection and centrifugal charging pumps inoperable.

(3) Fibrous insulation material from containment cable trays could potentially be swept into and block in excess of the design value of 50 percent of the containment recirculation sump screen area.

(4) Until recently, the licensee failed to correct a long-standing condition regarding the 1/4-inch particulate retention design basis value for the containment recirculation sump that created a common-mode failure vulnerability, which could have potentially clogged redundant trains of ECCS throttle injection valves and containment spray nozzles.

On September 8, 1997, the licensee initiated a dual unit shutdown, and issued a notification of an unusual event (NOUE), as a result of the licensee's inability to demonstrate to the team, using design basis documentation, that the ECCS system would have performed its safety function during post-LOCA conditions.

The team identified multiple examples of the licensee's failure to adequately perform 10 CFR Part 50.59 evaluations, as well as several apparent 10 CFR Part 50.59 unreviewed safety questions (USQs). For example, the licensee was operating the plant since 1988 above the design basis ultimate heat sink temperature of 76°F without having performed a 10 CFR 50.59 evaluation, and without considering the impact this would have on overall plant operation. An apparent USQ was created in 1988, when the plant operated for 22 days with an averaged ultimate heat sink (UHS) temperature of 81°F, creating the potential for the safety-related equipment in the control room to not perform its safety function under design basis assumptions. In addition, the licensee could not demonstrate that they could achieve a TS-required plant cooldown in 36 hours to 200°F using design basis assumptions at the elevated UHS temperatures or at the original design basis UHS temperature of 76°F.

Contrary to the assumptions stated in Chapter 9 of the UFSAR, during the Unit 2 1996 refueling outage, both CCW and ESW trains were removed from service, with the intention by the licensee on performing a dual CCW/ESW train outage. Although the dual train outage was not fully sustained as originally planned by the licensee, this operational condition would have placed the plant at increased risk, outside of its design basis, and in an unanalyzed condition. The licensee also failed to consider in their safety evaluation to support the Unit 2 full core offload the time-to boiling criteria as stated in Chapter 9 of the UFSAR.

The team identified other issues concerning: (1) failure by the licensee to analyze all the potential instrument air system failure modes that could render redundant trains of safety-related equipment inoperable, with the licensee consequently identifying a single failure vulnerability that could render the emergency safety feature (ESF) ventilation system inoperable, (2) failure to account for instrument uncertainties that affect safety-related operating procedures, calculations, and TS surveillances, (3) failure to correctly model the design performance characteristics of the CCW heat exchangers and to incorporate this information into appropriate accident analyses, plant modifications, and calculations, (4) failure to evaluate a long standing condition of low and high flows to some CCW-supplied components and to the CCW heat exchangers, (5) failure to identify ASA B31.1 and ASME

VIII code deficiencies associated with the RHR and CCW systems that increased the potential for overpressurization of these two systems, and (6) failure to maintain proper control of abandoned plant equipment, some of which has been abandoned since the early 1980s.

E1.0 Inspection Scope and Methodology

The primary objective of the design inspection at the D.C. Cook Nuclear Power Plant, Units 1 & 2, (CNP) was to evaluate the capability of the selected system to perform the safety functions required by their design bases, the adherence of the systems to their design and licensing bases, and the consistency of the as-built configuration and system operations with the Updated Final Safety Analysis Report (UFSAR).

The staff of the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation (NRR), selected two risk-significant systems to perform an in-depth, vertical slice review. These systems were the Residual Heat Removal (RHR) System, with the associated Emergency Core Cooling System (ECCS), and the Component Cooling Water (CCW) System. For guidance in performing the inspection, the team followed the applicable engineering design and configuration control portions of Inspection Procedure (IP) 93801, Safety System Functional Inspection (SSFI).

Appendix A identifies the open items and issues resulting from this inspection, and Appendix B lists the attendees of the exit meeting on September 12, 1997. Appendix C defines the various acronyms and abbreviations used in this report.

E1.1 Residual Heat Removal (RHR)

E1.1.1 Mechanical

E1.1.1.1 Scope of Review

The mechanical design review of the RHR system included design and licensing documentation reviews, system walkdowns, and discussions with cognizant system and design engineers. The team reviewed applicable portions of the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), Design Basis Documents (DBDs), flow and process diagrams and other system drawings, calculations, design change packages (DCPs), requests for change (RFC) packages, system operating procedures, inservice and surveillance test procedures, emergency operating procedures (EOPs), corrective action program documents, and operating experience (OE) reviews. The scope of the review included confirmation that design bases for the RHR system and related ECCS functions were adequately translated into the licensing bases, verification of the appropriateness and correctness of design assumptions, boundary conditions, and system models, and verification of the adequacy of testing requirements. Systems interfacing with the RHR system were reviewed to verify that the interfaces were consistent with the RHR system design and licensing bases and would not have an adverse effect on RHR safety functions. The ability of the RHR system to support operation of the centrifugal charging (CC) pumps and the safety injection (SI) pumps during the recirculation phase of a loss of coolant accident (LOCA) was also examined.

Specific topical areas covered during the mechanical design review included system thermal/hydraulic performance requirements (e.g., system capacity, pump net positive suction head (NPSH), and pump minimum flow); system design

pressure and temperature; overpressure protection; component safety and seismic classifications; component and piping design codes and standards; and single failure vulnerability.

E1.1.1.2 Findings

The team identified a number of interrelated issues that could have prevented the ECCS system from performing its safety function. These issues involved the RWST and containment recirculation sump level instrumentation, containment water inventory and volume, ECCS performance analysis deficiencies, an ECCS single failure vulnerability, potential back leakage to the RWST, and deficiencies associated with Emergency Operating Procedure (EOP) 01(02)-OHP 4023.ES-1.3, "Transfer to Cold Leg Recirculation," Revision 4, (henceforth referred to as the EOP) involving the manual transfer of the suction of the ECCS pumps from the RWST to the containment recirculation sump for postulated LOCA scenarios. Each of these issues are discussed below. Because of these interrelationships, a brief overview of the LOCA accident scenario is provided below.

Background - Overview of the LOCA Sequence

Following a postulated LOCA, operation of the ECCS can be divided into two distinct phases, the injection phase, and the recirculation phase. During the injection phase, all ECCS pumps, as well as the containment spray system (CTS) pumps, take their suction from a 350,000 gallon refueling water storage tank (RWST) via branch lines off a common 24-inch suction pipe from the bottom of the RWST. During the postulated LOCA scenario, borated water from the accumulators, reactor coolant system (RCS), and the RWST, combine with the melted borated ice water from the ice condenser (no credit for ice melt water volume assumed in the accident analysis), and is collected in the containment recirculation sump. Near the completion of the LOCA injection phase, when the RWST inventory has been nearly depleted, the suction of the RHR and CTS pumps is manually switched to the containment recirculation sump, beginning the LOCA recirculation phase. The operator in the control room accomplishes the changeover from injection to recirculation by a series of manual operator actions, as specified in the EOP. In accordance with this procedure, when the RWST water level decreases to the low alarm setpoint (nominally 32.23 % of span), the operator transfers the suction of the west RHR and CTS pumps first to the sump, and then both trains of the CC and SI systems' pumps are "piggy-backed" onto the west RHR pump, where they take suction from the discharge of the West RHR pump. When the RWST water level reaches 10% of span, the operator next transfers the east RHR and CTS pump suctions to the containment recirculation sump, after which the operators re-separate the CC and SI trains, and the suction lines from the RWST are isolated. The RHR pumps are automatically tripped on RWST low-low level (nominally 9.09% of span) as a backup feature, to protect the pumps from cavitation.

Hydraulic model testing of the licensee's containment recirculation sump performance during postulated LOCA scenarios was performed by the Alden Research Laboratory and documented in a September 1978 test report. The test report states that the tests were performed using a post-LOCA containment water level of elevation 602'10" and that at this elevation, the sump would

perform satisfactorily without the development of any severe vortices or other flow irregularities. The potential for vortices to exist for containment water levels lower than elevation 602'10" was not precluded by the Alden model tests. One recommendation from the Alden Report was the installment of vents in the roof of the sump as a performance enhancement.

A. RWST Level Instrumentation, RWST Volume, and Vortexing Issues

The team identified concerns with the RWST level instrumentation, the adequacy of the information provided to the operators, the adequacy of the surveillance on the TS-required 350,000 gallon RWST requirement, and the potential for vortex-induced pump failure during the switchover from the injection phase of a LOCA.

(1) RWST Level Setpoint Error Due to Flow-Induced Effects

During the injection phase of a large-break LOCA, both RHR pumps are taking suction off the RWST and cause relatively high flow rates to occur in the 24-inch ECCS suction pipe from the RWST. Near the termination of the injection phase, the operators prepare to switchover the ECCS pump suctions from the RWST to the containment recirculation sump. The switchover of the ECCS and CTS pump suctions is directed by the EOP, and is a completely manual, operator-performed evolution which is dependent on the information provided to the operator by two RWST level instrumentation channels, ILS-950 and ILS-951. The pressure taps for the two differential pressure (dP) transmitters used to determine RWST level are located on the ECCS suction pipe connected to the RWST. The high flow rates through the suction pipe result in a significant friction head loss between the tank and the pressure taps. This friction head loss negatively biases the static pressure sensed by the dP transmitters, resulting in an indicated RWST water level that is lower than the actual level.

This effect was previously identified to the licensee during the 1993 NRC System Based Instrumentation and Control Inspection (SBICI, Inspection Report No. 315-316/93-12), and was documented by the licensee in Condition Report (CR) 93-1212. In response to the SBICI finding, the licensee added Appendix A to Instrument Uncertainty Calculation Engineered Control Procedure (ECP) No. 1-RPC-09, Revision 2, dated November 1, 1994. The team's review of this calculation determined that the calculated friction head loss failed to incorporate an entrance loss factor at the point where the RWST water enters the suction pipe. Inclusion of this entrance loss factor would result in an additional RWST level error of about 1.3 feet at the maximum ECCS flow rate of 17,800 gallons per minute (gpm).

The team identified an additional level measurement error, also relating to the velocity of the water in the suction pipe, that had not been accounted for by the licensee. The team found that the licensee did not correct for the velocity head loss correction factor ($v^2/2g$), as stated by the Bernoulli equation, at the location of the pressure taps on the RWST suction pipe. This correction recognizes that the static pressure

measured at the pressure tap location will be reduced due to the velocity head of the flowing water. This correction, with a magnitude of about 2.8 feet at the maximum ECCS flow rate of 17,800 gpm, also results in an indicated RWST water level that is lower than the actual level.

The team's review of ECP No. 1-RPC-09 and CR 93-1212 indicated that negative biases of the type described above were interpreted by the licensee as in a conservative direction; that is, decreasing setpoints, such as the RWST low level alarm setpoint, would be reached prematurely. However, the team found that this conclusion failed to recognize that premature ECCS and CTS pump suction switchover following a LOCA could result in too little water being transferred from the RWST to the RCS and to the containment, which would be a non-conservative impact on the containment sump level. As a result of the team's concern, the licensee initiated CRs 97-2312 and 97-2413 to document this issue. The licensee also made a 10 CFR Part 50.72 event notification on August 22, 1997 (#32806), reporting the potential for premature ECCS switchover as an unanalyzed condition that significantly compromises plant safety and/or a condition not covered by the plant's operating and emergency procedures.

Preliminary investigations performed by the licensee determined that application of all of the velocity-induced errors plus instrument uncertainty could cause the actual RWST water level to be higher than the indicated level by approximately 20% of instrument span (about 6 feet of level) at the time that the low level alarm setpoint is reached. This would cause the operator to initiate switchover of the west RHR and CTS pump suctions before sufficient RWST water has been injected to provide for adequate sump inventory.

The team concluded that failure to adequately consider the uncertainties and bias resulting from flow-induced effects on the ECCS instrumentation demonstrates an apparent lack of adequate design control. The licensee also apparently failed to adequately evaluate the impacts of these uncertainties and bias on design basis safety analyses. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-01. (see Section E1.3.2.2B(1) for further discussion)

(2) Apparent Failure to Account for Instrument Loop Uncertainty in the Technical Specification (TS) RWST Volume Surveillance Requirement

The team found that the licensee had not accounted for the RWST level instrument loop uncertainty in the process setpoint value listed in TS Surveillance Procedure 01(02)-OHP 4030.STP.030, "Daily and Shift Surveillance Checks," Revision 25(23). The procedure indicates an RWST level acceptance criterion of greater than 89% (of span) as indicated on the control room RWST level recorder (MR-36). In Calculation ECP No. 1-G-39, Revision 1, the licensee calculated the normal total loop

uncertainties for the RWST level instrumentation as +3.07%, -3.75% of span. If the recorder is indicating 89% level, the actual water level could be 89% - 3.75% = 85.25% of span. Based on the information contained in Calculation ECP No. 1-2-I9-03, Revision 15, Calculation #3, 85.25% of span corresponds to an RWST water volume of about 349,000 gallons, which is less than the TS-required value of 350,000 gallons.

The team noted that this issue had previously been identified as CR 97-2165 by the licensee as a finding from their UFSAR Re-Validation Project review. The licensee initiated a revision to the procedure to raise the RWST level acceptance criterion to 90% to correct this deficiency. However, the team noted that the corrective action was not comprehensive because the team identified an additional procedure that contained a similar 89% level criterion (01-OHP 4021.017.003, Revision 8, "Removing Residual Heat Removal Loop from Service").

The team concluded that the incorrect RWST level acceptance criterion specified in the above-noted procedure could have allowed the RWST level to be less than the TS requirement. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-02. (See Section E1.3.2.2B(2) for further discussion)

(3) Apparent Failure to Consider the Potential for Vortexing in the RWST and Perform Timely Corrective Action

Following a postulated LOCA, after the operator has switched the suctions of the west RHR and CTS pumps, and the CC and SI pumps to the containment recirculation sump, the east RHR and CTS pumps could continue to draw water from the RWST until the RWST low-low level setpoint is reached (nominally 9.09% of span, elevation 613'-0"). The team questioned whether the potential for vortexing was considered by the licensee, since this low-low level setpoint is about 9 inches above the top of the 24-inch suction pipe. If significant vortexing and air entrainment were to occur, operation of the running RHR and CTS pumps could be adversely affected. The team noted that this issue, a fallout from the staff's 1993 SBICI, had not been resolved at the time of this inspection, even though the licensee had documented this concern in 1995 as a condition report (CR 95-1015). As a result of the team's concern with regard to consideration for vortexing in the RWST, the licensee issued another condition report (CR 97-2350) to address this issue.

The licensee provided the team with a copy of a draft calculation, ENSM 970606JJR, which indicated that incipient vortexing could occur at a RWST level of 12 inches above the RWST low-low level setpoint (elevation 613'-11.7"), and that 2% air entrainment could be experienced at a level of 2.5 inches below the low-low level setpoint (elevation 612'-9.3"). The licensee concluded that the impact of vortexing and air entrainment on pump performance would be insignificant because such operation would be for a short duration (about 2.5 minutes), and would be terminated by

the RWST low-low level RHR pump trip before exceeding the 2% air entrainment capability of the RHR pump (based on Westinghouse document WCAP-11916, "Loss of RHRS Cooling While the RCS is Partially Filled," dated July, 1988). The team's review of this calculation determined that instrument uncertainty for the RWST level instrumentation had not been taken into account by the licensee. Application of the instrument uncertainties (from Calculation ECP No. 1-CG-39, Revision 1, "Refueling Water Storage Tank Level," dated October 24, 1994), which are partially offset by the velocity-induced level measurement errors (described previously), could result in an actual water level that is below the top of the 24-inch ECCS suction pipe when the RWST low-low level pump trip occurs. This would increase the vortexing duration and the potential for air entrainment and pump degradation. The licensee responded to this concern by proposing to raise the RWST low-low level setpoint value from 9.09% to 11% of span.

The team concluded that the licensee failed to adequately consider the potential for vortexing and air entrainment when establishing the RWST low-low level setpoint. In addition, the licensee failed to take prompt corrective action to address this condition after it was identified by the staff's SBICI in 1993, and by the licensee in 1995 (CR 95-1015). 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Action) requires that conditions adverse to quality are promptly identified and corrected. These items are designated as URI 50-315,316/97-201-03 & 04, respectively.

B. EOP 01(02)-OHP 4023.ES-1.3 ECCS Manual Swapover Deficiencies

The EOP describes the steps necessary to transfer RHR and CTS pump suctions from the RWST to the containment recirculation sump following a design basis LOCA, as discussed above in Report Section E1.1.1.2. The licensee enters the EOP when the RWST water level decreases to the low level alarm setpoint (32.23% of instrument span). The initial step in the procedure is to check the containment water level, so that the RHR and CTS pumps have an adequate suction source when switchover is performed (i.e., there is sufficient water in the sump to assure that the pumps have adequate net positive suction head (NPSH) and that vortexing and air entrainment will not occur). Containment water level is monitored by two level sensors, NLI-320 and NLI-321. Water level is also measured in the lower containment sump by two sump level sensors, NLA-310 and NLI-311.

Based on the team's review of the EOP, several deficiencies were identified, and are discussed below.

(1) The Impact of Containment and Containment Sump Level Instrumentation Loop Uncertainties Were Not Properly Evaluated

The uncertainty calculations for the containment and containment sump level instrumentation loops (ECPs 1-2-N3-01, 1-RPC-14, and 2-RPC-14) did

not account for the impact of the loop uncertainties on post-accident containment levels, and did not include considerations for the RHR or CTS pump NPSH requirements, vortexing, or air entrainment. These calculations have the potential to impact the level setpoint criteria contained in the EOP. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-05.

(2) Containment Water Level Value Greater than 15 Percent

The EOP specified a containment water level value of greater than 15%, as indicated on both containment water level instruments NLI-320 and 321 (approximate elevation of 601'6"), as a condition for initiating manual switchover of the west RHR and CTS pump suctions. While this water level appears adequate for NPSH considerations, the licensee indicated that it does not assure the absence of ECCS pump vortex formation and air entrainment. The licensee corrected this procedural deficiency by revising the water level value in the EOP from greater than 15% to greater than 29%, which corresponds to a containment level of elevation 602'10", including instrument uncertainties.

(3) Containment Sump Level Greater than 97 Percent

The EOP contained a provision that if the indicated containment water level was less than the 15% value, pump suction switchover could still proceed if the indicated containment sump level was greater than 97% (approximate elevation of 599'4"). The top of the range of the containment sump level instruments (i.e., 100% of span) is approximately elevation 599'8". However, even at 100% of instrument span, containment water level would be less than elevation 602'10", the value required to assure the absence of vortexing. The licensee corrected this procedural deficiency by committing to delete any reliance on the containment sump level instruments in the next revision to the EOP.

(4) Inappropriate 10 CFR 50.59 Evaluation For Revision 2 to the EOP

Prior to June 1992, the EOP and the description of the manual swapover sequence in the UFSAR were in agreement. UFSAR Section 6.2.2 stated that the operator first transfers one ECCS train to recirculation (when the RWST low level alarm setpoint is reached), and then transfers the other ECCS train. An ECCS train consists of one CC pump, one SI pump, and one RHR pump.

In June 1992, the licensee revised the EOP (Revision 2) to piggy-back both CC and SI pumps on the west RHR pump, prior to re-separating the CC and SI trains after the manual swapover had been completed. Although the licensee was aware that Revision 2 of the EOP was inconsistent with the assumptions in the UFSAR, the UFSAR was not changed to agree with

the EOP until July 1997. The licensee documented the deficiency between the UFSAR and the EOP as CR 97-0067 in January 1997.

The team reviewed the EOP's procedural step sequences and became concerned with the potential for a single failure vulnerability to render both SI and CC pumps inoperable if the West RHR pump was to fail during the period of time both CC and SI pumps were piggy-backed on the West RHR pump. The team concluded that the 10 CFR Part 50.59 safety evaluations that were performed by the licensee for Revision 2, and later for Revisions 3, and 4 to the EOP (June 1992, January 1996, and January 1997, respectively) were not effective in identifying the potential single failure vulnerability that could have rendered both trains of SI and CC pumps inoperable. Operation of the plant with the potential for both trains of SI and CC pumps to be rendered inoperable, given a single failure could have increased the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. 10 CFR 50.59(a)(2) requires that proposed changes shall be deemed to involve an unreviewed safety question (USQ) if the proposed change increases the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. (See Section E1.5.2.B(3) for further discussion).

(5) Inappropriate Temporary Procedural Change to the EOP

On August 22, 1997, the licensee made a temporary change to Revision 4 of the EOP to the prerequisite steps that verify an adequate water level in the containment prior to switchover to the recirculation mode. The specific change was to raise the containment water level setpoint value needed prior to manual initiation of switchover to the recirculation mode from greater than 15 percent to greater than 29 percent. The change was characterized as a "non-intent" change to the procedure, and as allowed by TS 6.5.3.1, a full 10 CFR 50.59 screening was not completed prior to implementation. The licensee concluded that the change was conservative in nature and did not constitute a change to a procedure as described in the FSAR. The team determined that this conclusion was incorrect because the basic procedural steps are described in UFSAR Section 6.2.1, "Change-Over from Injection Phase to Recirculation Phase." In addition, UFSAR Section 6.2.1, Page 6.2-12, also contains the statement, "The detailed sequence for the changeover from injection to recirculation is given in an emergency operating procedure." The team concluded the licensee made inappropriate changes to the EOP without proper review and approval, as required by 10 CFR 50.59 and TS 6.5.3.1. (See Section E1.5.2A(1) for further discussion.)

The procedural deficiencies concerning the EOP could have allowed the operator to switchover the RHR and CTS pump suctions to the containment recirculation sump before the recirculation sump contained sufficient water volume to preclude vortexing and potential air binding of the ECCS pumps. In addition, the license did not recognize the creation of a single failure vulnerability that could render both SI and CC pumps inoperable. The licensee wrote a condition report (CR 97-2312) documenting this issue and made a 10 CFR Part

50.72 event notification relating to these concerns on August 22, 1997 (#32806), reporting the event as an unanalyzed condition that significantly compromises plant safety and/or a condition not covered by the plant's operating and emergency procedures.

C. Impact of the RWST and Containment Level Setpoint Errors On Post-LOCA Analyses

As discussed in items E.1.1.1.2 A and B above, successful operator switchover to the recirculation phase of ECCS operation following a LOCA is dependent on the containment recirculation sump receiving an adequate amount of water prior to the manual switchover from the RWST. The licensing basis for the plant was a containment water level at an elevation of 602'10", as documented in the Alden sump model test report and in the licensee's response to NRC Unit 2 FSAR Question 212.29 (contained in the Unit 2 FSAR Appendix Q, Amendment 78, October 1977). The licensee's response to Question 212.29, stated that this level was based on the transfer of the water inventory contained in the RWST with switchover occurring between the minimum level and the low level alarm setpoints, and that it neglected any contribution from the RCS or ice condenser ice melt.

The team requested a copy of the calculation that established the elevation at the 602'10" value, in order to review the assumptions and verify that any volumes of water that may not drain back to the containment sump were taken into account. However, the licensee could not produce this calculation, and initiated CR 97-2409 to address the inability to retrieve the calculation. In addition, further licensee evaluation indicated that under certain LOCA scenarios, the volume of water that is in the "active sump" (containment recirculation sump outside of the reactor annulus space) may not be adequate to support long term ECCS or CTS pump operation during the recirculation phase of a LOCA. This conclusion was reached because the licensee could not confirm that sufficient communication (i.e., drainage paths) existed between the active and inactive sumps within the containment, and that water was being removed, over time, from the active containment sump to the inactive sump via the containment spray system. The active containment sump level would also be potentially impacted by the reduced transfer of RWST volume, as previously noted in E1.1.1.2 A above.

Based on these concerns, the licensee declared both trains of ECCS and CTS inoperable in both units, made a notification of an unusual event (NOUE), entered TS 3.0.3, and commenced an orderly dual unit shutdown on September 8, 1997. The licensee made a 10 CFR Part 50.72 event notification relating to these concerns (#32890), as a condition outside the design basis and as a TS-required shutdown.

In addition, the team was concerned that two other safety analyses could potentially be impacted due to this issue, the containment analysis and the LOCA analysis. The containment analysis may be impacted due to the premature switchover from the RWST to the containment recirculation sump, resulting in the potential for the CTS system to be less effective in reducing containment pressure when spraying recirculated 190°F containment sump water rather than 100°F RWST water. The licensee stated that a preliminary Westinghouse

analysis indicated that the containment design pressure would not be exceeded. However, the analysis was not formalized or documented prior to the end of the inspection.

The LOCA analysis may be impacted because the volume of borated RWST water injected into the RCS may not be sufficient to maintain the core subcritical following the accident. The licensee stated that a preliminary Westinghouse analysis indicated that a return to criticality would not occur. However, the analysis was not formalized or documented prior to the end of the inspection.

The team concluded that the licensee could not demonstrate to the team, using design or licensing basis documentation, and considering the procedures and use of instrument uncertainties in place at the time, that there was adequate containment recirculation sump water volume following a LOCA. 10 CFR Part 50.46, requires that the ECCS be designed so that its calculated cooling performance meets the criteria for the long-term cooling function, as set forth in 10 CFR Part 50.46(a)(1)(ii). This item is designated as URI 50-315,316/97-201-06.

D. Single Active Failure Criterion

The team considered the licensee's interpretation of "single active failure," as stated in UFSAR Section 6.2.3, to be inconsistent with the staff's interpretation of the UFSAR definition. UFSAR Section 6.2.3 defines "active failure" as "the inability of any single dynamic component or instrument to perform its design function when called upon to do so by the proper actuation signal. Such functions include change of position of a valve or electrical breaker, operation of a pump, fan or diesel generator, action of a relay contact, etc." The licensee's interpretation was that postulated failure-to-run scenarios were not considered valid "active failures," as defined in UFSAR Section 6.2.3. Discussions with the licensee confirmed that their interpretation of their design and licensing basis was that single active failures would include failure of a pump to start on demand, but not a failure to run after completion of a successful start demand.

This issue became a concern during the team's review of the RHR system and the ECCS for single failure vulnerabilities. From the single failure analysis presented in UFSAR Table 6.2-6, it was apparent to the team that the only active failures considered were failures-to-start or initiate an action (e.g., failure of a pump to start, or failure of a valve to open/close). Failures-to-run were not considered valid active failures. The licensee contacted Westinghouse for confirmation of their position, but was informed by Westinghouse that they generally considered failure-to-run scenarios in their accident analyses, and that, generally, postulated failure-to-run cases were bounded by failure-to-start cases.

However, the team identified a failure-to-run case that was not bounded. This case arises from the instructions contained in the EOP, which were revised in 1992, for performing the RHR and CTS pump suction switchover. After the west RHR and CTS pump suctions have been switched to the containment recirculation sump, both trains of CC and SI pumps are aligned to take suction from the west RHR loop, and the suction lines from the RWST to the CC and SI pumps are then

isolated. In this configuration, prior to switchover of the east RHR pump suction, failure of the west RHR pump to continue running could result in the loss of both CC pumps and both SI pumps, as well as the west RHR pump. Injection flow to the RCS would be available only from the east RHR pump. This single failure could challenge the ability of the ECCS to mitigate the consequences of the LOCA, particularly small-break LOCAs, in which high and/or intermediate head injection capability is necessary for RCS inventory control. The licensee made a 10 CFR Part 50.72 event notification relating to this concern on September 10, 1997 (#32904), as an unanalyzed condition and outside the design basis, and also initiated CR 97-2449.

The team concluded that failure to preclude a single active failure when performing changes to the plant is contrary to the assumptions in the UFSAR and the design basis. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-07.

E. Containment Recirculation Sump Particle Retention Requirement

The containment recirculation sump contains fine mesh screens to prevent accident-generated debris larger than 1/4-inch from clogging and/or restricting flow through the ECCS and CTS systems. The 1/4-inch particulate exclusion criterion was originally based on the limiting size of the containment spray nozzle openings. An operating assessment review performed by the licensee, and as documented in CR 96-0076, determined that the ECCS throttle valves also have small openings that could be subject to debris clogging and flow blockage. The clearance through these throttle valves was determined to be 0.4125 inches. The team reviewed design requirements, design documents, and licensing commitments relating to the recirculation sump, and identified several concerns that are described below.

(1) The Sealing of 3/4-inch Sump Roof Vent Holes Without Performing a 10 CFR 50.59 Evaluation

In 1978, the licensee documented in a letter to the NRC proposed modifications to the containment recirculation sump (AEP letter no. AEP:NRC:00110, December 29, 1978). The sump modifications were enhancements resulting from recommendations contained in a 1978 Alden Research Laboratory Report. In 1979 (Unit 2) and 1980 (Unit 1), the licensee made modifications to the recirculation sumps to improve hydraulic performance and minimize the potential for vortex formation and air entrainment. As part of the modification package, 3/4-inch holes were drilled in the sump roof to vent air that could potentially be trapped beneath this roof. Subsequently, in 1996 (Unit 2) and 1997 (Unit 1), after questioning by the NRC resident inspectors as to what was the function of the holes, the licensee sealed the holes because they were not described in the UFSAR or indicated on any plant design drawings, consideration of the vent holes as a design requirement could not be verified, and the 3/4-inch holes were contrary to the 1/4-inch containment recirculation sump particulate requirement.

Although the Alden test results indicated that the sump would perform adequately without the vent holes, the licensee sealed the holes without performing a 10 CFR Part 50.59 safety evaluation on the modification to the plant. The licensee made an event notification on September 10, 1997 (#32903), regarding the elimination of the vent holes as contrary to a commitment made to the NRC in 1978 regarding enhancing sump performance through use of 1/4-inch vent holes, and also initiated CR 97-2344, which documented these issues and the failure to maintain the venting commitment. This event was later retracted on October 20, 1997, when the licensee determined that the sump would have performed its design basis function and that the vent holes were only an enhancement (although it was an NRC commitment and part of the licensing basis). Failure of the licensee to perform a 10 CFR Part 50.59 safety evaluation demonstrated a lack of licensee awareness or understanding of the plant design and licensing basis. (See Section E1.5.2B(4) for further discussion).

(2) Sump Screen Edge Gaps Exceeding 1/4-Inch Containment Recirculation Sump Particulate Retention Requirement

The licensee requested in RFC DC-12-2361 the removal of a horizontal perforated plate in the containment recirculation sump, and installation of a fine particulate screen behind the vertical grating at the sump entrance. However, installation of the fine particulate screen was deficient in that gaps at the edges of individual screen sections allowed passage of debris particles up to 1/2-inch. Together with the 3/4-inch sump roof vent holes, debris particles exceeding the 1/4-inch particulate retention requirement could have entered the ECCS and CTS systems. The licensee documented this condition as CR 96-0402 and CR 97-0668, and repaired the subject edge gaps, along with sealing the vent holes, as noted above. The licensee made a 10 CFR Part 50.72 event notification (#32875) to report that from 1979 until the sump screen edge gaps were repaired in 1997, the sump did not meet its design requirement, and also documented this condition in CR 97-2407.

However, the team's review of this event and the licensee's event report to the NRC, identified that the potential for plugging of the ECCS throttle valves was not previously considered in the reportability basis and not mentioned in the event notification to the NRC. The team believed that this was a potentially more limiting area of concern, which could render redundant trains of ECCS inoperable via a common-mode failure scenario if debris larger than .4125 of an inch were allowed to enter the ECCS. Common-mode failure of redundant ECCS injection pathways during the recirculation phase of a LOCA potentially could impact the adequacy of the ECCS to perform its long-term cooling function. 10 CFR Part 50.46, requires that the ECCS be designed so that its calculated cooling performance meets the criteria for the long-term cooling function, as set forth in 10 CFR Part 50.46(a)(1)(ii). This item is designated as URI 50-315,316/97-201-08.

F. Potential Clogging of Containment Recirculation Sump Screens

During this inspection, a regional inspector identified that fibrous material had been installed in electrical cable trays inside containment for both units. The licensee performed an investigation and found that this material, called "Fiberfrax," could potentially decompose and break into small pieces during a LOCA, and be transported to the containment sump, potentially clogging a significant portion of the recirculation sump screens. The potential existed for the screens to be blocked in excess of the design value of 50 percent of the area of the screens, creating the potential for the ECCS system not to perform its design function during the recirculation phase of a LOCA. This is a regional item and will be followed by subsequent regional inspection followup activities.

G. Recirculation Phase ECCS Leakage to the RWST

Following a postulated LOCA, with the ECCS operating in the recirculation mode, the ECCS piping located outside of the containment could contain significant amounts of radionuclides in the fluid. Leakage of these fluids past the valves that isolate the ECCS recirculation flow paths from the RWST could result in an unfiltered and unmonitored release to the environment and contribute to offsite and control room doses, since the RWST is vented to the atmosphere. This issue was identified to the industry by the NRC in NRC Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere." The licensee's assessment of this issue was documented in Analysis No. 960308.001, Revision 0. The team reviewed this analysis and noted that a total of six valves were identified as potential leakage paths, and that four of the six valves were not in a leak testing program. These valves and their associated functions are listed below.

(1) IMO-910 and IMO-911, which isolate the 8-inch CC pump suction line from the RWST. These are parallel valves.

(2) IMO-261, which isolates the 8-inch SI pump suction line from the RWST.

(3) IMO-262 and IMO-263, which isolate the 2-inch SI pump minimum flow line that returns to the RWST. These valves are in series.

(4) RH-130, which isolates the 8-inch RHR return line to the RWST.

The analysis stated that the total leakage through the paths identified above must be less than 9 gpm. This provides margin to the post-LOCA dose calculation assumption of 10 gpm of ECCS leakage to the RWST (documented in Unit 1 UFSAR Section 14.3.5). Allowable leakage rates were calculated as 1.356 gpm for each of the 2-inch SI minimum flow isolation valves (two in series), and 1.91 gpm for each of the other valves.

The team reviewed the inservice testing (IST) requirements for the above identified valves and determined that only IMO-262 and IMO-263 are leak-rate tested as Category A valves in accordance with the ASME Code, Section XI,

Subsection IWV-2200, Categories of Valves. The other four valves are designated as Category B valves and are not leak-rate tested. Consequently, the licensing basis control room and offsite dose calculation assumption of 10 gpm of ECCS leakage to the RWST is not being verified by the licensee. If the actual total valve leakage were greater than 10 gpm, the control room and offsite dose consequences of a LOCA could be increased. The licensee initiated CR 97-2450 to document this concern. 10 CFR Part 50, Appendix B, Criterion XI (Test Control), states that all testing required to demonstrate that components will perform satisfactory in service is identified and performed in accordance with written procedures which incorporates the requirements and acceptance limits contained in applicable design documents. This item is designated as URI 50-315,316/97-201-09.

E1.1.1.3 Conclusions

The team concluded that although the mechanical design of the RHR and supporting ECCS functions appeared generally acceptable, the ECCS system may not have performed its intended safety function under all design basis accident scenarios due to significant weaknesses in design and procedural control. The issues raised by the team concerning flow-induced bias and instrument uncertainties associated with the ECCS level instrumentation, the deficiencies with the EOP that created the potential for premature switchover from the RWST to the containment sump following a LOCA, the failure to demonstrate that there would be adequate water in containment following a LOCA, the identification of a single failure and common-mode failure vulnerabilities potentially rendering both trains of SI and CC pumps inoperable, as well as the potential for clogging of the containment recirculation sump screens, ECCS throttle injection valves, and containment spray nozzles during a LOCA, are significant deficiencies that could seriously impact the ability of the ECCS to function, as designed.

As a result of some of these concerns, the licensee elected to declare both trains of the ECCS and CTS system inoperable and commence an orderly shutdown of both units during the inspection.

E1.2 Component Cooling Water (CCW)

E1.2.1 Mechanical

E1.2.1.1 Scope of Review

The mechanical design review of the Component Cooling Water (CCW) system included design and licensing documentation reviews, a system walkdown, and discussions with system and design engineers. The team reviewed applicable portions of the UFSAR and TS, the CCW and ESW system interface, flow diagrams, physical drawings, vendor drawings, equipment specifications, DBDs, numerous calculations, operating and surveillance procedures, and CRs. The scope of the review included the appropriateness of the design, bounding design conditions, validity of design assumptions, verification of design input, vulnerability of system components, overpressure protection, and adequacy of tests and surveillances.

The team placed special emphasis on the effects on plant operations when the ultimate heat sink (UHS) temperatures were above the maximum design basis UHS temperature of 76°F; on the cooldown analysis performed by Westinghouse for the licensee; on thermal performance of the CCW heat exchangers; on the uncertainties inherent in recording, measuring and testing equipment; and on the dependence of safety-related functions on non-safety-related equipment.

E1.2.1.2 Findings

A. Operation of the Plant Above the Essential Service Water/Ultimate Heat Sink Maximum Operating Temperature

Since 1988, the licensee has operated the plant from several days to several weeks on a yearly basis above the UFSAR design basis UHS temperature limit of 76°F. In the summer of 1988, for a period of 22 continuous days, the plant exceeded the 76°F UFSAR maximum essential service water (ESW) intake (or the UHS) temperature limit, with lake temperatures averaging 81°F, occasionally with the 24-hour averaged lake temperatures recorded at 84-85°F, not taking into account a 3.5°F instrument uncertainty. Impact on plant operation was not fully evaluated by the licensee, no 10 CFR Part 50.59 evaluation was performed, and the UFSAR had not been updated to reflect plant operation above the 76°F design basis value. The team determined that operation of the plant above the maximum UHS temperature limit potentially constituted an unreviewed safety question (USQ) with regard to reduction of safety margin as defined in the TS Bases for the control room emergency ventilation temperature limit of 95°F (see Section E1.2.1.2 E, Control Room Emergency Ventilation Surveillance and Equipment Qualification), and adversely impacted the plant cooldown analysis.

FSAR Table 9.5-3 states that the service water inlet temperature to the CCW heat exchanger is 76°F. This value is the design basis for the heat removal capability of the ESW system. This value was also used in numerous calculations and analyses assessing the heat removal capability of ESW as well as CCW, whose heat gain is ultimately being removed by ESW.

The team reviewed numerous licensee engineering memorandums issued between 1988 and 1994 that attempted to justify plant operation above the 76°F UHS temperature limit. The memorandums were focused on specific topics, such as identifying the limiting component of concern (i.e., EDG jacket water coolers, CCW-side of the RCP thermal barriers), the impact on the control room air conditioning system, or determining if the calculated containment peak pressure identified in the safety analysis would be adversely impacted. However, none of these memorandums accounted for the overall impact on the plant from operating at the elevated lake temperatures, such as the impact on the control room equipment qualification or the design basis requirement to be able to shutdown the plant in 36 hours at 200°F with one train of CCW operating at 95°F. Although these memorandums were safety reviews, no apparent 10 CFR 50.59 evaluation had been performed to determined if operation above 76°F constituted an USQ, or if there would be a reduction in the safety margin, or to account for the overall impact to the plant from operating above the design basis UHS temperature limit of 76°F. The team noted also that use

of unsubstantiated engineering memorandums was a previous Region III NRC inspection issue and notice of violation in 1994 (NRC Inspection Report 50-314/315 94007).

In addition, team interviews with the licensee revealed that the licensee's current operating plant procedures would allow plant operation with Lake Michigan temperatures above the ESW design basis temperature limit of 76°F, without requiring any immediate action by the licensee. After the team identified this discrepancy to the licensee, the procedure was inappropriately revised to limit the temperature to 85°F. Although the licensee indicated to the team that the procedural revision from 87.5°F to 85°F was in the conservative direction, the team noted that this change still allowed the licensee to operate the plant outside of the design basis limit of 76°F, thus, the change to the procedure was performed without an appropriate 10 CFR Part 50.59 evaluation. The licensee again revised the procedure to re-instate the original design basis maximum allowable ESW temperature of 76°F. (see Section E1.5.2A(2) for further discussion)

Further, plant operation at and above the maximum UHS temperature limit, which was controlled by instrument setpoint values specified in ESW Operating Procedure 12-OHP 4021, was based on actual instrument readings without accounting for a 3.5°F instrument uncertainty. As a result of the team's questioning, the licensee calculated that when the ESW intake temperature monitor instrument indicates 87.5°F, the actual temperature may be 91°F when accounting for instrument uncertainty (see Section E1.3.2.2B(6) for further discussion). The licensee had not evaluated plant operation or the impact on the safety analysis for peak containment pressure or plant cooldown at the 91°F value. After the team pointed out the potential discrepancy to the licensee, ESW Operating Procedure 12-OHP 4021 was again revised to require that when the ESW temperature monitor indicates 72°F, accurate resistance temperature detectors (RTDs) are to be used for a more precise determination of ESW temperature.

As a result of the team's concerns regarding past plant operation above the maximum design basis UHS temperature limit, the licensee initiated three one-hour non-emergency notifications in accordance with 10 CFR Part 50.72, (b)(1)(ii)(A), (B) unanalyzed condition and outside the design basis. 10 CFR 50.72 # 32740 on August 8, 1997 (Units 1 and 2 operated outside design basis of 76°F lake temperature); 10 CFR 50.72 # 32822 on August 26, 1997 (procedure allowed CCW operation above the UFSAR maximum temperature of 95°F); and 10 CFR 50.72 # 32843 on August 29, 1997 (Units 1 and 2 operated in an unanalyzed condition when Lake Michigan temperatures were above 76°F for 22 consecutive days during August 1988).

The team concluded that since 1988, the licensee has seasonally, and on occasion, operated the plant outside the design basis when ESW temperatures are above 76°F. These changes were made without performing a 10 CFR Part 50.59 safety evaluation to evaluate the overall impact on the plant at the higher lake temperatures. (see Section E1.5.2B(1) for further discussion)

B. TS 3.0.3 Plant Cooldown Capability and Cooldown Analyses

The licensee was not able to demonstrate to the team that they could achieve the TS-required 3.0.3 cooldown of 200°F RCS temperature in 36 hours using design basis analyses and the assumption of 1 train of CCW operating at a maximum temperature of 95°F, with one train of RHR, at a lake temperature of 76°F.

Technical Specification 3.0.3 states that when a limiting condition for operation is not met, the plant is to be brought to a cold shutdown condition within 36 hours. TS Table 1.1 defines Cold Shutdown as an average RCS temperature of equal to or less than 200°F. In addition, an NRC staff Safety Evaluation, dated September 10, 1973, states that one CCW pump and one CCW heat exchanger serve the needs of a Unit during either full power operation or cooldown. The team concluded that the TS 3.0.3 cooldown requirement and the assumption in the staff's Safety Evaluation was to demonstrate that at 100 percent power, the plant would be capable of achieving 200°F within 36 hours, using one train of CCW.

The team requested licensee calculations and documentation that demonstrate the ability of the plant to meet the requirement of TS 3.0.3 and the assumption stated in the staff's 1973 safety evaluation. The documentation provided to the team by the licensee was Calculation NEMP 960519AF, CCW LOCA/Cooldown Analysis for the Unit 2 Uprating Program, dated June 25, 1996, and the Westinghouse Cooldown Analysis Report, submitted as Letter AEP-96-104, dated July 19, 1996. However, in these calculations, the licensee assumed a maximum ESW temperature above the design basis maximum temperature of 76°F (at 87.5°F), and operation of the CCW system above the UFSAR maximum temperature limit of 95°F (up to 120°F).

In addition, for both of the above analyses, and in other design basis documentation, such as the UFSAR Chapter 14 accident analysis, the licensee identified, as a result of the team's questioning, that they had incorrectly modeled the CCW heat exchanger as a pure counter flow heat exchanger, as opposed to the more accurate model of a single pass (CCW), two pass (ESW) design. This error resulted in a non-conservative increase in heat exchanger capacity of about 14%. The team was further concerned with this error because the licensee had submitted to the NRC two major analyses which also contained this error. These analyses were the Unit 2 Steam Generator 30% Tube Plugging Program, that was submitted on August 26, 1995 and approved by the staff on March 13, 1997, and the Unit 2 Uprate Program, that was submitted in 1996 and was under current review by the staff. As a result of the team's concern, the team asked the licensee on August 20, 1997, as to what impact this error would have on any previous analysis performed by the licensee or submitted to the NRC. Subsequently, due to competing resource limitations from issues raised by the team, the licensee issued letter AEP:NRC:1223M, Donald C. Cook Nuclear Plant Unit 2, Request to Suspend NRC Review of Unit 2 Uprate, dated September 10, 1997, requesting the staff to temporarily suspend NRC review of the Unit 2 Uprate Program.

Although the licensee indicated to the team that sufficient margin should exist to compensate for the CCW heat exchanger modeling error without

impacting the UFSAR Chapter 14 conclusions, the team did not have the opportunity to pursue a detailed review of the analysis contained in UFSAR Chapter 14.

The team concluded that the licensee was unable to demonstrate by existing analysis that they could achieve the TS-required 3.0.3 cooldown of 200°F RCS temperature in 36 hours using design basis assumptions of 1 train of CCW operating at a maximum temperature of 95°F, with one train of RHR, at a lake temperature of 76°F.

10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. The team concluded that this requirement was potentially not met because the licensee could not demonstrate using existing design basis assumptions that the plant could achieve the TS-required cooldown of 200°F in 36 hours. In addition, the licensee also failed to correctly translate the as-built design of the CCW heat exchanger into safety-related calculations and analyses. These items are designated as URI 50-315/316/97-201-10 and URI 50-315/316/97-201-11, respectively.

C. Spent Fuel Pool Cooling Capacity During the 1996 Unit 2 Outage

In 1996, the licensee performed a full core offload during the Unit 2 refueling outage. To support this activity, the licensee performed analyses and a 10 CFR 50.59 evaluation (Safety Review Memorandum, Unit 2 Refueling Outage Proposed Full Core Offload and Addendum 1, dated March 11, 1996 and March 20, 1996, respectively) in order to demonstrate the adequacy of the CCW cooling capacity of the spent fuel pool with the current SFP heat load, and to show that a USQ would not be created as a result. However, the licensee's analyses indicated that at a CCW system temperature of 90°F, there was essentially no margin available to maintain the SFP bulk temperature below the design basis maximum temperature of 160°F, using one train of CCW at the maximum design basis CCW operating temperature of 95°F. This design basis requirement is stated in UFSAR Section 9.4.3, "Spent Fuel Pool Cooling System."

In an attempt to comply with the UFSAR requirement, the licensee elected to implement administrative controls by performing a temporary non-intent change to CCW System Operating Procedure 01/02-OHP 4021.016.003, "Operation of the CCW System During Reactor Startup and Normal Operation," to prevent operation of the CCW system at 120°F, and to reflect the revised administrative limit of 80°F on CCW system temperature to require the CCW system temperature to remain below 80°F. This procedure change was performed without a 10 CFR Part 50.59 review because the procedural change was characterized by the licensee as a non-intent change in the conservative direction, therefore the provisions and guidance of TS 6.5.3.1, "Technical Review and Control," would apply. The team questioned whether this was an appropriate TS characterization and interpretation of the requirements as stated in 10 CFR Part 50.59 (see Section E1.5.2A(3) for additional discussion).

The team was concerned that the licensee's 10 CFR Part 50.59 evaluation failed to properly recognize the safety consequence that the CCW system could not perform its design function to remove the required SFP heat load at its maximum operating system temperature of 95°F, as stated in the UFSAR. After questioning the licensee whether this condition constituted operation outside of the design basis or a potential USQ, the licensee made an event notification on August 26, 1997 (#32824) regarding failure to consider CCW maximum operating temperature design assumptions, as stated in the UFSAR.

In addition, the team's review of Event Notification 32824 revealed that the licensee reported that the 10 CFR 50.59 evaluation had addressed UFSAR Section 9.4.3 on the SFP bulk pool temperature requirements, and another UFSAR assumption dealing with the time-to-boil events. UFSAR Section 9.4.3, "Spent Fuel Pool Cooling System," indicates that an acceptable scenario would be a SFP heat loading scenario that meets the 160°F peak bulk pool temperature and a 5.74 hours time-to-boil criteria (in the event that both CCW loops of the SFP cooling system become inoperable).

However, the team questioned whether Event Notification #32824 was an accurate account of the reportable event because the team could not identify where time-to-boil criteria was considered in the 10 CFR Part 50.59 evaluation. The team was further concerned whether the licensee could have met this criteria, given that the safety evaluation concluded that the 160°F SFP bulk temperature limit could not be maintained using design basis assumptions with the CCW system operating at 95°F. After informing the licensee of this concern, the licensee subsequently made a revision to the event notification to state that time-to-boil events were not addressed by the safety review (see Section E1.5.2B(2) for further discussion).

Further review by the team of UFSAR Section 9.4 indicated that the operational conditions and limitations that were present during the 1996 Unit 2 full-core offload refueling outage were not explicitly addressed in the UFSAR, where one unit is fully offloaded with less than the maximum anticipated fuel assembly loading, plus one complete core. The UFSAR states that under the maximum anticipated SFP heat loading condition, the bulk SFP temperature is analyzed to remain below 180°F, assuming 3420 spent fuel assemblies plus one complete core in the pool, and with only one CCW cooling train available. However, the UFSAR also states that this scenario is not part of the design basis and results in unacceptable bulk pool temperature. The team could not explain the basis for this statement in the UFSAR other than under certain loading conditions, a full core offload with the maximum anticipated heat load would be unacceptable.

The team concluded that with operation of a single CCW cooling train under these conditions, as described by the licensee's 1996 Unit 2 SFP analysis and in the 10 CFR Part 50.59 evaluation, that this would constitute operation outside of the design basis. In addition, the team concluded that there was also the potential for a reduction in the time to boil the SFP water, with plant operation at the design basis CCW temperature limit of 95°F, creating the potential for a USQ. The deficiencies with the 10 CFR 50.59 evaluation are further discussed below and in Section E1.5.2B(2).

D. Dual CCW/ESW Train Outage During Refueling Inconsistent with Design Basis

The licensee identified during their recent SFP UFSAR Re-Validation effort, that during the 1996 Unit 2 refueling outage, the coincident dual train CCW outage was inconsistent with UFSAR Table 9.5-2, Note 3, regarding CCW train limitations for SFP cooling during LOCA injection and recirculation scenarios. UFSAR Table 9.5-2 identifies design, normal, post-accident and cooldown flow demands on the CCW system. The CCW flow rate is not identified for either the post accident or cooldown modes of operation. For post-accident conditions, Note 3 of the table states that the SFP heat exchanger is assumed to be on the non-accident unit. Further, the removal of two CCW trains from service during a refueling outage on one unit, with the other unit at power, would create a single failure vulnerability during a design basis accident scenario on the operating unit due to the fact that CCW cooling to the SFP heat exchanger from the operating unit auto-isolates on a containment Phase A signal. An accident and initiation of the Phase A signal on the operating unit would constitute a complete loss of SFP cooling as an anticipated consequence of the automatic design features of the plant during a LOCA scenario and without having to consider a single active failure.

As a result of the team's questioning whether a dual CCW train outage constitutes operation outside of the design basis, the licensee initiated a one hour, non emergency event notification of this in accordance with 10 CFR Part 50.72(b)(1)(ii)(B) Outside Design Basis, Report Number 32823 on August 26, 1997. This condition was also documented by the licensee as CR 97-2341. (See Section E1.5.2A(3) for additional discussion)

E. Control Room Emergency Ventilation and Equipment Qualification

The team performed a special review of the control room emergency ventilation system (CREVS) and evaluated its ability to maintain the control room at temperatures that the control room equipment was qualified for. This review was undertaken because the team was concerned that the design function of the CREVS could potentially be adversely impacted when ESW (UHS) intake temperature is above the maximum design basis temperature of 76°F and because the licensee did not perform periodic CREVS performance testing to ensure the heat removal capability using only safety-related equipment.

In discussions with the licensee, the licensee stated that, based on their calculations, and assuming no credit for the nonsafety-related chillers, if ESW (UHS) intake temperatures approach 66°F or higher, the TS control room ambient temperature requirement of 95°F would be exceeded. This was of concern to the team because 66°F was 10 degrees below the design basis UHS temperature limit of 76°F, and the period of time that the plant could potentially operate with lake temperatures at or above this value, was significant. As a result, the team concluded that the associated CREVS TS Surveillance operability requirement 4.7.5.1, requiring verification that the control room temperature is less than or equal to 95°F, does not appear to adequately demonstrate operability of the CREVS system, since the TS requirement is met using the normal mode of control room ventilation.

After 1988, when the licensee realized that lake temperatures could exceed 85°F and be sustained above 76°F, the licensee performed an analysis to determine the estimated worst case control room temperatures using lake temperatures of 87.5°F. The design basis temperature for which the control room equipment was originally qualified, using an UHS temperature of 76°F, without credit for the nonsafety-related chillers, was estimated to be 104°F (licensee memorandum, "Design ESW Temperature," dated September 4, 1992). Licensee Calculation DCCHV12CR11N, Revision 0, "Control Room Temperature Evaluation," dated June 22, 1990, estimated that the qualified life of the most limiting equipment is approximately 800 hours at 104°F (i.e., the solid state protection system, SSPS and the nuclear instrumentation system, NI). The licensee subsequently found that control room temperatures of up to 118°F could be reached at lake temperatures of 87.5°F (Memorandum "Operation at High Lake Temperature," dated August 3, 1988, and Memorandum, "Design ESW Temperature," dated September 4, 1992). The team was told by the licensee that at control room temperatures of approximately 120°F, the qualified life of the SSPS and NI systems was 12 hours, although this value was not incorporated into the design or licensing basis. The team found that the licensee had performed calculations to determine worst case control room temperature conditions, but no attempt had been made to estimate the plant's condition and ability to mitigate design basis accidents during the 22 days in 1988, where the plant operated with an average UHS temperature of 81°F.

In 1990, the licensee was granted a change to the CREVS TS Surveillance requirement 4.7.5.1.a, to reduce the maximum control room temperature requirement from 120°F to 95°F (AEP Letter to NRC, NRC:0398U Technical Specification Change Limit, dated September 10, 1990). The licensee told the team that the reason for the TS change was that at control room temperatures near 120°F, qualification of the control room equipment could not be assured. The licensee stated in their 1990 TS amendment submittal that at a control room temperature of 80°F, the qualified life of the control room equipment would be in excess of the 40-year life of the plant, while the qualified life of the limiting equipment at 95°F is estimated to be 15,000 hours. Because temperature excursions above 80°F were expected to be limited in number and very short in duration, the licensee and the NRC staff, in its safety evaluation, presumed that operation at the TS limit is acceptable because the portion of the time the control room would be at elevated temperatures is small in comparison to the qualified life of the equipment. However, the team determined that this assumption was non-conservative and does not reflect the potential for the plant to operate with elevated lake temperatures above 76°F. In addition, the team determined that the original TS allowing plant operation with a control room temperature of up to 120°F (with the SSPS and NI systems only qualified for 12 hours) was incompatible with the design basis temperature limit of 104°F.

The team had the following concerns regarding the adequacy of the CREVS to meet the design basis with respect to past plant operation.

(1) Potential USQ with Regard to Control Room Equipment Qualification

The licensee's design and licensing basis for qualification of the control room equipment is 15,000 hours at 95°F, or 800 hours at 104°F.

TS Surveillance Requirement 4.7.5.1a, requires that the CREVS shall be demonstrated to be operable by verification that the control room temperature is less than or equal to 95°F. TS Bases 3/4.7.5, "Control Room Emergency Ventilation System," state that the operability of the CREVS ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions and at temperatures at which control room equipment is qualified for the life of the plant.

However, the team found that the CREVS is not capable of performing this function without the aid of the nonsafety-related chillers. The licensee calculated that without credit for the nonsafety-related chillers, if the UHS temperature averages 81°F, control room temperatures would subsequently reach 110.4°F. At this temperature, the control room equipment qualification would be reduced from the design and licensing basis of 15,000 hours to 6.25 days.

In addition, in August, 1988, for a period of 22 consecutive days, the plant operated with an UHS temperature averaging 81°F, although the nonsafety-related chillers remained operable. However, the licensee did not have approved procedures in place to alert personnel as to a course of action if the chillers were lost, or to shut down the plant if control room temperatures rose above 95°F. This event also constituted operation outside of the design basis because the safety-related CREVS would not be able to maintain the control room equipment qualified beyond 6.25 days with UHS temperatures averaging 81°F.

The team concluded that the 1988 event potentially constituted a USQ with regard to a reduction in the safety margin, as defined in the plant's TS Bases, and with regard to the creation of an accident or malfunction different than those previously evaluated in the safety analysis report. TS Bases 3/4.7.5, "Control Room Emergency Ventilation System," states that "continued operation at the TS limit (of 95°F) is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit." The team determined this assumption was no longer conservative given that the equipment could actually exceed its qualified life under design basis assumptions (see Section E1.5.2B(1) for further discussion).

As a result of questions posed by the team, the licensee subsequently determined that for the 22 days in 1988, they had operated in an unanalyzed condition and reported the event to the NRC on August 29, 1997 (10 CFR Part 50.72 Event Notification #32843).

(2) Documentation was Lacking to Demonstrate that the Plant Could Safely Shutdown with the Control Room Equipment Remaining Operable, Under Design Basis Assumptions During Elevated Lake Temperatures

The team concluded that the licensee's evaluation of the qualified life of the control room equipment at elevated lake temperatures of 87.5°F,

compared to the qualified life at the UHS temperature at 76°F, lacked appropriate documentation. Results and conclusions from calculation DCCHV12CR11N, Rev.0, Control Room Temperature Evaluation, dated June 22, 1990, indicated that the minimum panel life at an ESW temperature of 87.5°F is 12 hours and the effect of the elevated lake temperature will be evaluated to determine if the panels will survive long enough to allow safe shutdown of the plant. However, the licensee could not produce adequate documentation that would demonstrate this shutdown capability and that accounted for the conclusion reached in the calculation that the plant could safely shutdown with the safety-related control room equipment remaining operable. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires, in part, that measures shall be established for the selection and review for suitability of application of materials, parts, equipment and processes that are essential to the safety related functions of the structures, systems, and components. This item is designated as URI 50-315,316/97-201-12.

F. Previously Unanalyzed Failure Modes in the Instrument Air System

As part of the special review of the CREVS system, the team evaluated the ability of the CREVS's air-operated valves to perform their safety function. The air supply to these valves was provided by the plant's nonsafety-related instrument air system.

The team asked the licensee if air-operated valves (AOVs) could fail in their present position if the AOVs were exposed to full air header pressure upon loss of an upstream air regulator. Questions poised by the team during the CREVS review prompted the licensee to perform a design safety review of the instrument air system. As a result of the licensee's initial review, the licensee performed additional reviews and identified that potential failures of air regulators could result in over-pressure scenarios to air-operated components that had not been previously considered in the Failure Modes and Effects Analysis. Consequently, the licensee identified that modifications performed to the emergency safety feature (ESF) ventilation system between December, 1996 and August, 1997 introduced the possibility that a single failure could result in the loss of both trains of the ESF ventilation system. This system provides safety-related room ventilation to pump motors, such as the RHR and CC pumps. The licensee reported this as 10 CFR Part 50.72 Event Notification 32939, on September 19, 1997. In addition, the licensee also identified that 20 additional safety-related relief valves to the air headers for each unit needed to be installed to provide adequate overpressure protection downstream of the 20 psig, 50 psig, and 85 psig control air regulators in order to mitigate the effects of a postulated failed air regulator.

The team concluded that failure to analyze all potential failure modes that could render redundant trains of safety-related equipment inoperable was inconsistent with the plant's design assumptions and conclusions reached in the safety analysis. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-13.

G. CCW Flow Rates for Several Components Not Representative of UFSAR Values

The team determined that several components that are cooled or serviced by the CCW system receive CCW flow rates significantly different than the values stated in the UFSAR. Components which receive CCW flow rates that are not representative of UFSAR-specified values are the RCP thermal barriers, RCP upper bearing oil coolers, CCP mechanical seal, gear oil, and bearing oil heat exchangers, and the CCW system heat exchangers.

CCW flow rates to the RCP thermal barriers in both units have ranged between 25 and 35 gpm. The UFSAR-specified nominal and accident flow rates listed in Table 9.5-2 are 140 gpm per train, or 35 gpm per RCP, with UFSAR Table 9.5-2, Note 3 allowing the flow rates to be as low as 28 gpm.

CCW flow rates to the Unit 2 RCP motor upper bearing oil coolers have ranged between 120 to over 200 gpm. The UFSAR-specified upper/lower RCP motor bearing nominal flow rates are 420 gpm per pump (100 gpm upper; 5 gpm lower).

The CCP as-left flow rates for Unit 2 after completion of the March 28, 1997, CCW system flow balance test was in excess of 63 gpm. UFSAR-specified flow rates are 31 gpm for normal conditions, with 40 gpm specified during accident and cooldown conditions.

The calculated CCW flow rate through the shell-side of the CCW heat exchangers during a plant cooldown is 8700 gpm (Cooldown Calculation SAE/FSE-C-AEP.AMP-0088, "D.C. Cook Units 1 & 2 RHR Cooldown Analysis for a JPO," dated August 20, 1997). The UFSAR-specified flow rate for cooldown is approximately 8000 gpm (based on 4.0×10^6 lb/hr). Although the system description indicates that the heat exchangers are designed for 8000 gpm and have specific design features that account for vibration, the team could find no evidence that evaluations were performed to demonstrate the structural adequacy for CCW system operation above the UFSAR-specified flow rate of 8000 gpm.

The team was concerned that the licensee has operated the plant with CCW-supplied flow rates to the above listed components significantly different than the values listed in the UFSAR, especially since the CCW system was allowed, by procedure, to operate above its design basis maximum temperature limit of 95°F. No engineering analysis was performed that demonstrated the acceptability of these conditions, and the UFSAR was not updated to reflect current plant operation. Operation of the plant contrary to the assumptions and values listed in the UFSAR potentially demonstrates that adequate design control measures were not established or maintained. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-14.

In addition, the team was concerned that the actual flow rates through the RCP thermal barriers have ranged between 25 and 35 gpm, without having performed a 10 CFR Part 50.59 evaluation to determine the effects of these low flows at CCW system temperatures of up to 120°F. The team discovered documentation that the licensee had previously considered these lower flows to be of minimal

significance. The licensee stated that the CCW flow to the thermal barrier heat exchangers can be reduced to essentially no flow as long as RCP seal injection is maintained during the SI event (AEP Memorandum, dated October 4, 1990). However, the team stressed the importance of the RCP thermal barrier function to the licensee and concluded that the purpose of the RCP thermal barrier is to protect the RCP from damage and a potential RCP seal LOCA in the event that seal injection is lost. Acceptance by the licensee of conditions with low CCW flows through the thermal barriers with concurrent high CCW system temperatures on the basis that seal injection is normally maintained, and without performing a 10 CFR Part 50.59 evaluation, is not consistent with the assumptions stated in the design and licensing basis of the plant (see Section E1.5.2B(6) for further discussion).

H. GL 89-13 Testing

The team performed a design and performance review of the CCW/ESW heat exchangers and the EDG heat exchangers associated with cooling of the EDG jacket water, lube oil, and aftercoolers. This review was performed based on the preliminary team findings associated with the elevated lake temperatures and in order to determine the adequacy of the testing performed by the licensee and based on the criteria contained in the licensee's program guidance for Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

The team found that the licensee's maximum fouling factor acceptance criterion for the CCW/ESW heat exchangers is 0.00169 or less. This value is the maximum allowable fouling acceptable for the CCW/ESW heat exchanger in order to remove the design heat load. In accordance with the licensee's GL 89-13 program guidance, the licensee may operate the plant at the maximum permitted fouling rate for the duration of an operating cycle. However, the team was concerned that this approach could be non-conservative because there would be no margin to accommodate additional fouling should fouling occur over the operating cycle. Since this maximum allowable fouling factor is used in the licensee's accident and cooldown analyses, the licensee could potentially operate the plant with an actual fouling factor that exceeds the one used in the accident and cooldown analyses. In addition, the team found that the licensee did not include instrument uncertainties in the test acceptance criteria when determining fouling factors. This could add an additional non-conservatism to the calculated fouling factor value. 10 CFR Part 50, Appendix B, Criterion III (Design Control), requires that applicable regulatory requirements and the design basis be correctly translated into specifications, drawings, procedures, and instructions. This item is designated as URI 50-315,316/97-201-15.

The team also found that the licensee's GL 89-13 performance trending of EDG heat exchanger degradation was ineffective. Performance trending of EDG heat exchangers consisted of flowing ESW cooling in-series through the EDG jacket water, lube oil, and aftercoolers. ESW outlet temperatures were recorded and trends were charted over several tests. Results of the temperature profile from the heat exchangers, which was used as a measure of heat exchanger degradation, indicated that temperature values were relatively constant over the several testing periods performed by the licensee. However, the team

identified that the heat exchanger outlet temperature was controlled by temperature control valves, which automatically regulate temperature by changing the flow rate through the heat exchangers. Thus, the data collected only was an indication that the temperature control valves were functioning, and was not an indication of whether there was heat exchanger degradation. 10 CFR Part 50, Appendix B, Criterion XI, Test Control, states that all testing required to demonstrate that components will perform satisfactory in service is to be identified and performed in accordance with written procedures which incorporates the requirements and acceptance limits contained in applicable design documents. This item is designated as URI 50-315,316/97-201-16.

E1.2.1.3 Conclusions

The team concluded that the mechanical design of the CCW system is generally acceptable, although the team identified that plant operating procedures had allowed the system to be operated outside of the design and licensing bases. However, the team determined that the existing design of the Instrument Air (IA) system may have been deficient, given the failure by the licensee to fully evaluate all of the potential failure modes and effects, and that several modifications to the IA system were necessary based on subsequent analysis by the licensee. The team also determined that the licensee had not performed adequate 10 CFR Part 50.59 evaluations to account for plant operation above the maximum UHS temperature with its effects on the CCW system, to adequately evaluate accident consequences and risks to the plant during the 1996 Unit 2 dual CCW/ESW train and refueling outage; and to account for flow rates for some CCW-supplied components that were inconsistent with the UFSAR-specified flow values.

E1.3 Electrical/Instrumentation and Controls

E1.3.1 Electrical

E1.3.1.1 Scope of Review

The team assessed the design and performance capability of the licensee's Electrical Distribution System, with specific focus on the sources of power to the RHR and CCW systems, that are required to remain functional during and following the design basis events. The team reviewed the emergency diesel generators (EDGs), 250Vdc batteries, battery chargers, 250Vdc motor control centers (MCCs), 4160Vac switchgear, 600Vac load centers, 600Vac associated buses, breakers, relays and other miscellaneous components.

The EDGs were reviewed to assess the adequacy of their power rating, ability to start and accelerate the assigned safety loads in the required time sequence, and compliance with the single failure criteria and to the applicable separation requirements. The 250Vdc station batteries, battery chargers and their connected loads were reviewed to assess load current profile and compliance with single failure criteria and applicable separation requirements. The 4kV essential buses, 600Vac essential buses and MCCs with their connected loads were reviewed to assess load current and short circuit current capabilities, voltage regulation, protection and adequacy of cable connections between loads and sources, and compliance with single failure

criteria and applicable separation requirements. The team also conducted plant walk downs of electrical equipment including the cable vault below the control and battery rooms, and verified the installed equipment and cables for configuration, equipment rating and separation criteria.

E1.3.1.2 Findings

Inadequate Justification to Demonstrate Operability of the Unit 2 250 Vdc CD Battery Train

During a walkdown of the Unit 2 Train "CD" 250Vdc Battery Room, the team noted Battery Cell #34 was on an individual cell equalize charge. Temporary modification (TM) 2-IHP-5021.EMP.009, dated June 19, 1997, was performed to provide for a portable charger and cabling that was installed on the cell, which allowed it to be on a continuous equalize charge. The licensee told the team that on June 19, 1997, voltage for Cell #34 was found less than the TS-required minimum voltage of 2.13V. The licensee was able to raise the voltage for Cell #34 above the TS-required voltage to a value of 2.214 Vdc within the TS-required 2-hour LCO. A prompt operability evaluation was performed by the licensee that concluded the battery train was "Operable" since the TS-required voltage was restored within the TS LCO. However, the team determined that this prompt operability evaluation was inadequate because the voltage readings for cell #34, upon which operability of the cell was based, were not taken with the cell on a float charge, as required by TS Surveillance 4.8.2.3.2(b)1, but with the cell on an equalize charge. After the cell's voltage had been raised to above the TS-minimum voltage, the cell remained on a continuous equalize charge for 51 consecutive days, after which the licensee decided to replace the cell on August 11, 1997. The decision to replace the cell was based, in part, on the team's questioning of this issue and the fact that the licensee determined that the cell was still consuming significant amperage and was not at full charge.

The team was also concerned that the licensee's prompt operability determination did not consider the fact that Cell #34 was in a degraded condition, and that the cell showed signs of internal short-circuiting, as evident by dendrite formation on the positive plates and substantial sediment accumulation at the bottom of the cell, and was near end of life. The licensee indicated to the team that the TM was performed because continuous charging over a longer period of time was necessary in order to restore full charge to the cell. However, the cell was consuming an excessive amount of amperage over the 51-day period relative to what should be the expectation for a heathy cell exposed to the same equalize charge and time period. The prompt operability determination also did not determine if the plant's TS would allow a component to remain operable with a continuous equalize charge being applied beyond the 2-hour TS LCO.

The team concluded that although there was not adequate evidence to suggest that the battery train could not perform its function, and that the licensee had an analysis that concluded the battery train could perform its function without the cell (115 cells verses the nominal 116 cells), there was not reasonable assurance that Cell #34 had been restored to an operable condition, especially considering that Cell #34 was made a pilot cell by the licensee

during this event and that proper restoration of operability is required for TS operability. Therefore, the team determined that on June 19, 1997, the licensee did not have adequate justification to restore Cell #34, and subsequently the train, to an Operable status, per TS 3.8.2.3 after the battery train had been declared inoperable. This item is identified as URI 50-316/97-201-17.

E1.3.1.3 Conclusions

The team concluded that with the exception of the Unit 2 CD battery train, the portions of the electrical design that were reviewed appeared adequate and were operating within the design limits for components that perform the safety functions of the ECCS and CCW systems.

E1.3.2 Instrumentation and Controls (I&C)

E1.3.2.1 Scope of Review

The team reviewed applicable sections of the UFSAR, TS, licensing Safety Evaluation Reports, Design Basis Documents (DBD), System Descriptions (SD), vendor documents, and other documentation, conducted interviews, and performed walkdowns of the RHR and CCW systems, as related to the instrumentation and controls features, to ensure the ability of the RHR and CCW systems to meet UFSAR commitments and TS requirements. The review consisted of inspection of instrument installations, setpoints, electrical separation, equipment qualification, and 10 CFR Part 50, Appendix R shutdown provisions. The team also reviewed modifications performed by the licensee to verify the as-built plant to the design and licensing bases.

E1.3.2.2 Findings

A. Since 1990, the Licensee Has Not Always Met the 10 CFR Part 50, Appendix R Borated Water Requirement For an Alternate Flow Path

As a result of questions poised by the team regarding the reactor coolant makeup requirement after a postulated Appendix R fire, the licensee reviewed two Appendix R calculations. As a result of this review, the licensee reported on August 28, 1997, that they had been in an unanalyzed condition by not meeting the requirements of 10 CFR Part 50, Appendix R due to an apparent failure by the licensee to maintain the 87,000 gallon RWST requirement at various times in the past since 1990. The licensee issued Condition Report 97-2358 to address not properly controlling the availability of a borated water inventory when one of the units was in Mode 5 or 6. 10 CFR Part 50.72 Event Notification #32839 was made to the NRC documenting this event.

The team found that the licensee is required to maintain available for the respective unit a borated water supply from the RWST of the opposite unit in order to meet the Appendix R requirements to mitigate the consequences of a postulated fire. Separate from the Appendix R requirement, are TS requirements TS 3.3.5 (applicable in Modes 1-4), and TS 3.1.2.7 (applicable in Modes 5 & 6). The TS requirements are to maintain available a borated water

supply for each of the respective units, in order to mitigate the consequences of a boron dilution event.

During the course of the team's review, the licensee stated that the calculation of record for the Appendix R requirement, Mechanical Calculation (MC) Th-90-02, "RCS Volume Make-up Required After Appendix R Fire," dated February 20, 1990, was revised in 1990, that resulted in an increase in the previous volume of borated water from 30,268 gallons to 87,000 gallons. The calculation also noted that the 87,000 gallons requirement was less than TS 3.1.2.7 minimum requirement of 90,000 gallons when a unit is in operational modes 5 or 6. This statement was determined to be misleading because the TS requirement did not satisfy the Appendix R requirement for an "alternate" borated water supply.

As a result, the team concluded that since 1990, the licensee had made two errors that contributed to potentially not always complying with the 10 CFR Part 50, Appendix R requirement. The first error was that the plant operating procedure that implements the Appendix R requirement, Procedure PMI-4100, was not revised to reflect the newer, 1990 calculation of record for the minimum borated water inventory requirement of 87,000 gallons. At the time of this inspection, the plant operating procedure still referenced the older 30,268 gallons requirement. The second error discovered by the licensee as a result of their review, and while investigating questions posed by the team, was that the newer calculation of record, MC TH-90-02, had led the licensee to believe that the Appendix R 87,000 gallon requirement was "enveloped" by the TS requirement for 90,000 gallons. However, the TS requirement, which is germane only to the respective unit, is not relevant to the requirement to maintain available 87,000 gallons of borated water from the "other" unit. Therefore, the licensee reported that at various times since 1990, when the "other" unit had depleted its RWST to less than 87,000 gallons of water, they were unaware that they were no longer in compliance with the Appendix R requirement to maintain available an adequate supply of borated water from the "other" unit.

In summary, the licensee apparently did not always maintain available for the respective unit an alternate borated water supply from the RWST of the opposite unit as required by 10 CFR Part 50, Appendix R. It was apparent to the team that the causes for this error was that the licensee failed to recognize that the newer calculation of record identifying the 87,000 gallons requirement should have superseded the older 1986 calculation reflecting the 30,268 gallons requirement, the controlling operating procedure (PMI-4100) for the Appendix R requirement had not been revised to reflect the correct minimum inventory requirement of 87,000 gallons in the RWST, and that the newer calculation of record misled the licensee into believing that the TS requirement (TS 3.1.2.7) of 90,000 gallons enveloped the 87,000 gallon Appendix R requirement. Thus, the team concluded that procedural and calculational controls were not adequate for controlling the availability of a borated water inventory for Appendix R operation during Modes 5 and 6. This contributed to the plant operating in an unanalyzed condition at various times since 1990. 10 CFR Part 50 Appendix B Criterion III (Design Control), requires, in part, that regulatory requirements and the design basis be correctly translated into procedures and instructions. This item is designated as URI 50-315,316/97-201-18.

B. Setpoint Control Program - Instrument Uncertainty Calculations

As a result of instrument uncertainty concerns identified with the RWST, containment, and containment sump level instrumentation, the team performed a broader, programmatic review of the licensee's Setpoint Control Program, focusing on the adequacy of instrument uncertainty calculations. This included a review of the administrative procedures governing the structure and conduct of the program and the review of selected instrument and control loops. The team reviewed design calculations, associated operating procedures, design and licensing basis, vendor data, calibration records, operator logs, surveillance records, and installed equipment and interviewed engineering and technical support staff. Areas examined included setpoints for equipment interlocks, alarms, design specifications, operating conditions, UFSAR commitments and technical specifications.

The team identified a number of problems, including failure to perform uncertainty calculations, failure to properly account for bias and correction factor errors, as well as failure to properly incorporate the results of the uncertainty calculation into procedures. Based on the number of examples identified by the team, the team was concerned that other examples could exist since a system-based audit approach was used by the team. These examples, which are also discussed in more detail in other sections of this report, are summarized below.

(1) The RWST level instrument uncertainty calculations (ECPs 1-2-I9-03, 1-RPC-09 and 2-RPC-09) did not include an entrance loss factor at the point where the RWST water enters the suction pipe, a velocity head correction factor ($v^2/2g$). (URI 50-315,316/97-201-01)

(2) The licensee had not accounted for the RWST level instrument loop uncertainty in the process setpoint value listed in TS Surveillance Procedure 01(02)-OHP 4030.STP.030, "Daily and Shift Surveillance Checks," Revision 25(23) and in Operating Procedure 01-OHP 4021.017.003, Revision 8, "Removing Residual Heat Removal Loop from Service." (URI 50-315,316/97-201-02)

(3) The licensee failed to adequately consider the potential for vortexing and air entrainment when establishing the RWST low-low level setpoint. (URI 50-315,316/97-201-03)

(4) The uncertainty calculations for the containment and containment sump level instrumentation loops (ECPs 1-2-N3-01, 1-RPC-14, and 2-RPC-14) did not evaluate the impact of the loop uncertainties on post-accident containment levels, and did not include adequate considerations for the RHR or CTS pump NPSH requirements, vortexing, or air entrainment. (URI 50-315,316/97-201-05)

(5) There was no calculation for the CCW heat exchanger outlet temperature loop uncertainty. Only the high and low temperature alarm values were accounted for by the ECP (ECP 1-2-C4-02). This item is designated as URI 50-315,316/97-201-19.

(6) There was no calculation for the ESW intake temperature loop uncertainty. The licensee performed this calculation while the team was onsite, and identified that there was a +/- 3.52°F uncertainty value associated with the instrument loop (see Section E1.2.1.2A for further discussion). This item is designated as URI 50-315,316/97-201-20.

(7) There was no calculation for the control room temperature loop uncertainty. The licensee performed this calculation while the team was onsite, and identified that there was a +/- 5.35°F uncertainty value associated with the control room temperature instrument loop. This item is designated as URI 50-315,316/97-201-21.

The examples listed above demonstrated an apparent lack of adequate design control within the Setpoint Control Program that could affect safety-related equipment or impact the assumptions and requirements found in the plant's TS and UFSAR.

E1.3.2.3 Conclusions

The team found that the design and installation of the instrumentation and control aspects of the RHR and CCW systems were generally acceptable. However, the team was concerned that programatic weaknesses and a lack of adequate program oversight may exist in the licensee's Setpoint Control Program, given the licensee's apparent failure to properly account for instrument uncertainties and bias that could adversely impact the ability of the ECCS system to perform its safety function, as well as impacting the assumptions and operations of the CREVS, ESW, and CCW systems.

E1.4 Consistency and Adequacy of the Design Bases, UFSAR, and TS

E1.4.1 Scope of Review

The team reviewed the appropriate UFSAR and TS sections for the RHR and related ECCS support systems, the CCW and support and interface systems, and the associated electrical and instrumentation and controls sections, to verify consistency between the UFSAR descriptions, TS requirements, and design basis documentation. The team also reviewed portions of the licensee's UFSAR Re-Validation Project with respect to the systems reviewed by the team. The team also focused on the licensee's understanding of their design and licensing basis, and their ability to retrieve design-related information.

E1.4.2 Findings

Prior to this inspection, the licensee did not consider that the UFSAR was a design basis document. Consequently, the licensee evaluated the respective

information contained in the UFSAR on a case-by-case basis, as to what was to be considered design basis information. Because the licensee did not always have prior knowledge of the design basis when performing safety evaluations supporting plant design or procedural changes, the impact and significance of these changes, relative to the assumptions contained in the UFSAR, were not always recognized by the licensee.

The licensee had performed a limited scope review of the UFSAR in April 1996 to verify that the UFSAR information is consistent with the actual plant design and operating procedures. As stated in the licensee's February 6, 1997 response to the staff's request for information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design basis information, inconsistencies were identified by the licensee between the UFSAR and the existing plant design and procedures. The licensee stated that none of the discrepancies were found to be safety significant, but a more extensive review of the UFSAR would be performed. Although the team found instances where plant operation contrary to the assumptions in the UFSAR was significant (i.e., plant operation above the UFSAR UHS temperature and the single failure vulnerability in EOP OHP 4023.ES-1.3), the team recognized that the licensee's review was still in progress at the time of this inspection and the team did not perform a full review of the licensee's response to the NRC staff's 10 CFR 50.54(f) letter.

The team identified a number of concerns with regard to the consistency and adequacy of the design basis, UFSAR, and TS. These are discussed in sections A through D below.

A. Apparent Lack of Understanding of the Design Bases

The team's review of the ECCS, RHR, and CCW/ESW systems revealed that, prior to this inspection, the licensee had an apparent lack of understanding of what constitutes the plant's design basis, the role of the UFSAR, and how each of these are affected by 10 CFR Part 50.59. An example of this was the licensee's belief that they could operate the plant above the UFSAR-specified maximum ultimate heat sink temperature of 76°F, and still be within their design and licensing basis, without performing an appropriate 10 CFR Part 50.59 evaluation and subsequent updating of the UFSAR to reflect the change. Although the maximum ESW intake temperature was stated in both the FSAR and UFSAR (see Section E1.2.1.2 for further discussion), and was used as input into numerous safety-related calculations of record, the licensee did not believe, at the time of this inspection, this was a design basis value. The licensee had operated the plant on occasion above the 76°F UHS temperature limit almost every year since 1988, and had not performed a 10 CFR 50.59 evaluation. Another example was that approved procedures permitted plant operation of the CCW system above the UFSAR-specified maximum temperature of 95°F without having performed a 10 CFR 50.59 evaluation (see Section E1.2.1.2 on plant operation above the ESW/UHS maximum operating temperature). Still another example was the licensee's inability to demonstrate a design basis commitment that the plant could be brought to a cold shutdown condition in 36 hours, to a RCS temperature of 200°F, using only one RHR and CCW train (see Section E1.2.1.2B). Prior to this inspection, the licensee did not believe they were committed to demonstrate this by analysis using one RHR and CCW

train, although this was the assumption that was stated in the staff's September 10, 1973, SER regarding the safety review of the CCW system.

B. UFSAR/TS Inconsistencies with RWST Volume

Based on a review of the licensee's documentation, the team did not clearly understand the design basis requirement for the RWST water volume. UFSAR Section 6.2.2 (page 6.2-17) stated that the RWST is maintained with a minimum of 350,000 gallons of borated water above the bottom of the RWST suction pipe. This statement appears to be accurate, since the bottom of the discharge pipe (approximate elevation is 610'3") is the zero level reference point. However, when the RWST level decreases below the top of the ECCS discharge pipe (approximate elevation 612'3"), vortexing and air entrainment are possible, and ECCS and CTS pumps should not be permitted to be in operation in order to avoid potential pump damage. This would indicate that between elevation 610'3" and 612'3", approximately 27,000 gallons would not be usable. In addition, on page 6.2-17, the UFSAR stated that a minimum volume alarm is provided to assure that 350,000 gallons of useable water are in the RWST.

Additional confusion existed when the team reviewed TS Section 3/4.5.5, which required the licensee to maintain available a minimum "contained" volume of 350,000 gallons of borated water in the RWST. The TS Bases stated that the contained (RWST) water volume limit includes an allowance for water not useable because of tank discharge line location or other physical characteristics. This would imply that not all of the 350,000 gallon volume is useable. The team noted that the licensee discussed some of these observations in a memorandum, dated November 4, 1993, and as documented in condition report CR 93-1212. This issue was also more recently discussed during the licensee's UFSAR Re-Validation Project, and as documented by CR 97-2165. Based on the discrepancies noted above, this issue is designated as IFI 50-315,316/97-201-01.

C. UFSAR RHR Pump NPSH Discrepancies

Several documentation discrepancies were noted regarding RHR pump NPSH values as stated in the UFSAR.

(1) UFSAR Section 6.1.1 (page 6.1-9), Engineered Safety Features Criteria, contains a discussion on NPSH criteria, in which the minimum temperature of the RWST water is stated as 70°F. However, for NPSH calculations, the maximum temperature value should be used, which is stated in the licensee's DBD document as 100°F. The team found that this minimum temperature value was used in the past by the licensee as input in NPSH calculations and that Design Basis Document (DBD) Action Item Number AI-DB-12-ECCS-25, documented this error. The team noted that the UFSAR statement has not been revised to reflect the correct temperature value of 100°F.

(2) UFSAR Table 6.1-1 indicates that for the recirculation mode of operation (i.e., ECCS pumps drawing suction from the containment recirculation sump), the available RHR pump NPSH at runout flow is

31 feet (at 4500 gpm) for Unit 1, and 30.5 feet (at 5050 gpm) for Unit 2. The licensee could not produce any NPSH documentation to substantiate these UFSAR values during this inspection.

(3) UFSAR Table 9.3-2 states that the required NPSH value as 11 feet at the maximum flow rate. However, this value appears to be contrary to the value listed in the vender manual. The team's review of the certified pump characteristic curves contained in the RHR pump installation, operation, and maintenance manual (Document Number VTD-INCR-0012, Ingersoll Dresser Pub. # 016-32294) indicates that the correct NPSH value that should be listed in the UFSAR is 19 feet at the maximum flow rate of 4500 gpm. The 11 feet NPSH value corresponds to the design flow rate of 3000 gpm.

D. Inconsistencies with ECCS Level Instrumentation and Equipment Allowed Outage Times (AOTs)

The operator relies on the information provided by the RWST and containment water level instrumentation in order to perform the ECCS pump suction switchover as directed by the EOP. These instruments do not automatically actuate any safety system functions, even though their continued operability is vital to the performance of the ECCS. The team reviewed the licensee's TS in order to identify the RWST water level (ILS-950, 951) and containment water level instrumentation (NLI-320, 321) requirements. The review determined that the RWST level and containment water level instruments are not included in the engineered safety feature (ESF) actuation instrumentation sections of the TS; however, these instruments are included in the TS, as described below.

(1) Section 1.6, Definition of OPERABLE - For an ECCS subsystem to be OPERABLE per TS Section 3/4.5.2, all necessary attendant instrumentation (i.e., the level instrumentation) that is required for the subsystem to perform its function must also be functional.

(2) Section 3/4.3.3.8, Post-Accident Instrumentation - RWST level, containment water level, and containment sump level are all included in this section. A post-accident instrumentation channel has a 30 day AOT.

Discussions with the licensee (including control room operators) confirmed that a 30 day AOT would be applied by the licensee to the subject level instruments in accordance with TS Section 3/4.3.3.8. The licensee stated that this was consistent with the Improved Westinghouse Owners Group Standard TS, and reflected the passive nature of the instruments (i.e., no critical automatic action is assumed to occur from these instruments). However, the team concluded that these level instruments are required to perform their function in order to allow the operator to accomplish the ECCS transition from the injection phase to the recirculation phase of operation. It would appear that the RWST and the containment water level instrumentation AOTs should appropriately be governed by requirements consistent with the AOT requirements of the ECCS subsystems that they support (i.e., 72 hours), in order for those

ECCS subsystems to maintain the same operability criteria. This item has been referred to the NRC staff for further review. This item is designated as IFI 50-315,316/97-201-02.

E1.4.3 Conclusions

The team concluded that prior to this inspection, the licensee did not have an adequate understanding of their design basis. As stated in the licensee's February 6, 1997 response to the staff's request for information pursuant to 10 CFR 50.54(f), regarding the adequacy and availability of design basis information, and as identified in other documentation, such as the licensee's CCW Compliance Review Effort, inconsistencies exist between the UFSAR, TS, and plant procedures and design. As a result of these and other findings contained within this report, the licensee committed to a revision of their February 6, 1997, submittal regarding their response to the staff's request for information pursuant to 10 CFR 50.54(f) on the adequacy and availability of design basis information.

E1.5 Design Control

E1.5.1 Scope of Review

The team reviewed engineering and design documents (drawings, calculations, specifications, etc.) and various operating, surveillance, and test procedures for both the RHR and CCW systems and the support systems, as discussed in previous sections of this report. The team also assessed the adequacy of the licensee's design control process and the safety evaluations performed that support changes to the plant design and safety-related procedures.

E1.5.2 Findings

A. Temporary Procedural Change Implementation Deficiencies Relative to the 10 CFR 50.59 Review Process and TS 6.5.3.1, "Technical Review and Control"

The team identified that the licensee made temporary, non-intent changes to several safety-related operating procedures, where the team determined there was an apparent change in intent or that involved a controlling parameter change, without proper management review and/or first performing a 10 CFR 50.59 screening. These procedures were OHP 023.4023.ES-1.3, "Transfer To Cold Leg Recirculation," OHP 4021.019.001, "Operation of the ESW System," and OHP 4021.016.003, "Operation of the CCW System During Reactor Startup and Normal Operation." The temporary change process and the specific changes to these three procedures are discussed below.

The licensee's guidance for making temporary changes to safety-related procedures is part of their change sheet process that is described in Procedure OPM.003, "Operations Department Procedure Change Sheet Process," Revision 0. This procedure was written to implement the requirements contained in TS Section 6.5.3.1a, "Technical Review and Control." TS Section 6.5.3.1a, allows the licensee to make temporary changes to procedures, without the normal review and approval, as long as there is no change to the intent of

the approved procedure, and so long as the change is approved by two members of the plant staff, where at least one of the staff members holds a current Senior Reactor Operator (SRO) license. The licensee may defer the full review and approval process of the procedural change for up to 14 days after implementation. For changes to procedures that involve a change to the intent, full approval of the change shall be performed prior to implementation. In addition, Plant Manager Instruction PMI-2010, "Instructions, Procedures, and Associated Indexes Policy," Revision 23, which contains additional related guidance, states that a change of intent is a change that redefines the limits of intended use for either the stated purpose or objective, major activity(ies), operating mode(s), or range(s) or limit(s) of installed equipment or system operation.

The team reviewed the following three temporary procedure changes and determined that each of these procedures were inappropriately revised under the temporary change process. The licensee inappropriately characterized the temporary changes as non-intent changes. However, the team determined that a change to the intent did occur, and full review and approval was not obtained prior to implementation of the temporary change. Details of these changes are described below.

(1) On August 22, 1997, EOP 01-OHP 4023.ES-1.3, Transfer to Cold Leg Recirculation, Revision 4, was temporarily revised with Change Sheet No. 1, that raised the containment water level permissive value and deleted the operator reliance on the containment sump water level instrumentation, in response to the team's concerns regarding vortexing in the containment recirculation sump. The licensee viewed raising the containment level setpoint as a conservative change with respect to continued ECCS pump operation during the recirculation phase of a LOCA. However, the temporary change was an apparent non-conservative change with regard to the effect on the RWST low-low level setpoint. This is an apparent change of intent, not only with regard to the non-conservatism on RWST level, but with respect to the change in range or limit of system operation, as defined by PMI-2010. This change was implemented prior to a complete 10 CFR 50.59 screening and full signature approval.

In addition, a 10 CFR Part 50.59 safety evaluation screening was completed on August 25, 1997. The licensee concluded that a 10 CFR Part 50.59 safety evaluation was not required because the procedure changes were conservative in nature and the change did not constitute a change to a procedure as described in the FSAR. However, the team determined that this conclusion was incorrect because the basic procedural steps contained in the EOP were described in UFSAR Section 6.2.1, "Change-Over from Injection Phase to Recirculation Phase." Section 6.2.1 of the UFSAR also contained an additional sentence that stated, "the detailed sequence for the changeover from injection to recirculation is given in an emergency operating procedure."

(2) 12-OHP 4021.019.001, "Operation of the Essential Service Water System," Revision 13 - Change Sheet No. 3, dated August 14, 1997, reduced the maximum ESW operating temperature from 87.5 to 85°F. Although the licensee performed this change as a conservative, non-intent change, the team pointed out that the new 85°F limit still allowed the plant to operate outside of the design basis limit of 76°F. In addition, the change was to a controlling parameter which changed the operational limit on the maximum UHS temperature. Therefore, the licensee should have made the determination that a 10 CFR 50.59 screening and evaluation was required before changing the procedure.

(3) 2-OHP 4021.016.003, "Operation of the CCW System During Reactor Startup and Normal Operation," Revision 7, Change Sheet No. 1, dated August 28, 1997, deleted the allowance to operate the CCW system at 120°F, which was above the UFSAR-specified maximum operating temperature of 95°F. The licensee made this change as a non-intent change. However, the team determined that this was a change to the intent because it involved a change to the CCW system's operational limit.

The team concluded that the licensee's Procedure OPM.003, "Operations Department Procedure Change Sheet Process," Revision 0, regarding the criteria that defines what constitutes a non-intent change to a procedure, and defining what is a change to the intent of a procedure was not consistent with the requirements contained in TS 6.3.5.1 and with regard to performing changes to the plant, as described in 10 CFR Part 50.59. Additionally, the team determined that the individuals performing the safety screenings (i.e., SROs) lacked appropriate training to determine that changes to a procedure with regard to system operating or controlling parameters, are also to be considered changes to the intent of a procedure. These safety screening deficiencies resulted in the licensee performing temporary changes to procedures without proper approval and/or review, as required by 10 CFR 50.59 and TS 6.5.3.1.

As a result, the licensee directed on August 27, 1997, that until otherwise directed, approval of any procedure change sheets would not be granted unless submitted with an approved safety evaluation screening. The examples listed above (Examples 1-3) are designated as URI 50-315,316/97-201-23.

B. 10 CFR Part 50.59- Potential USQs and Safety Evaluation Deficiencies

The team identified several apparent USQs that may have existed as a result of changes to plant operations and/or to procedures. These examples occurred during plant operations in the summer of 1988, resulting in operating both units above the UHS maximum design basis temperature, and during preparation and preliminary operations for the 1996 Unit 2 full core offload with a dual CCW/ESW train outage. In addition, the team identified multiple examples where the licensee had made changes to the plant without performing a 10 CFR 50.59 evaluation, and several examples where 10 CFR 50.59 evaluations were deficient due to the narrow scope of the evaluation and erroneous conclusions.

Although these examples have been discussed in other sections of this report, they are repeated here in order to summarize the related issues. The specific examples are listed below.

(1) The team determined that the licensee had operated the plant above the maximum UHS temperature limit without performing a 10 CFR 50.59 evaluation, which also potentially created an USQ with regard to a reduction of safety margin as defined in the TS Bases for the control room emergency ventilation temperature limit of 95°F. (Report Section E1.2.1.2E(1))

(2) The licensee's 10 CFR Part 50.59 evaluation, dated March 11, 1996 and March 20, 1996, respectively, that was performed to evaluate the consequences of the 1996 Unit 2 full-core offload, failed to recognize the significance that the CCW heat exchanger could not perform its function under the design basis assumptions that were stated in the UFSAR and other licensing basis documentation. In addition, the licensee also failed to address UFSAR Section 9.4, regarding the criteria for SFP cooling time-to-boil events and subsequently failed to identify that the conclusions reached in the evaluation would have potentially reduced the time-to-boiling in the SFP, given the assumptions stated in the SFP loading calculation and in the UFSAR. Reduction in the time-to-boiling criteria potentially impacts the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the safety analysis report and is an apparent USQ. (Report Section E1.2.1.2C 1)

Additionally, this safety evaluation also failed to identify that a dual CCW/ESW train outage with one unit in refueling and the other unit at power, was contrary to the UFSAR assumptions and placed the plant in an unanalyzed condition and outside of the design bases. This condition also potentially increased the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the safety analysis report, creating the potential for a USQ. (Report Section E1.2.1.2D)

(3) The 10 CFR Part 50.59 safety evaluation that was performed by the licensee for Revision 2, dated June 1992, to EOP OHP 4023.ES-1.3 was not effective in identifying that the revision was creating a single failure vulnerability that could render one RHR pump and both safety-related trains of SI and CC pumps inoperable. Subsequent procedural revisions (Revisions 3 & 4, dated January 1996, and January 1997, respectively) failed to identify the single failure vulnerability. (Report Section E1.1.1.2D)

(4) In 1996 (Unit 2) and 1997 (Unit 1), the licensee filled in the containment recirculation sump roof vent holes without performing a 10 CFR 50.59 evaluation. The licensee stated that the holes were sealed because they were not indicated on any plant design drawings and because the they could not locate any design

requirement for consideration of the vent holes. (Report Section E1.1.1.2E 1)

(5) Procedure 01/02-OHP 4021.016.003, "Operation of the Component Cooling Water System During Reactor Startup and Normal Operation," was revised to remove the statement, "allowing three hours at CCW temperatures of 120°F." The licensee implemented the revision under the auspices of a non-intent procedural change, as allowed by TS 6.5.3.1. However, the team determined that this revision constituted a change to the intent of the procedure, contrary to the conclusion reached by the licensee. (Report Section E1.2.1.2C)

(6) The licensee consistently operated the plant with less than the UFSAR-specified CCW flows through the RCP thermal barriers and without performing a 10 CFR 50.59 evaluation. This issue is of concern because the CCW system has operated above the maximum design basis CCW temperature limit of 95° F, and was allowed to operate at temperatures up to 120°F, without evaluating the impact on the RCP thermal barriers with the reduced CCW flows. (Report Section E1.2.1.2G)

(7) The licensee identified that they have operated the plant without over pressure protection to the RHR system, contrary to the assumptions stated in UFSAR Chapter 9.3. This change to the design basis feature that provides over pressure protection to the RHR system was to defeat the interlocks associated with ICM-129 and IMO-128, RHR hot leg inlet isolation valves, when operating in Mode 4. However, the change was performed without performing a 10 CFR 50.59 evaluation. (Report Section E1.5.2C 3)

The team determined that for the plant design changes listed in Examples 1-7 above, there were apparent failures to perform, or adequately perform safety evaluations, and apparent failures to identify USQs, as required by 10 CFR Part 50.59.

Based on the examples listed in Sections A and B above, and as supported by discussions elsewhere in this report, the team was concerned that the licensee apparently lacks an adequate understanding of the 10 CFR 50.59 evaluation process. This conclusion is based on the significant number of deficiencies involving 10 CFR 50.59 evaluations reviewed by the team, the several apparent USQs identified by the team, which were previously unknown to the licensee, and the several temporary procedural change implementation deficiencies relative to the 10 CFR 50.59 review process and TS 6.5.3.1, "Technical Review and Control," listed in Report Section E1.5.2A above.

C. Apparent B31.1 and ASME VIII Code NonCompliance

The licensee is committed to the American Standard Code for Pressure Piping (ASA B31.1), 1967 edition. The team's review of the RHR and CCW systems

identified several examples where plant drawings and design specifications do not conform with the B31.1 Code requirements. These examples are listed below.

(1) Apparent Conflict with the Piping Specification and Classification of CCW Piping Inside Containment.

The team identified that the piping design specifications and the information indicated on the CCW flow diagram for portions of the CCW piping to and from the RCP thermal barrier were not in agreement. Flow Diagram, "Component Cooling Unit No 1," OP-1-5135 Revision 33, dated January 3, 1996, contained two different pipe class specifications for the same portion of CCW piping. These specifications were M-12 and M-12X class piping. It appeared to the team that these two pipe classifications were associated with that portion of CCW piping that is required to withstand full RCS pressure in the event of a RCP thermal barrier tube rupture. The team further noted that on CCW Flow Diagram, "Component Cooling Unit No 1," OP-1-5135, Revision 33, dated January 3, 1996, the CCW relief valves SV62-1, 2, 3, and 4 on the CCW discharge side for each RCP thermal barrier inside containment have a set pressure of 2485 psig. This portion of piping was also designated as pipe class M-12.

Upon the team's request, the licensee could not find any document that described what type of pipe specification was designated as M-12X. The licensee reviewed this discrepancy and issued CR 97-2303, that indicated the correct piping specification was M-12. As identified in CR 97-2303, the licensee agreed to revise the drawing to delete the reference to the M-12X specification. However, Pipe Class Specification M-12 was designated by the licensee as piping with a design pressure of 150 psig and a design temperature of 115°F. Given this apparent discrepancy between the licensee's pipe specification and the CCW relief valve set pressure, the team requested the licensee to confirmed that low pressure piping was not actually installed in a system that could be exposed to full RCS pressure. Based on the team's concerns, the licensee issued CR-97-2349 to address the pipe specification M-12 designation of 150 psig and 115°F for these pipe segments and the potential impact on the licensee's Small Bore Reconstitution Program.

(2) Apparent Deviation from the B31.1 Code Requirement for Overpressure Protection in the CCW System

The team identified that intervening stop valves were installed in portions of low pressure CCW piping between the source of high pressure (CCW return piping from the RCP thermal barriers) and the CCW surge tank. This is applicable to both units, and is contrary to the B31.1 Code requirements. The low pressure CCW return lines from the RCP thermal barriers have valves which could isolate portions of low pressure piping from the CCW surge tank, which is the designated overpressure protection device.

UFSAR section 9.5.3 states that the relief valve on the CCW surge tank is sized to relieve the maximum flow rate of water which enters the surge tank following a rupture of a RCP thermal barrier cooling coil. The set pressure assures that the design pressure of the CCW system is not exceeded. Paragraph 122.6.1 of the B31.1 Code requires that there shall be no intervening stop valves between piping being protected and its protective device or devices. The licensee could not produce any documentation that a deviation to the Code requirement was approved. Based on the team's concern, the licensee documented this finding in CR 97-2437. The team noted that although this issue was an apparent unapproved code deviation, there was no recognized failure mode (other than human error) that would cause these block valves to isolate the CCW system from the surge tank. Consequently, the team determined that with appropriate controls accepted by a code deviation, this deficiency would not impact the overall adequacy of the mechanical design of the CCW system.

(3) RHR Low Pressure Protection Interlock Defeated During Mode 4 Operations

In followup to the team's questions concerning RCS overpressurization of the RHR system, the licensee identified that at various times in the past while operating in Mode 4, the licensee was defeating the interlocks associated with ICM-129 and IMO-128, RHR hot leg inlet isolation valves. The purpose of the interlocks is to prevent the operators from opening the valves when the RCS pressure is above approximately 400 psig and the valves automatically close when RCS pressure exceeds 600 psig. The licensee reported that the reason for defeating the interlocks was to prevent inadvertent closure and loss of RHR suction during shutdown cooling operation when in Mode 4.

However, this change to the design basis by defeating the interlocks also defeats the over pressure protection afforded by the automatic closure function, as described in Chapter 9.3 of the UFSAR. The licensee reported that this change had been performed without performing a 10 CFR 50.59 evaluation (see Section E1.5.2B7 for further discussion). A CR was initiated on September 11, 1997, to document this finding. The licensee reported this event to the NRC on September 11, 1997 under 10 CFR Part 50.72(b)(2)(I), Degraded or Unanalyzed Condition While Shutdown (event # 32914).

(4) Apparent Lack of Overpressure Protection for the CCW Heat Exchangers

The team identified that there was no apparent overpressure protection design features to the CCW heat exchangers. UFSAR Table 9.5-1 identifies ASME B&PV Code Section VIII 1968 Edition as the applicable code for the CCW heat exchangers. Flow Diagram - Component Cooling Unit No. 1, OP-1-5135A Revision 32, dated April 28, 1994, does not identify any overpressure protection for the CCW heat exchangers. The licensee reviewed the applicable Section VIII code paragraphs and found applicable requirements for overpressure protection within paragraphs

UG125(a) and UG125(c). The licensee could not produce documentation that they were granted a code deviation to this requirement. CR 97-2435 documents this finding.

For examples 1 through 4 above, the team concluded that proper design control was not maintained regarding industry standards, and those standards apparently were not appropriately incorporated into the plant design and operating procedures. 10 CFR Part 50 Appendix B Criterion III (Design Control), requires, in part, that design bases are correctly translated into specifications, and drawings; and that appropriate quality standards are specified and included in design documents and deviations from such standards are controlled. These examples are collectively designated as URI 50-315,316/97-201-24.

D. Plant Equipment Abandoned In Place Without Proper Review and Control

The licensee has abandoned equipment in place without following approved procedures that address the proper administrative actions and controls.

Policy 227000-POL-5400-02, Revision 0, "Treatment of Abandoned In Place Items," dated July 14, 1995, provides guidance for addressing equipment abandoned in place. This policy (section 6.4.1) states that items abandoned in place should typically be removed as part of the design change process. The policy also requires that items that are to be abandoned in place (not removed) during the design change process shall be documented in the project acceptance/design change process, and that appropriate drawings and information databases shall be updated to reflect the configuration status of abandoned in place equipment.

During a walkdown of the licensee's safety-related pump rooms, the team noticed that the reciprocating charging pumps were abandoned in place. After concerns were raised by the team, the licensee identified at least 8 pieces of plant equipment had been abandoned in place, (the reciprocating charging pumps were abandoned in the early 1980s), without following the above established policy guidance.

These examples are:

(1) Reciprocating Charging Pumps for both units
(2) 15 gpm Waste Evaporator
(3) South Boric Acid Evaporator
(4) Oil lines on Unit 1 Main Turbine EHC Fluid System
(5) Concentrates Holding Tank and Transfer Pumps
(6) Flux Mapping System CO_2 Fill Equipment
(7) Air Assist Piping for Draining Condenser Waterbox
(8) Main Steam Conductivity Instrumentation

The team concluded that for the examples listed above, the licensee did not follow established policy guidance contained in Policy 227000-POL-5400-02, Revision 0, "Treatment of Abandoned-In-Place Items." 10 CFR Part 50 Appendix B Criterion III (Design Control), requires, in part, that design changes, including field changes shall be subject to design control measures

commensurate with those applied to the original design. This item is designated as URI 50-315,316/97-201-25.

E. Design Documentation Discrepancies

(1) Design Calculations Not Revised to Account for the Higher UHS Lake Temperatures Above 76°F.

Since 1988, when it became known to the licensee that the UHS lake temperature could exceed the maximum design basis value of 76°F, important to safety design calculations were not updated to incorporate the higher temperatures. Examples of these calculations are listed below.

(a) Calculation TH-86-09, Revision 0, Cooldown Rate Achievable with 1 RHR Pump and Heat Exchanger, dated December 19, 1986. This calculation uses 76°F as the maximum ESW temperature and also uses 95°F as the CCW temperature leaving the CCW heat exchanger. This calculation was not updated to account for the higher lake temperatures.

(b) Calculation HXP880614MWD, Revision 0, RHR Heat Exchanger Primary Outlet Exit Temperature, dated June 21, 1988. This calculation uses 95°F as the CCW temperature exiting the CCW heat exchanger. Based on an increase of ESW temperature from 76 to 87.5°F, the temperature of CCW should have been revised to 108°F.

(c) HXP900531AF, Revision 1, ESW Flow Requirements, dated June 27, 1990. This calculation uses CCW temperature of 95°F. Based on an increase of ESW temperature from 76°F to 87.5°F, the temperature of CCW should have been revised to 108°F.

(2) RHR Pump NPSH Calculation

During the review of the RHR pump NPSH Calculation ENSM 791107AVF, several non-conservative inputs were identified by the team. These were:

(a) The source of a curve showing RHR pump required NPSH could not be determined, and the curve showed less required NPSH than the certified pump characteristic curves. The licensee initiated CR 97-2223 to address this item.

(b) The NPSH available calculation used the minimum RWST temperature (70°F), as opposed to the maximum RWST temperature (100°F). This item was documented in DBD action item AI-DB-12-ECCS-25.

(3) RWST Level Instrument Uncertainty Calculations

During the team's review of RWST level instrument uncertainty calculation ECP No. 1-RPC-09, Revision 2, several errors were noted in Calculation No. 1, Page 23 of 24 of the ECP, as listed below.

(a) The RHR pump centerline elevation was incorrectly identified as Elevation 598'9". The correct elevation is 575'0" as shown on drawing 1-5415-18, Auxiliary Building RHR, Containment Spray & Safety Injection Pumps Suction Piping. The team noted that this error had been previously identified by the licensee in CR 97-2211.

(b) The required NPSH value for the RHR pump is incorrectly given in UFSAR Table 9.3-2 as 11 feet. This corresponds to the RHR pump design flow rate of 3000 gpm. However, for the worst case RHR pump runout flow rate of 4500 gpm, the required NPSH value is 19 feet, as determined from the certified pump characteristic curves contained in the RHR pump installation, operation, and maintenance manual (document no. VTD-INCR-0012; Ingersoll Dresser Pub. # 016-32294).

None of the above calculational deficiencies appear to impact the conclusions reached by the licensee regarding adequate NPSH. However, these calculations are additional examples where there was an apparent lack of adequate design control.

(4) Drawing Discrepancies

The team identified the following drawing discrepancies.

(a) Condition Report CR 96-1373 identified that the CCW flow requirement for each waste gas compressor is 50 gpm each. UFSAR table 9.5-2 identifies flow as 50 gpm and this has been interpreted as 50 gpm total or 25 gpm each. The system flow diagram Component Cooling Unit No 1 OP-1-5135 Revision 33 dated January 3, 1996, identifies the flow to the equipment as 50 gpm total or 25 gpm to each compressor unit and was not identified as an affected document to be corrected in CR 96-1373.

CR 97-0904 identifies the CCW flow rate for the steam generator blowdown samples as 57 gpm. This flow rate is not reflected in the individual branch line flows or main header flow summations on diagram Component Cooling Unit No 1 OP-1-5135 Revision 33, dated January 3, 1996.

The licensee concluded that the flow rates as depicted on the diagrams were there to provide additional information. CR 97-2313 was initiated to correct (enhance) the drawings, as required.

(b) As noted in Section E1.1.1.2e of this report, a horizontal perforated plate that was part of the original recirculation sump design was removed in 1979 by RFC DC-12-2361. The team reviewed drawing 1-2-3178-14, Containment Bldg. Slab at elevation 598'9 3/8", Masonry Sections, and noted that the perforated plate was still shown in one of the sections on this drawing. The licensee initiated CR 97-2344 to address this drawing discrepancy.

(c) The team identified an error in wiring diagrams (engineering and plant systems), 1-93011, 1-93048, 2-93011 and 2-93048 for 4kV Auxiliary Buses T11A, T11D, T21A, and T21D. The note box located at D1 on all prints does not reflect the correct termination of jumper wires when the spare component cooling pump is powered from the respective cubicle. The licensee initiated CR 97-2304 to evaluate the discrepancy.

Drawing discrepancies have the potential to impact plant operating procedures, and maintenance activities that use drawings, and affect assumptions used in related calculations. The team concluded that these deficiencies were apparent failures to maintain adequate design control, as required by 10 CFR Part 50, Appendix B, Criterion III. These examples (a, b, and c) are collectively designated as URI 50-315,316/97-201-26.

F. Procedural Deficiency Involving the Potential for CCW Flashing

Calculation NEMP 960519AF, "CCW LOCA/Cooldown Analysis for U2 Uprating Program," stated that Procedure OHP4021.001.004, "Plant Cooldown from Hot Standby to Cold Shutdown," contained the caution statement, "Avoid CCW fluid flashing in the RHR heat exchangers by limiting RHR parameters to 350 ^{0}F @ 1000 gpm," between steps 4.21 and 4.22. However, the team determined that this procedure does not contain this caution statement identified in Calculation NEMP 960519AF. The team was concerned that the potential for water hammers and other damaging transients may not have been properly addressed by operating procedures. 10 CFR Part 50 Appendix B Criterion III, Design Control requires, in part, that the design be correctly translated into procedures and instructions. This item is designated as URI 50-315,316/97-201-27.

E1.5.3 Conclusions

The team identified a number of examples in multiple discipline areas involving a lack of adequate design control. These included failure to perform, or adequately perform 10 CFR Part 50.59 evaluations, inadequate design documentation for containment water volume and TS 3.0.3 plant cooldown analysis, failure to identify potential USQs, identification of American Standard Code for Pressure Piping (ASA) B31.1 and ASME code deficiencies, setpoint control program deficiencies, improper control of abandoned equipment, calculation and procedural deficiencies, and drawing deficiencies. The team concluded that some of these examples allowed the plant to operate outside of its design basis, and placed the plant in an unanalyzed condition.

Other examples, such as the B31.1 and ASME VIII code deficiencies, increase the potential for overpressurization of the CCW and RHR systems. Collectively, the team concluded that the licensee has not maintained adequate design control for portions of the RHR, ECCS, and CCW systems.

Appendix A

List of Open Items

This report categorizes the inspection findings as unresolved items (URIs) and inspection follow-up items (IFI) in accordance with Chapter 610 of the NRC Inspection Manual. A URI is a matter about which the Commission requires more information to determine whether the issue in question is acceptable or constitutes a deviation, nonconformance, or violation. The NRC may issue enforcement action resulting from its review of the identified URIs. By contrast, an IFI is a matter that requires further inspection because of a potential problem, because specific licensee or NRC action is pending, or because additional information is needed that was not available at the time of the inspection.

Item Number	URI orIFI	Title - Issue (Report Section)
50-315,316/97-201-01	URI	Apparent failure to recognize and evaluate all RWST level measurement error and uncertainties. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2A)
50-315,316/97-201-02	URI	Incorrect RWST level acceptance criterion specified in TS surveillance procedure could have allowed the RWST level to be less than the TS requirement. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2A2)
50-315,316/97-201-03	URI	Apparent failure to consider potential for vortexing and air entrainment when establishing the RWST Low-Low Level setpoint - 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2A3)
50-315,316/97-201-04	URI	Apparent failure to take prompt corrective action after the 1993 SBICI finding regarding the potential for vortexing and air entrainment in the RWST, and after documented by the licensee in 1995 in CR 95-1015. 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Action) (E1.1.1.2A3)
50-315,316/97-201-05	URI	The uncertainty calculations for the containment and containment sump level instrumentation loops do not account for the impact on the post-accident containment water levels (ECPs 1-2-N3-01, 1-RPC-14, and 2-RPC-14), and do not consider the potential for vortexing, air entrainment, or NPSH requirements. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2B3)

Item Number	URI orIFI	Title - Issue (Report Section)
50-315,316/97-201-06	URI	Apparent failure to demonstrate, using design basis documentation, that there was adequate containment recirculation sump water volume following a LOCA. 10 CFR Part 50.46 (ECCS performance criteria) (E1.1.1.2C)
50-315,316/97-201-07	URI	Apparent failure to preclude a single active failure when performing changes to the plant, which is contrary to the assumptions in the UFSAR and the design basis. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2D)
50-315,316/97-201-08	URI	Apparent failure to maintain the 1/4-inch containment recirculation sump particulate retention requirement, which could allow the ECCS throttle valves and containment spray nozzles to become inoperable. 10 CFR Part 50.46 (ECCS performance criteria) (E1.1.1.2E2)
50-315,316/97-201-09	URI	ECCS pump suction valves not leak-rate tested to confirm accident analysis assumption. 10 CFR Part 50, Appendix B, Criterion XI, (Test Control) (E1.1.1.2G4)
50-315,316/97-201-10	URI	Apparent failure to demonstrate, using design basis documentation, that the plant could perform a TS 3.0.3 shutdown in 36 hours to 200°F using one CCW train and design basis assumptions. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2B)
50-315,316/97-201-11	URI	Apparent failure to correctly translate the as-built design of the CCW heat exchanger into safety-related calculations and analyses. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2B)
50-315,316/97-201-12	URI	Apparent lack of documentation to demonstrate that the control room equipment was qualified at worst case operating temperatures in the control room. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2E2)
50-315,316/97-201-13	URI	Apparent failure to analyzed all potential failure modes of the instrument air system that could render redundant trains of safety-related equipment inoperable. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2F)
50-315,316/97-201-14	URI	Operation of the plant with CCW-supplied flows to safety-related and important to safety components contrary to the values stated in the UFSAR. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.2.1.2G)

Item Number	URI orIFI	Title - Issue (Report Section)
50-315,316/97-201-15	URI	Apparent failure to establish controls to prevent potential operation of the CCW system with the CCW heat exchangers above the maximum fouling factor value established by the GL 89-13 testing program. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.2.1.2H)
50-315,316/97-201-16	URI	Performance testing of the EDG heat exchangers was not able to detect degradation, as required by the licensee's GL 89-13 testing program. 10 CFR Part 50, Appendix B, Criterion XI, (Test Control) (E1.2.1.2H)
50-315,316/97-201-17	URI	Inadequate justification to return the Unit 2 250 Vdc Battery Train CD to an operable status (E1.3.1.2)
50-315,316/97-201-18	URI	Apparent failure to maintain adequate design and procedural controls that allowed the plant to operate in Modes 5 and 6 without an adequate volume of borated water in the other unit's RWST in order to meet Appendix R requirements. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.2A)
50-315,316/97-201-19	URI	Apparent failure to perform instrument uncertainty calculation for the CCW heat exchanger outlet temperature loop uncertainty. 10 CFR Part 50 Appendix B, Criterion III, (Design Control) (E1.3.2.2B5)
50-315,316/97-201-20	URI	Apparent failure to perform instrument uncertainty calculation for the ESW intake temperature loop uncertainty. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.2B6)
50-315,316/97-201-21	URI	Apparent failure to perform instrument uncertainty calculation for the control room temperature loop uncertainty. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.2B7)
50-315,316/97-201-22	URI	Apparent programmatic deficiency with the Setpoint Control Program concerning the ability to perform and account for instrument uncertainties. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.3)
50-315,316/97-201-01	IFI	UFSAR and TS inconsistencies with RWST volume (E1.4.2B)
50-315,316/97-201-02	IFI	The RWST and the containment water level instrumentation AOTs should appropriately be governed by consistent AOT requirements (E1.4.2D)

Item Number	URI orIFI	Title - Issue (Report Section)
50-315,316/97-201-23	URI	Performing changes to safety-related procedures without apparent proper review and/or approval, contrary to the provisions of TS 6.5.3.1 and 10 CFR Part 50.59 requirements. (E1.5.2A)
50-315,316/97-201-24	URI	Apparent failure to maintain proper design control regarding industry standards and codes. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.5.2C)
50-315,316/97-201-25	URI	Apparent failure to maintain adequate design control and follow established procedures for equipment abandoned in place. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.5.2D)
50-315,316/97-201-26	URI	Apparent failure to maintain adequate drawing control that has the potential to impact plant operating procedures, and maintenance activities that use drawings. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.5.2E)
50-315,316/97-201-27	URI	Apparent failure to adequately translate design basis assumptions into Plant Procedure OHP4021.001.004, Plant Cooldown from Hot Standby to Cold Shutdown. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (1.5.2F)

Appendix B

Exit Meeting Attendees

NAME	ORGANIZATION
Bob Najuch	Stone & Webster
Bob Hogenmiller	Stone & Webster
Maty Yeminy	Stone & Webster
Venie Varma	Stone & Webster
Dennis Vandeputte	Stone & Webster
Stuart Richards	USNRC/NRR
John Thompson	USNRC/NRR
John B. Hickman	USNRC/NRR
Bruce Burgess	USNRC/RIII
Mark Ring	USNRC/RIII
Dave Butler	USNRC/RIII
Bruce Bartlett	USNRC/Senior Resident Inspector
Joseph Maynen	NRC Resident Inspector
Brian Fuller	NRC Resident Inspector
Byron Bradley	AEPNGG/NEPM
Michael Finissi	Mgr., Electrical Systems
Jeff McClelland	Electrical Systems
Vic Suchodoiski	Electrical Systems Support
A. Alan Blind	AEP/Site VP
James A. Kobyra	AEP/Chief Nuclear Engineer
Eugene E. Fitzpatrick	AEP Exec. VP NUCGEN
Paul A. Barrett	AEP Perf. Assurance Director
Donald R. Hafer	Plant Engineering Director
Steve Brewer	Director, Regulatory Affairs
Kenneth R. Baker	Prod. Engineering Director
Paul G. Schoepf	S.R. Machine System Manager
Mark S. Ackerman	AEP Mechanical Design Manager
Pat Mangan	AEP Mechanical Design Manager
Gary A. Weber	AEP/Nuclear Engineering Staff
Robert G. Vasey	AEP/Licensing
Randy Ptacek	AEP/Licensing
Mary Beth Depuvyst	AEP/Licensing
Gordon P. Arent	AEP/Operations
Frank Pisarsky	AEP/Engineer
Lenny Ormson	AEP/Engineer
Leah Smart	AEP/Licensing
Richard Strasser	AEP/Operations
John Schrader	AEP/Operations
Roger Rickman	AEP/Operations
Billy R. Zemo	AEP/ENPP
S.K. Fadow	AEP/IEL Engineering
J. Kovack	AEP/I&C Engineering
J. Tilly	AEP/Operations
Bob Westerhof	Palisades/Conf. Control
Kerry Tonead	Palisades/Licensing

Exit Meeting Attendees (Continued)

NAME	ORGANIZATION
Daniel T. Malone	Palisades/Mangr, Conf. Control
Thomas Kruft	AEP Corporate Communications
Jeffrey Sankay	AEP/Electrical
Robert Hennen	AEP/ENSM
Danny Boston	AEP/ENSB
Sanford Wolf	Interal Perf. Supervisor - PPA
Walter McCrory	Engineering - PLE/ENSM
Amy Olvera	Licensing
Tom Quaka	PM, & Install. Services
Michelle Popko	Engineering-ENSM
Mike Barfelz	Licensing
Steve Fischeft	Licensing
Abe Loffi	Perf. Engineering
Dave Seipel	OPS Training
Dennis Loope	Training
Lisa Tatrault	Training
Tom Andert	Chemistry
Dan Buman	Cooper Nuclear Station
Jeb Kingseed	Nuclear Safety & Analysis
Karl Toth	Nuclear Licensing
Jim Benes	Nuclear Engineering
Kevin Henderson	Maintenance
Terry Wagoner	Maintenance
John Boesch	Maintenance
Doug Noble	Radiation Protection
Joel Wiebe	Performance Assurance
Herb Brinkmann	EPRI Consult to Licensing
Patrick Russell	Plant Protection
Alberto Verteramo	Rx Engineering/PT Mgr.
Terry Postlewait	NE Design
William R. Burgess	Operations
Dale Spencer	PRA
Lee H. Van Ginhoven	Materials Management
Rich Vonk	Operations
John (Ted) Conrad	Operations
Ned Wollenslegel	Operations
Mark Palen	PPA
Robert Gillespie	Operations
Dennis R. Bean	PA
Andre Feliciano	ENSM
Roderick Simms	Performance Assurance
Doug Malin	Nuclear Fuel
Dennis Kruer	Performance Assurance

Appendix C

List of Acronyms

AEP	American Electric Power
AEPSC	American Electric Power Service Corporation
AH	Ampere Hour
ANSI	American National Standards Institute
AOT	Allowed Outage Time
AR	Action Request
ASA	American Standard Code for Pressure Piping
ASME	American Society of Mechanical Engineers
BAST	Boric Acid Storage Tank
BLDG	Building
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CNP	Cook Nuclear Plant
CR	Condition Report
CREVS	Control Room Emergency Ventilation System
CSD	Cold Shutdown
CTS	Containment Spray
DBA	Design Basis Accident
DBD	Design Basis Document
DCP	Design Change Package
dP	Differential Pressure
ECCS	Emergency Core Cooling System
ECP	Engineering Control Procedure
ECP	Engineered Control Procedure
EDCR	Engineering and Design Change Request
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure, also referred to as 01(02)-OHP 4023.ES-1.3, "Transfer to Cold Leg Recirculation"
ESF	Engineered Safety Feature
ESFA	Engineered Safety Features Instrumentation and Actuation
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GL	Generic Letter
gpm	gallons per minute
HSD	Hot Shutdown
HVAC	Heating Ventilating and Air Conditioning
Hx	Heat Exchanger
I&C	Instrumentation and Control
ICD	Instrument Configuration Document
IFI	Inspection Follow-up Item
IP	Inspection Plan
IST	Inservice Testing
JPO	Justification for Past Operation
kV	Killovolt

List of Acronyms (Continued)

LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LTOP	Low Temperature Overpressure Protection
MOV	Motor Operated Valve
MWt	Megawatt
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OE	Operating Experience
OP	Operating Procedure
PMI	Plant Manager Instruction
PPC	Plant Process Computer
PR	Problem Report
PSICP	Plant Instrument Setpoint Control Program
psid	Pounds Per Square Inch, Differential
psig	Pounds Per Square Inch, Gauge
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFC	Request For Change
RHR	Residual Heat Removal
RTD	Resistance Temperature Detector
RWST	Refueling Water Storage Tank
SBICI	System Based Instrumentation and Control Inspection
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SFP	Spent Fuel Pool
SI	Safety Injection
SSFI	Safety System Functional Inspection
SSPS	Solid State Protection System
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USQ	Unreviewed Safety Question
Vac	Volt Alternating Current
Vdc	Volt Direct Current