UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

July 14, 2006

Charles D. Naslund, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT - NRC SPECIAL INSPECTION REPORT 05000483/2006011

Dear Mr. Naslund:

On April 11-14, 2006, the U.S. Nuclear Regulatory Commission (NRC) conducted a special inspection at your Callaway Plant. The inspection effort continued with in-office and additional on-site reviews through June 16, 2006. The purpose of the inspection was to evaluate the impact of the discovery that component cooling water would not be established to the residual heat removal heat exchangers until after the postloss-of-coolant accident recirculation phase was initiated. The enclosed report documents the inspection findings, which were discussed on June 26, 2006, with Mr. Tim Herrmann and members of your staff.

The inspection was conducted as a result of your staff's identification, during a plant simulator exercise, that component cooling water to the residual heat removal heat exchangers would not have been established until the containment recirculation phase of emergency core cooling system injection had been initiated. The failure to establish procedures that were consistent with the safety analysis could have challenged the ability of the emergency core cooling system in performing its safety functions during the containment recirculation phase. As discussed in detail in the enclosed report, because the underlying safety concern was corrected on March 30, 2006, and does not represent a current safety concern, the inspection focused on the circumstances that lead up to your staff identifying this condition, AmerenUE's response, including the root cause and extent of condition reviews, and the identification of any generic issues related to the design and operating practices that resulted in this condition.

This inspection report documents several opportunities prior to March 27, 2006, including operating experience and review of other emergency operating procedure deficiencies, to identify that the established emergency operating procedures did not ensure that the facility would be operated in accordance with the safety analysis. In addition, the inspection team identified that, after the condition was identified, the immediate actions that were taken to place the plant in a configuration to meet the safety analysis did not adequately consider the component cooling water system response to a loss of offsite power. The plant was subsequently placed in a configuration that supports the design basis component cooling water system requirements.

Based on the results of this inspection, the NRC identified two findings, each evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there was a violation associated with each of the findings. These violations are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. These noncited violations are described in the subject inspection report. In addition, a licensee-identified violation, which was determined to be of very low safety significance, is listed in the report. If you contest these violations or the significance of the violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Callaway Plant facility. In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

William B. Jones, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-483
License: NPF-30

Enclosure:
Inspection Report 05000483/2006011
w/attachments: Supplemental Information
Timeline Describing CCW to RHR Heat Exchangers Problem
Charter Memorandum dated April 10, 2006

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-483
License:	NPF-30
Report:	05000483/2006011
Licensee:	AmerenUE
Facility:	Callaway Plant
Location:	Junction Highway CC and Highway O Fulton, Missouri
Dates:	April 11-14, 2006, with additional on-site in-office inspection through June 16, 2006
Team Leader:	D. Dumbacher, Senior Resident Inspector, Project Branch B
Inspectors:	G. Pick, Senior Reactor Inspector, Engineering Branch D. Loveless, Senior Reactor Analyst
Approved By:	W. B. Jones, Chief, Project Branch B, Division Reactor Projects

SUMMARY OF FINDINGS

IR 05000483/2006011; 04/11-06/16/06; Callaway Plant: Special Inspection to evaluate AmerenUE's discovery that component cooling water flow to the residual heat removal heat exchangers would not have been established until after the postloss-of-coolant accident recirculation phase was initiated.

This report covered the initial on-site inspection conducted April 11-14, 2006, with in-office review and additional on-site inspection conducted through June 16, 2006, by a special inspection team consisting of one resident inspector, one region-based reactor inspector, and one region-based senior reactor analyst. Two noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to take adequate corrective action to prevent recurrence of a significant condition adverse to quality. Specifically, AmerenUE failed to correct the Emergency Operating Procedure deficiencies associated with Final Safety Analysis Report requirements following an April 15, 1998 notification of the same deficiencies at another standardized nuclear unit power plant system plant. At that time AmerenUE did not identify and correct similar deficiencies involving the component cooling water system support function for residual heat removal heat exchangers. The Emergency Operating Procedure deficiencies were discovered by plant personnel on March 27, 2006, during a simulator exercise involving the transition to the emergency core cooling system recirculation phase. Problem identification and resolution crosscutting aspects were identified for the failure to adequately identify and correct Emergency Operating Procedures deficiencies to ensure operation within the design basis.

  This issue was more than minor because it affected the Mitigating Systems cornerstone objective of equipment reliability. The failure to provide for component cooling water system flow through the residual heat removal heat exchangers for initial containment recirculation could result in a loss of the component cooling water system and thus become a much more significant safety concern. AmerenUE's evaluation of the condition was considered for the time allowable to establish component cooling water flow before a loss of the component cooling water system would occur. AmerenUE provided an evaluation that demonstrated a loss of component cooling water would not occur based on the timing of operator actions. Because the timing did affect the probabilistic risk assessment for human reliability, a Phase 3 risk assessment was performed by an NRC senior reactor analyst. The analyst determined that the finding was of very low safety significance, Green. AmerenUE entered this issue into their corrective action program as Callaway Action Request 200602565 (Section 03).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for AmerenUE's failure to implement appropriate corrective actions for maintaining component cooling water flow consistent with design basis requirements. On April 11 and 12, 2006, AmerenUE placed the Train A component cooling water system in a configuration which could result in component cooling water pump runout in the event of a loss-of-coolant accident coincident with a loss of offsite power. Crosscutting aspects associated with problem identification and resolution were identified for the failure to implement appropriate corrective actions to ensure the component cooling water system remained operable for other design basis events.

  This issue was more than minor because it affected the Mitigating Systems cornerstone objective of equipment reliability in that a loss of one train of the component cooling water system could cause other mitigating equipment (i.e., pumps and heat exchangers) to fail and thus become a much more significant safety concern. Using the NRC Inspection Manual Chapter 0609, Significance Determination Process, Phase 1 Screening Worksheet, the finding was determined to be of very low safety significance because it did not result in a loss of safety function for a single train for greater than its Technical Specification allowed outage time. AmerenUE entered this issue into its corrective action program as Callaway Action Request 200602995 (Section 04.02).

B. Licensee-Identified Finding

A violation of very low significance, which was identified by AmerenUE, has been reviewed by the inspectors. Corrective actions taken or planned by AmerenUE have been entered into AmerenUE's corrective action program. This violation and the corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

01 Background

01.1 Summary of Discovery and Immediate Response to Component Cooling Water (CCW) System Operability for Emergency Core Cooling System (ECCS) Containment Recirculation

On March 27, 2006, operations personnel were conducting emergency operating procedure (EOP) validations on the plant simulator to verify "time critical" manual operator actions. During this activity a senior reactor operator identified a concern with the timing of CCW initiation during ECCS containment recirculation. Although the validation actions were not specifically being conducted to validate the time at which CCW would be initiated, the operator noted that CCW may not be established to the residual heat removal (RHR) heat exchangers until after the postloss-of-coolant accident (post-LOCA) recirculation phase was automatically initiated. Subsequently, the Callaway Training Department requested that Wolf Creek Generating Station provide information on CCW initiation for ECCS recirculation and a calculation for the allowed maximum design basis CCW temperatures from a previous NRC violation (50-482/9812-01).

On March 29, 2006, Callaway received the requested information and the review was completed on March 30, 2006. Corrective Action Request 200602565 was initiated the same morning. The concern with the timing of CCW initiation during ECCS containment recirculation was then relayed to the Operations shift crew who aligned CCW to the RHR heat exchangers to provide continuous flow during power operation.

In accordance with Management Directive 8.3, " NRC Incident Investigation Program," the NRC determined that a special inspection was warranted, in part, on the basis of the potential safety significance of a loss of CCW. AmerenUE established a root cause team and a past operability determination team on April 6, 2006. The NRC chartered a special inspection which began on April 11, 2006. The inspection team completed all aspects identified in the charter on June 16, 2006. The team used NRC Inspection Procedure 93812, "Special Inspection," to perform the scope identified in the inspection charter, dated April 10, 2006. The charter may also be found in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS) under Accession Number ML061010217. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Reading Room).

01.2 Impact of CCW Initiation to RHR Heat Exchangers Following Post-LOCA Recirculation Phase

The Callaway Final Safety Analysis Report (FSAR), Section 9.1.3.2.3 and Table 6.3-8, specified that the operators initiate CCW flow to the RHR heat exchangers as the refueling water storage tank (RWST) level neared the automatic transfer setpoint and prior to the recirculation phase. This was significant because the automatic transfer of RHR pump suction from the RWST to the containment recirculation sump would introduce hot water, approximately 265EF, to the RHR heat exchangers. Containment ECCS recirculation, without CCW cooling flow to the RHR heat exchangers, would heat up the shell side of the RHR heat exchangers to temperatures in excess of the design bases CCW system temperature and possibly cause boiling of the CCW water.

02 Prior Opportunities to Address Emergency Operating Procedure Deficiencies

02.01 Generic Communications Related to Containment Recirculation

The following provides a summary of selected generic communications applicable to Callaway ECCS containment recirculation and CCW initiation.

10/14/76 Westinghouse issued Letter SLBE 6-803 recommending automatic CCW initiation to the RHR heat exchangers prior to the swapover point. Callaway plant, owned by Union Electric Company, was part of the Standardized Nuclear Unit Power Plant System (SNUPPS) group. SNUPPS documented that the manual action was acceptable as operators were expected, with training, to safely perform the requirement and because automatic action would result in additional unnecessary surveillances. The letter stated that automatic function could be backfitted by the NRC at the FSAR stage.

5/29/80 Westinghouse issued SNUPPS Letter SNP-3346. The letter stated that CCW must be aligned to the RHR heat exchanger prior to swapover in the recirculation mode.

12/82 Generic Letter 82-33, SUPPLEMENT 1 TO NUREG-0737-REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY, Section 7.1, established requirements for licensees to reanalyze transients and accidents and prepare technical guidelines. These analyses were to identify critical operator tasks and were to be the bases for upgraded EOPs. AmerenUE's commitments were to provide a procedures generation package, including a program for validating EOPs. Callaway had several opportunities to validate that CCW is established to RHR heat exchangers prior to transfer to the cold leg recirculation phase.

02.02 Licensee Documents Addressing Callaway Containment ECCS Recirculation

The following provides a summary of selected corrective action and licensing documents involving the Callaway ECCS containment recirculation and CCW initiation.

1980 to 1982 Callaway FSAR was issued. FSAR Section 9.1.3.2.3 and Table 6.3-8 stated that the CCW initiation must be initiated prior to ECCS recirculation mode swapover.

1984 Callaway EOPs were initiated and required, in Procedure ES 1.3, "Transfer to Cold Leg Recirculation," that the CCW to the RHR heat exchangers be initiated. The Westinghouse emergency response guideline (ERG), for Procedure E-1, "Loss of Reactor or Secondary Coolant," did not have a step to open the CCW inlet valves to the RHR heat exchangers. The ERG basis to Procedure ES 1.3, step 2, specifies that the step to align CCW was a "verify" step that assumed previous CCW flow initiation to each RHR heat exchanger.

4/15/1998 Callaway initiated a corrective action document, SOS 98-1577, noting that the NRC had issued Wolf Creek Generating Station a 10 CFR 50.59 violation highlighting that late initiation of the CCW to the RHR heat exchangers could result in 270EF recirculation sump water being introduced to the RHR heat exchangers. Without cooling this could result in exceeding the design temperature of the CCW system and cause boiling to occur (Wolf Creek Generating Station PIR 973483).

5/5/98 In response to SOS 98-1577, Callaway recognized that Procedure E-1 did not have a step prior to entry to Procedure ES 1.3 and added Step 14 to open the CCW inlet valve to each RHR heat exchanger. The change was made as a temporary change notice (TCN 98-0427). The added step was not validated to ensure it would address the concern.

9/5/2002 Callaway corrective action document Callaway Action Request (CAR) 200205499 stated that the Callaway EOP validation process had validated Westinghouse recommendations in regard to EOP steps to enter cold leg recirculation. The CAR stated that Callaway Plant had no interim configuration issues and that FSAR Table 6.3.2 commitments for timing actions during the swapover were met.

1/2/2004 Callaway corrective action document CAR 200400017 noted that Wolf Creek Generating Station required that the CCW inlets to each RHR heat exchanger be opened in 90 seconds or less following the automatic sump swapover. The CAR initiator asked if Callaway had any similar concerns. The Callaway accident analysis group identified no concerns.

1/27/2005 Callaway corrective action document CAR 200500564 stated that FSAR Table 6.3-8 assumed that CCW flow is aligned to the RHR heat exchangers before the RWST low-low-1 swapover point is reached. The initiator questioned why the RWST outflow analysis did not explicitly include times to align CCW flow to the RHR heat exchangers. The response to the CAR was that steps not directly associated with the swapover were not appropriate.

03 Corrective Actions to Address CCW Initiation on Containment Recirculation

a. Inspection Scope

The inspectors reviewed AmerenUE's actions to evaluate EOP deficiencies prior to identifying the concern with the timing of CCW initiation during ECCS containment recirculation on March 27, 2006. The team considered whether AmerenUE's corrective action program had opportunities to identify and prevent the EOP deficiencies associated with ECCS recirculation cooling. Specifically, the inspectors reviewed whether AmerenUE's past reviews adequately considered:

1) System safety function - classification and prioritization of the problem commensurate with its safety significance

2) EOP validation - identification of corrective actions which are appropriately focused to correct the problem

3) Licensing bases requirements including 10 CFR 50.59 reviews

4) Operability/reportability issues

5) Review of operational experience

b. Findings

Failure to Identify and Correct Inadequate EOPs

Introduction: The inspectors identified a Green noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to identify and implement appropriate corrective actions for EOP deficiencies associated with CCW cooling to RHR heat exchangers as required to respond to a large-break LOCA.

Description: On March 27, 2006, during performance of EOP validations on the plant simulator, AmerenUE recognized that CCW would not be established to the RHR heat exchangers until after the post-LOCA recirculation phase was automatically initiated. The automatic transfer of each RHR pump suction path from the RWST to the containment recirculation sump would introduce hot water, approximately 265EF, to each RHR heat exchanger prior to CCW flow being established. This could result in the CCW system exceeding it's design basis maximum temperature.

Callaway FSAR, Section 9.1.3.2.3 and Table 6.3-8, required that operators initiate CCW to the RHR heat exchangers as the RWST level neared the automatic transfer setpoint and prior to the recirculation phase of a LOCA. The hot water, without CCW cooling flow, would heat up the shell side of the RHR heat exchangers to temperatures in excess of the design bases CCW system temperature and possibly create boiling of the CCW water in the RHR heat exchangers. Procedure E-1 "Loss of Reactor or Secondary Coolant," as written, had manual operator actions to align cooled CCW water to the RHR heat exchangers which could not be performed prior to reaching the RWST lo-lo-1 level setpoint. This would cause a delay in cooling hot containment recirculation sump water.

AmerenUE reviewed the simulator data and initiated a CAR on March 30, 2006. AmerenUE established a plant lineup that provided continuous CCW flow through each RHR heat exchanger until a permanent resolution could be established. This addressed the immediate safety concern. The team verified that the failure to meet assumptions in the accident analyses had no impact on peak containment temperatures and pressures for the LOCA accident sequences as peak conditions are mostly a function of the containment spray system function and not the time of initiation of CCW into the RHR heat exchanger.

AmerenUE performed heat transfer calculations and EOP validations to ensure that no boiling of the stagnant CCW water would have occurred prior to initiating CCW water in step 2 of ES 1.3. The heat transfer calculations determined that, over the range of performance of different operating crews, 8 to 37 seconds of margin existed between the initiation of opening the CCW valves and the onset of boiling. Based on plant inservice testing, the valves were fully opened in 50 to 51 seconds. As a result of these very low margins of time to boil, AmerenUE performed impact studies associated with the collapse of steam bubble formation and steam slug flow analyses for the tube region of the RHR heat exchangers. These studies resulted in approximately 60 seconds to boil off the volume (approximately 12 percent of the total volume) of CCW water above the heat exchanger tubes. The conclusion was there would be no significant water hammer or steam slug flow forces created by the collapse of steam that would have been formed. The team independently reviewed the calculations and supporting documentation for these conclusions. This review included EPRI-NP-6766, “Water Hammer Prevention, Mitigation and Accommodation,” and NRC NUREG-CR-6519, “Screening Reactor Steam/Water Piping Systems for Water Hammer.”

AmerenUE documented the following opportunities to have identified and implemented appropriate corrective action to address the inadequate EOP and safety system design aspect:

- Westinghouse Letters SLBE 6-803 and SNP-3346
- Callaway initial FSAR reviews
- Callaway corrective action documents directly associated with the issue (SOS 98-1577, CAR 200106536, CAR 200400017, CAR 200202808, CAR 200503084, and CAR 200507150)
- Operational experience associated with SNUPPs plant (Wolf Creek), NRC Safety System Engineering inspection finding (05000482/1998-012)
- Two EOP change requests and associated 10 CFR 50.59 screening reviews associated with Procedures E-1 and ES 1.3
- Wolf Creek corrective action document problem identification Report PIR 973483

In addition, the team considered the following documents in their assessment of the overall corrective action effectiveness to address the EOP deficiencies associated with containment ECCS recirculation and impact on the supporting safety system design aspect.

- NRC Generic Letter 82-33 response and inclusion into Technical Specification (TS) 5.4.1.b

- Reviews associated with the initiation of Callaway EOPs versus Westinghouse ERGs

- NRC Inspection Report 05000483/2004006 and finding 05000483/2003006-02 documented critical operator EOP response times being exceeded. The deficiency resulted in critical operator response times taking longer than assumed in the accident analysis. AmerenUE review identified three similar extent of condition reviews but missed the noncompliance with FSAR assumptions described in this finding.

- Callaway corrective action documents directly associated with the issue, CAR 200205499 (Callaway EOP validation process had responded to Westinghouse OE regarding EOP steps to enter cold leg recirculation) and CAR 200500564 (FSAR Table 6.3-8 assumed that CCW flow is aligned to the RHR heat exchangers before the RWST low-low-1 swapover point is reached)

<u>Analysis</u>: In accordance with NRC Inspection Manual Chapter 0612, Section 05.01, “Screen for Performance Deficiencies,” the team determined that this issue constituted a performance deficiency because AmerenUE repeatedly failed to identify and correct the issues related to a previous NRC finding (05000483/200306-02), CAR 200500564 and other identified significant conditions adverse to quality. Consequently, AmerenUE had operated the plant for years with the potential for boiling in the shell side of each RHR heat exchanger following a postulated a large-break LOCA. Each missed opportunity to correct inadequate emergency operating procedures was a result of ineffective corrective action reviews, a lack of understanding of the accident analysis and licensing bases, and poor interface between AmerenUE’s accident analysis and emergency operating procedures writers groups.

Phase 1 Screening Logic, Results, and Assumptions

In accordance with NRC Inspection Manual Chapter 0612, Section 05.03, “Screen for Minor Issues,” the inspectors determined that the finding was more than minor. This finding was associated with the equipment performance, reliability, attribute of the mitigating systems cornerstone and was determined to affect the objective of that cornerstone. Specifically, the finding could have resulted in the loss of CCW following a postulated large-break LOCA.

The inspectors evaluated the issue using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Following a postulated large-break LOCA, the component cooling water system would not have functioned without quick operator action because of boiling in the RHR system heat exchanger. This represents a loss of the system safety function. Therefore, the screening indicated that a Phase 2 estimation was required.

Phase 2 Estimation for Internal Events

In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, "User Guidance for Determining the Significance of Reactor Inspection Findings for At-Power Situations," the inspectors estimated the risk of the subject finding using the Risk-Informed Inspection Notebook for Callaway, Revision 2. The inspectors made the following assumptions:

1) The performance deficiency that resulted in inadequate EOPs (failure to establish CCW cooling to the RHR heat exchangers until after the post-LOCA recirculation phase was automatically initiated) existed from April 15, 1998, until March 30, 2006, when licensee personnel revised the procedure to correct the deficiency. Therefore, this deficiency affected plant risk for an extended period of time and the Phase 2 exposure window of greater than 30 days was used to estimate the risk impact of the deficiency over a 1-year assessment period.

2) The failure to establish CCW cooling to the RHR heat exchangers prior to the recirculation phase, following a postulated large-break LOCA, would have resulted in a complete loss of the CCW system without operator intervention.

3) Table 2 of the Risk-Informed Inspection Notebook identified that worksheets for all initiating events except the total loss of service water were applicable when a finding affected the CCW system. However, the senior reactor analyst determined that this performance deficiency only impacted the plant during a large-break LOCA. Therefore, none of the sequences on any other worksheet were applicable or quantified.

4) Table 1 of the Risk-Informed Inspection Notebook identified that the initiating event likelihood for a large-break LOCA having an exposure time window of greater than 30 days was 5. The inspectors noted that the performance deficiency did not increase the likelihood of a large-break LOCA.

5) Given Assumption 2, the inspectors adjusted the low pressure recirculation mitigation in the large-break LOCA worksheet from a credit of 3 to a credit of 0 because cooling would have been lost to the sump without CCW.

6) Despite not being required before the recirculation phase began, the actions to establish CCW cooling to the RHR heat exchangers were proceduralized in the emergency operating procedures. Additionally, operating crews being tested in the plant simulator were able to establish CCW prior to the postulated failure of the CCW system as defined by licensee calculations. Therefore, the inspectors gave operator recovery credit in the worksheet indicating that sufficient time was available to implement the actions, operators had been trained in the procedures that could be implemented entirely from the main control room, and that no special equipment was necessary to complete the actions. Therefore, as

defined in NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, Table 4, "Remaining Mitigation Capability Credit," the inspectors gave a Recovery of Failed Train Credit (P_{CREDIT}) of 1.

Based on the above assumptions, only Sequence 1 of the large-break LOCA worksheet was applicable. The resulting sequences are provided in Table 1 below:

Table 1
Phase 2 Worksheet Results

Initiator	Sequence	Initiating Event Likelihood	Mitigating Functions	Result
Large-Break LOCA	1	5	Operator Recovery of CCW	6

By application of the counting rule, the internal event risk contribution of this finding to the change in delta core damage frequency (ΔCDF) was of low to moderate risk significance (White). The approximate value of this frequency ($\Delta CDF_{PHASE\ 2}$) was calculated by the senior reactor analyst to be 3.3 x 10^{-6}.

Phase 3 Analysis

Assumption 6 made during the Phase 2 estimation process was overly conservative and did not completely represent the actual probability that operators would fail to establish CCW cooling to the RHR heat exchangers prior to the time that the CCW system would no longer be capable of performing its intended safety function following a postulated large-break LOCA. Therefore, the senior reactor analyst performed a modified Phase 2 estimation to better indicate the risk of the subject performance deficiency.

Internal Initiating Events:

The analyst utilized the simplified plant analysis Risk H (SPAR-H) method used by Idaho National Engineering and Environmental Laboratories (INEEL) during the development of the SPAR models and published in NUREG/CR-6883, INEEL/EXT-02-10307, "The SPAR-H Human Reliability Analysis Method," as an appropriate tool for evaluating the probability that operators would establish CCW cooling to the RHR heat exchangers in a timely manner following a postulated large-break LOCA.

The probability ($P_{RECOVERY}$) that operators failed to properly perform the EOPs and/or failed to perform them prior to the failure of the CCW system upon demand was calculated to be 2.0 x 10^{-2}. In calculating this failure probability, the analyst assumed that the nominal action failure rate of 0.001 should be adjusted by multiplying this nominal rate with the following performance shaping factors:

, Available Time: 10

The available time was barely adequate to complete the action. Licensee operating crews in the plant simulator took up to 2-1/2 minutes after the switchover to recirculation to establish CCW flow to both RHR heat exchangers. By licensee calculations, this action would have occurred approximately 1 minute prior to boil off of the CCW water in the isolated RHR heat exchangers.

, Stress: 2

Stress under the conditions postulated would be high. Multiple alarms would be initiated, causing loud, continuous noise in the main control room. Additionally, the operators would readily identify that a large break had occurred in the reactor coolant system and would understand that the consequences of their actions would represent a threat to plant safety.

, All remaining performance shaping factors were considered to be nominal under the subject conditions.

Using this more realistic operator recovery credit, the analyst recalculated the change in core damage frequency as follows:

$$\Delta CDF = \Delta CDF_{PHASE\ 2} \div P_{CREDIT} * P_{RECOVERY}$$

$$= 3.3 \times 10^{-6} \div 0.1 * 2.0 \times 10^{-2}$$

$$= 6.6 \times 10^{-7}$$

By modification of the Phase 2 estimation and in accordance with NRC Inspection Manual Chapter 0609, Appendix A, Attachment 1, Phase 3, “Risk Evaluation Using Any Risk Basis that Departs from the Phase 1 or Phase 2 Process,” the analyst determined that the internal event risk contribution of the subject finding to the ΔCDF was of very low risk significance (Green). The best estimate value of this frequency was calculated by the senior reactor analyst to be 6.6×10^{-7}.

External Events

The plant-specific SDP worksheets do not currently include initiating events related to fire, flooding, severe weather, seismic, or other external initiating events. In accordance with Manual Chapter 0609, Appendix A, Attachment 1, step 2.5, "Screen for the Potential Risk Contribution Due to External Initiating Events," experience with using the site-specific Risk-Informed Inspection Notebook has indicated that accounting for external initiators could result in increasing the risk significance attributed to an inspection finding by as much as one order of magnitude. Therefore, the analyst assessed the impact of external initiators because the Phase 2 SDP result provided a risk significance estimation of 7 or greater. However, the analyst determined that the likelihood that an external event could result in a large-break LOCA was so small as to be negligible to the quantification of the risk of the subject performance deficiency.

Potential Risk Contribution from Large Early Release Frequency (LERF)

In accordance with Manual Chapter 0609, Appendix A, Attachment 1, step 2.6, "Screen for the Potential Risk Contribution Due to Large Early Release Frequency (LERF)," the analyst determined that the finding needed to be screened for its potential risk contribution to LERF using Manual Chapter 0609, Appendix H, “Containment Integrity Significance Determination Process,” because the estimated ΔCDF result provided a risk significance estimation of greater than 1×10^{-7}.

According to Appendix H, Section 4.1, the subject performance deficiency represented a Type A finding because the finding influenced the likelihood of accidents leading to core damage. As documented in Appendix H, Table 5.1, the only accident sequences that would lead to LERF for a pressurized water reactor with a large-dry containment like Callaway’s would be steam generator tube ruptures and intersystem LOCAs. The analyst noted that the only affected core damage sequence involved a large-break LOCA initiator. These sequences do not typically result in containment bypass accidents.

Based on the above, and in accordance with Appendix H, the analyst screened out all accident sequences related to the finding as not significant to LERF.

Conclusion

The performance deficiency resulted in a finding that was of very low risk significance (Green). The best estimate change in core damage frequency was 6.6×10^{-7}, representing the risk related to internal initiators. The change in risk related to external events, as well as the change in LERF, was determined to provide only negligible increase in risk.

The inspection team found that this finding has crosscutting implications in the problem identification and resolution performance area. AmerenUE’s inadequate evaluations resulted in not correcting a licensing basis safety issue.

Enforcement: Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI, "Corrective Action," required that conditions adverse to quality are promptly identified and corrected. Further, the requirement states that, in the case of significant conditions adverse to quality, measures shall be taken to ensure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, the corrective actions taken for a previous NRC finding (05000483/200306-02) and CAR 200500564, as well as other identified opportunities to correct the deficient EOP, were a significant condition adverse to quality where the measures taken to ensure that the cause of the condition is determined and corrective action taken to preclude repetition were not effective. This finding is a noncited violation (NCV 05000483/2006011-01) consistent with Section VI.A of the NRC Enforcement Policy. AmerenUE entered this issue into its corrective action program as CAR 200602565.

04 Adequacy of Planned or Completed Corrective Actions

a. Inspection Scope

The team reviewed AmerenUE's immediate corrective actions needed to ensure the function of the RHR heat exchangers in a large-break LOCA event and those actions to prevent recurrence of a failure of the EOPs to meet licensing bases accident analyses.

The corrective actions implemented by AmerenUE involved three sets of actions. The first set of actions was to establish continuous CCW flow through the RHR heat exchangers. This would preserve the FSAR described licensing basis and ensure that the hot containment recirculation sump water does not boil the CCW in the RHR heat exchangers.

The second set of actions was to have a root cause team formed to determine the extent of condition of missed licensing bases related requirements as they apply to the EOPs. This team was to provide immediate evaluation and communication of requirements not clearly met and provide input to the root cause determination for CAR 200602565.

The third set of actions was to ensure that plant procedures and planned maintenance associated with the CCW system were reviewed to ensure compliance with TSs and other aspects of the current licensing bases.

The inspectors independently reviewed the adequacy of AmerenUE's initial and planned corrective actions.

b. Observations and Findings

.1 Upon discovery of the EOP deficiency, AmerenUE placed the plant in a configuration that ensured adequate ECCS flow to the RHR heat exchangers during a large-break LOCA and for containment ECCS recirculation. AmerenUE's root cause team had sufficient resources allocated and performed a thorough extent of condition review of licensing bases requirements pertaining to the EOPs. Three minor licensing bases conflicts with the EOPs were identified and actions were assigned to immediately resolve the conflicts. These are discussed in Section 04.3 of this report. Initial reviews were performed on plant procedures and planned maintenance but they did not effectively prevent a potential CCW pump runout scenario associated with a LOCA with a loss of offsite power event as discussed in Section 04.2.

.2 Corrective Action to Establish Continuous CCW Flow to RHR Results in Possible CCW Runout Conditions

Introduction: On April 12, 2006, the inspectors identified a Green, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, due to AmerenUE's failure to implement and adequately communicate the corrective actions identified to permanently establish CCW to RHR heat exchangers on a large-break LOCA. This failure resulted in potential CCW pump runout conditions for a LOCA with loss of offsite power.

Description: On March 30, 2006, CAR 200602565 (CCW alignment for large-break LOCAs), operability determination, operator night orders, and the acceptance criteria for Engineering Procedure ETP-EG-0002, “Component Cooling Water System Flow Verification,” Revision 4, had identified that CCW train flow should not be run above 7250 klbm/hr due to potential pump runout concerns. On April 11, 2006, CCW Pump A was started to augment Train A CCW Pump C flow to address a Train A charging pump oil cooler low flow alarm. The operating shift then aligned CCW flow to the spent fuel pool to balance Train A CCW system flows. The combined flow for the two CCW pumps was in excess of 8400 klbm/hr. On April 12, 2006, the inspectors informed AmerenUE operating personnel that this alignment was in conflict with the operability determination and may not ensure CCW pump operability in a postulated LOCA with a loss of offsite power. The design of the sequencer for the essential Train A 4160 volt bus would only start CCW Pump A. Pump A would experience a runout condition, causing damage to the pump and possible pump failure. On failure of Pump A, Pump C would automatically start and subject Pump C to the same conditions. Pump C’s reliable operation would then be challenged.

The 10 CFR 50.59 safety evaluation screening for changes to Procedure OTN-EG-00001, “Component Cooling Water System,” Revision 25, on April 7, 2006, discussed potential runout system alignment but did not result in a change to the annunciator response procedure describing what maximum flow for a single CCW pump would be. CAR 200602995 was written to address the inspectors’ concerns. This CAR identified that on two occasions, each less than 14 hours duration on April 11 and 12, 2006, flow in Train A was in excess of 7250 klbm/hr flow. CAR 200602995 also stated that TS 3.7.7, Limiting Condition for Operation, Action A, for an inoperable CCW pump was missed but the 72-hour allowed outage time was not exceeded.

Analysis: The team determined that this finding constituted a performance deficiency. AmerenUE’s corrective action for the EOP deficiency resulted in subsequent plant configurations that did not ensure that a single CCW pump would not be subjected to pump runout conditions. This issue was more than minor because it affected the Mitigating Systems cornerstone objective of equipment reliability and capability of systems that respond to initiating events in that established CCW pump operating flow conditions would not have ensured an operable Train A CCW pump. Using the NRC Inspection Manual Chapter 0609, Phase 1 Screening Worksheet, the finding was determined to be of very low safety significance since it did not result in a loss of safety function for a single train for greater than its TS allowed outage time.

The inspection team found that this finding has crosscutting implications in the problem identification and resolution performance area. The inadequate CCW system flow was a result of inadequate corrective action described in CAR 200602565.

Enforcement: Corrective actions for operability of the RHR system prescribed in CAR 200602565 failed to ensure that each train of the CCW system was available to provide RHR heat exchanger cooling. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI, "Corrective Action," required that conditions adverse to quality are promptly identified and corrected. Contrary to 10 CFR Part 50,

Appendix B, Criterion XVI, on both April 11 and 12, 2006, AmerenUE corrective actions lead to improper CCW system conditions that challenged the CCW train to RHR heat exchanger safety function.

This finding is an NCV (NCV 05000483/2006011-02, Inadequate Corrective Actions Result in Possible CCW Runout Conditions) consistent with Section VI.A of the NRC Enforcement Policy. AmerenUE entered this issue into its corrective action program as CAR 200602995.

.3 Comprehensiveness of the Licensee's Determination of the Extent of Condition

a. Inspection Scope

Through interviews and documentation reviews, the team evaluated the comprehensiveness of AmerenUE's extent of condition review for the failure to implement FSAR design bases requirements. Specifically, the team assessed whether licensee personnel had adequately reviewed procedures and engineering issues associated with the newly established CCW to RHR heat exchangers alignment. Also the inspectors independently reviewed the FSAR and EOPs to assess whether other associated licensing bases requirements were met.

The inspectors reviewed the licensing documents identified by AmerenUE that were not directly met by the current revision of the Callaway EOPs. These were:

- FSAR, Chapter 6, Engineering Safety Features, Section 6.2.2, stated that containment spray cannot be terminated until completion of the injection phase. Procedure E-1, step 7, allowed both containment spray pumps to be turned off, prior to completing the injection phase, provided containment pressure had been reduced below 5.5 psig. Having containment pressure less than 5.5 psig ensured that no adverse impact on dose consequences would occur.

- FSAR, Chapter 9, Auxiliary Systems, Section 9.2.1.2.2.3, stated that auxiliary feedwater low suction pressure signal opened the essential service water system isolation valves to ensure essential service water supply to the auxiliary feedwater system. The FSAR provided only a percent margin to the ultimate heat sink total volume for essential service water system use and not a specific time requirement for the operators to secure the auxiliary feedwater use. The license calculated the time based on the percent margin and determined the operators have 65 minutes to complete the task. This was validated as having sufficient time to perform the task to realign the auxiliary feedwater system suction away from the ultimate heat sink.

- FSAR, Chapter 15, Safety Analysis, Section 15.6.3.2.2, and Table 15.6.1 stated that, following a steam generator rupture accident, a cooldown to RHR conditions using the intact steam generator atmospheric steam dumps must be initiated at approximately 60 minutes from the start of the event. Since this occurs after the leak from the ruptured steam generator tubes is stopped, no additional radioactivity is released. This requirement had no logic bases documented and is being reviewed by AmerenUE.

b. Observations and Findings

The inspectors found that the corrective actions, in response to normal and off-normal procedures associated with the CCW system were generally being correctly applied to the Callaway Plant. CCW annunciator response procedures associated with CCW loads were not appropriately addressed as discussed in the finding in Section 04.02. The inspectors also found AmerenUE had not fully evaluated smaller and medium break LOCA scenarios or possible CCW pump loss of net positive suction head. AmerenUE provided data and calculations to show that these cases with elevated initial CCW temperatures still resulted in no significant impact.

.4 Evaluation of Licensee’s Initial Root Cause Determination

a. Inspection Scope

The team reviewed AmerenUE’s preliminary root cause determination of the failure to implement FSAR design bases requirements for independence, completeness, and accuracy.

b. Observations and Findings

The inspectors found AmerenUE’s direct cause determination to be accurate. This initial licensee report, however, did not emphasize why so many opportunities to identify the issue were missed. Organizational interface ineffectiveness and an inadequate corrective action program allowed AmerenUE’s organization to repeatedly miss opportunities to understand an unanalyzed safety issue. Specifically the bases and sequence of FSAR requirements were not researched when questions arose. This lead to inaccurate and incomplete initial reviews by CAR lead responders and prevented licensee management from becoming appropriately engaged.

The team noted that AmerenUE had identified three preliminary causes of not having established procedures that would ensure RHR heat exchanger cooling in a LOCA prior to automatic introduction of hot containment recirculation sump water. These were discussed in significant condition adverse to quality CAR 200602565.

- The 1984 EOP revision did not match requirements provided in the FSAR wording. The FSAR wording had remained unchanged since initial issue.

- AmerenUE had at least three opportunities to identify and correct the problem. Reviews of corrective action documents and the emergency procedures were narrowly focused.

- Ineffective communication between Callaway and Wolf Creek contributed to the narrow focus of the reviews and corrective action evaluations.

.5 Discussion of the Potential Impact Associated with Boiling in the RHR Heat Exchanger CCW Side

The team reviewed the engineering calculations to evaluate whether the safety function of heat removal from the containment sump following a LOCA could be achieved. Specifically, the concern related to aligning CCW to the shell side of the RHR heat exchangers after the RHR suction path was realigned to the containment sump from the RWST. Because the containment recirculation sump water would be at a saturation temperature of approximately 265EF, boiling of CCW on the shell side of the RHR heat exchangers would occur with no shell side fluid flow.

AmerenUE performed four separate calculations that determined: (1) the temperature rise in the shell side of the heat exchangers, (2) the heat exchanger voiding rate, (3) the magnitude and impact of any resulting water hammer, and (4) the CCW inlet impingement plate response to a water hammer.

The actions to ensure CCW flow is delivered to the shell side of the heat exchangers were specified in the EOPs; consequently, the delivery of CCW to the shell side of the heat exchangers was a function of operator time.

The team determined that AmerenUE used appropriate design inputs, calculation methodologies, and conservative assumptions. The calculations determined that the time required to boil and subsequently void the shell side of the heat exchangers would have been prevented by prior operator action. Further, although it was anticipated that CCW would be delivered to the heat exchangers prior to voiding of the heat exchangers shell, AmerenUE demonstrated the collapse of a steam bubble that would void the space above the heat exchanger tubes would create a water hammer of small magnitude. The maximum force predicted for this case was equivalent to approximately 115 psig.

.6 Generic Implications

a. Inspection Scope

The team reviewed AmerenUE's design bases to determine whether generic issues related to the design and operating practices existed with other Callaway systems or other nuclear plants.

b. Observations and Findings

The team, with assistance from AmerenUE, and the NRC's Office of Nuclear Reactor Regulation, determined that other Westinghouse primary water reactor plants without automatic CCW initiation may not be in full compliance with their licensing bases. This issue is being reviewed by the NRC.

4OA6 Meetings, Including Exit

On April 14, 2006, the team presented the status of the inspection, to date, to Mr. Tod Moser, Manager, Plant Engineering.

On May 2, 2006, the team leader conducted an exit meeting with Mr. Tim Herrmann, Vice President, Engineering, and other members of his staff.

On June 26, 2006, the team leader conducted a supplemental exit meeting with Mr. Tim Herrmann, Vice President, Engineering, and other members of his staff.

While proprietary information was reviewed, no proprietary information is being retained or is included in this report.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by AmerenUE and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

TS 5.4.1.b, “EOP Program," required that requirements of NUREG 0737 as described in Generic Letter 82-33 be adhered to, ensuring that applicable accidental analysis licensing bases are correctly translated into emergency procedures. Contrary to this, AmerenUE had not translated the FSAR described licensing basis into EOPs. Callaway’s requirements to initiate CCW flow to the RHR heat exchangers prior to the opening of the containment recirculation sump valves on a postulated large-break LOCA were not met. Specifically FSAR, Section 9.1.3.2.3, and Table 6.3-8 required that operators initiate CCW to the RHR heat exchangers as the RWST level neared the automatic transfer setpoint prior to the recirculation phase of a LOCA.

The particular function to prevent boiling in the RHR heat exchangers on a large-break LOCA required subsequent analysis to ensure the RHR and CCW functions were not unrecoverable. Through calculations, AmerenUE was able to demonstrate that using the actual EOP step location, operator action occurred in time to prevent boiling.

This issue is more than minor because it was similar to Example 3.I of Appendix E of Manual Chapter 0612. It was necessary for AmerenUE to perform a calculation to determine whether the existing EOPs were acceptable. Because there was available margin in the time to boil and time to RHR heat exchanger tube uncovery calculations, this issue was confirmed not to involve a loss of function of the system in accordance with Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment. Therefore, this issue screens as Green during Phase 1 of the SDP as described in Manual Chapter 0609, Appendix A, Attachment 1.

This issue was identified in AmerenUE’s corrective action program as CAR 200602565.

ATTACHMENTS: SUPPLEMENTAL INFORMATION
TIMELINE DESCRIBING CCW TO RHR HEAT EXCHANGERS PROBLEM
CHARTER MEMORANDUM DATED APRIL 10, 2006

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Fuller, System Engineer
B. Huhmann, Supervising Engineer, Nuclear Engineering Systems, Mechanical
M. Jennings, Operating Supervisor
S. Maglio, Superintendent, Systems Engineering
J. Milligan, Shift Manager, Operations
K. Mills, Supervising Engineer, Regional Regulatory Affairs/Safety Analysis
T. Moser, Manager, Plant Engineering
S. Petzel, Engineer, Regional Regulatory Affairs
T. Herrmann, Vice President, Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000483/2006011-01	NCV	Failure to Recognize and Correct Inadequate Emergency Procedures (Section 03)
05000483/2006011-02	NCV	Inadequate Corrective Actions Result in Possible CCW Runout Conditions (Section 04.2)

LIST OF DOCUMENTS REVIEWED

Calculations

Number	Title	Revision
EJ-M 18 (Wolf Creek)	RHR Pump Recirculation Operation versus Time of Initiation of CCW Flow to RHR Heat Exchangers	1
C-4176-00-01	Callaway RHR Heat Exchanger Shell Side Temperature Rise	0
C-4176-00-02	Callaway RHR Heat Exchanger Transient Voiding Rate	0
C-4176-00-03	Likelihood and Magnitude of Water Hammers in the Callaway RHR Heat Exchanger	0
M-EG-05, ADD 2	Calculate the NPSH Available to the CCW Pumps With the Surge Tank Empty	0
BN–16	Maximum RWST Transfer Volumes and Swapover Times	0

Callaway Action Requests

199100746	19860054	199801577	200106536	200202808
200205499	200400017	200500564	200503084	200507150
200602565	200602908	200602992	200602995	

Drawings

Number	Title	Revision
FSAR Figure 9.2-3, Sheet 1	Piping and Instrumentation Diagram - CCW System	NA
FSAR Figure 9.2-3, Sheet 2	Piping and Instrumentation Diagram - CCW System	NA
M-23EG01(Q)	Piping Isometric CCW System, Auxiliary Building, Train A	6
M-23EG03(Q)	Piping Isometric CCW System, Auxiliary Building, Train B	7
M-23EG04(Q)	Piping Isometric CCW System, Auxiliary Building, Train B	3
M-23EG05(Q)	Piping Isometric CCW System, to Fuel Building, Train B	2
5736	Vertical Residual Heat Exchanger Outline Drawing	5
5739	Vertical Residual Heat Exchanger Details	3
5740	Vertical Residual Heat Exchanger Details	4

Miscellaneous Documents

Number	Title	Revision/Date
Memorandum —4176-00-01	Summary of Screening Criteria for the Evaluation of Steam Water Hammer at Power Plants	April 12, 2006
50.59 Screen RFR 23374	Evaluate the Use of Gothic Software	38204
Safety Evaluation	Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP	Revision 3, September 10, 2001

Miscellaneous Documents

Number	Title	Revision/Date
Inspection Report 05000 482/97-201	Wolf Creek Generating Station Design Inspection,	February 23, 1998
EPRI NP-6766	Water Hammer Prevention, Mitigation and Accommodation	July 1992
PRA Evaluation Request 06-269	Risk Assessment for CCW Flow to RHR Heat Exchanger Issue	38819
NUREG/ CR-6519	Screening Reactor Steam/Water Piping Systems for Water Hammer	September 1997
PIR 973483	Wolf Creek Performance Improvement Request Describing USAR and EMG ES-12 Conflicts	35731
Draft Revision 2	Event & Causal Factors Chart, CAR 200602565 CCW Flow Requirements to RHR Heat Exchangers Not Meeting FSAR	38818
Formal Safety Evaluation for RFR 19025	Callaway FSE for Evaluating FSAR Chapters 6 and 15 as a Result of Calculation BN-16, Revision 0 Which Determined the Maximum Times for Swapover of emergency core cooling system and CS Pumps from Injection Phase to Recirculation Phase for 5 Cases	1998
FSAR Section 9.1	Fuel Pool Cooling System	5/97
FSAR Section 9.2	Component Cooling System	5/97
FSAR Table 6.3-8	Sequence of Changeover Operation from Injection to Recirculation	5/97
Night Order	Callaway Night Order, CCW Alignment Requirements based on CAR 200602565	38813

Miscellaneous Documents

Number	Title	Revision/Date
Night Order	Callaway Night Order, CCW Alignment Requirements based on CAR 200602565	38806

Computer History Printouts

Type	Period
RHR Pump B Current	19 minutes on 2/11/2004
RWST Temperature	3 years beginning 1/1/2001
CCW Flow & Valve Operation	50 seconds of operation to show flow versus valve position

Procedures

Number	Revision	Subject
E-1	5	Loss of Reactor or Secondary Coolant
E-1	6	Loss of Reactor or Secondary Coolant
OTA-RK-00020	1	Annunciator Response Window 52B for CCW Pump A or C Pressure Low
OTA-RK-00020	1	Annunciator Response Window 54B for CCW Pump B or D Pressure Low
OTN-EG-00001	25	CCW System
OTN-EG-00001	26	CCW System
OTO-BB-00002	23	Rcp Off-Normal
OTO-EG-00001	9	CCW System Malfunction
EOP E-1	1B2	Loss of Reactor or Secondary Coolant

Procedures

Number	Revision	Subject
ES-1.3 ERG (background)	NA	Westinghouse Owner Group ERG Background for ES-1.3
ES-1.3 ERG	NA	Westinghouse Owner Group ERG for ES-1.3
ES-1.3	0	Transfer to Cold Leg Recirculation
ES-1.3	5	Transfer to Cold Leg Recirculation
ES-1.3	6	Transfer to Cold Leg Recirculation

ACRONYMS

CAR	Callaway Action Request
CCW	component cooling water
ΔCDF	delta core damage frequency
CFR	*Code of Federal Regulations*
ECCS	emergency core cooling system
EOP	emergency operating procedure
ERG	emergency response guideline
ESW	essential service water
FSAR	Final Safety Analysis Report
INEEL	Idaho National Engineering and Environmental Laboratories
LERF	Large Early Release Frequency
LOCA	loss-of-coolant accident
NCV	noncited violation
RHR	residual heat removal
RWST	refueling water storage tank
SDP	significance determination process
SNUPPS	Standardized Nuclear Unit Power Plant System
SPAR	simplified plant analysis risk
TS	Technical Specification

TIMELINE DESCRIBING CCW TO RHR HEAT EXCHANGERS PROBLEM

October 14, 1976	Westinghouse issued letter SLBE 6-803 recommending automatic CCW initiation to the RHR heat exchangers prior to the swapover point. Callaway Plant, owned by Union Electric Company, was part of the SNUPPS group. SNUPPS felt that manual action was acceptable as operators are expected to be trained and felt that automatic action would result in additional unnecessary surveillances. The letter stated that automatic function could be backfitted by the NRC at the FSAR stage.
May 29, 1980	Westinghouse issued SNUPPS Letter SNP-3346. It stated that CCW must be aligned to the RHR heat exchangers prior to swapover in the recirculation mode.
1980 to 1982	Callaway FSAR issued. In two locations it was stated that the CCW initiation must be prior to recirculation mode swapover. (Section 9.1.3.2.3 and Table 6.3-8.
December 1982	Generic Letter 82-33, Section 7.1, established requirements for licensees to reanalyze transients and accidents and prepare technical guidelines. These analyses were to identify critical operator tasks and were to be the bases for upgraded EOPs. AmerenUE's EOPs were to provide a procedures generation package, including a program for validating EOPs. Callaway had several opportunities to validate that CCW is established to RHR heat exchangers prior to the transfer to the cold leg recirculation phase.
June 7, 1905	Callaway EOPs were initiated and required, only in Procedure ES 1.3, that the CCW to the RHR heat exchangers be initiated. This was contrary to the FSAR sections requiring prior initiation. The Westinghouse ERG, for the Procedure E-1 "Loss of Reactor or Secondary Coolant," response, also did not have a step to open the CCW inlet valves to the RHR heat exchangers. The ERG clearly identified, in the basis to Procedure ES 1.3, step 2, that the step to align CCW was a "verify" step that assumed previous attempts to initiate CCW flow to the RHR heat exchangers.
April 15, 1998	Callaway initiated a corrective action document, SOS (previous CAR name) 98-1577, noting that the NRC had issued Wolf Creek a 50.59 violation highlighting that late initiation of the CCW to the RHR heat exchangers could result in 270EF recirculation sump water being introduced to the RHR heat exchangers. Without cooling, this could result in exceeding the design temperature of the CCW system and cause boiling to occur. (Wolf Creek PIR 973483).

May 5, 1998	Callaway recognized that Procedure E-1 did not have a step prior to entry to Procedure ES 1.3 and added a step to open the CCW inlet valve to each RHR heat exchanger. However the change was made as a temporary change notice (TCN 98-0427) and the 50.59 screening question addressing whether the change was to a procedure as described in the FSAR was answered "NO."
CAR 200205499	Callaway CAR 200205499 stated that the Callaway EOP procedure validation process had validated OE14159 in regard to EOP steps to enter cold leg recirculation. The CAR stated that Callaway had no interim configuration issues and that FSAR 6.3.2 commitments for timing actions during the swapover were met.
January 2, 2004	Callaway CAR 200400017 noted that Wolf Creek nuclear power plant required that the CCW inlets to each RHR heat exchanger be opened in 90 seconds or less following the automatic sump swapover. The CAR initiator asked if Callaway had any similar concerns and the accident analysis group replied "No."
January 27, 2005	CAR 200500564 stated that FSAR Table 6.3-8 assumed that CCW flow is aligned to the RHR heat exchangers before RWST low-low-1 swapover point is reached. The initiator questioned why the RWST outflow analysis did not explicitly include times to align CCW flow to the RHR heat exchangers. The response to the CAR was that steps not directly associated with the swapover were not appropriate.
March 20, 2006	Licensed operator retraining to perform EOP validations questioned whether CCW initiation to RHR heat exchangers was time critical on a large-break LOCA.
March 30, 2006	CAR 200602565 was initiated describing the discovery of the simulator EOP validation. The Operations department placed the RHR heat exchanger CCW alignment in a safe condition.
April 7, 2006	Operability determination and 50.59 screening for changes to Procedure OTN-EG-00001 (CCW system) describe the extent of condition for the current CCW to RHR heat exchangers alignment. Each describes a maximum 7250 klbm/hr flow rate for a single CCW train due to pump runout concerns during a large-break LOCA scenario with loss of offsite power to an engineered safety features bus. Concern is that only a single CCW pump will be sequenced onto the bus with a CCW system alignment for two pump operation.
April 10, 2006	Callaway forms root cause and engineering teams to address the EOP/CCW issue.

April 11, 2006	NRC charters a Special Inspection Team to respond to the discovery that CCW would not be initiated to the RHR heat exchangers prior to auto swapover to the recirculation phase on a large-break LOCA.
April 11, 2006	Low flow on the Train A charging pump oil cooler occurs. Shift operators review annunciator response Procedure OTA-RK-00020 guidance, start a second Train A CCW pump, and increase Train A flow to 8400 klbm/hr.
April 12, 2006	NRC inspector questions the conflict with the operability determination and the actions by the operating crew in response to the 4/11/06 low CCW flow on the charging pump.
April 13, 2006	CAR 200602995 describes two times when the 7250 klb m/hr CCW pump limit was exceeded. One was approximately 14 hours on 4/11/06 and again for 10 hours on 4/12/06.
April 14, 2006	Initial onsite inspection completed by NRC team

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

April 10, 2006

MEMORANDUM TO: David Dumbacher, Resident Inspector, Callaway Station
Project Branch B, Division of Reactor Projects

Greg Pick, Senior Reactor Inspector
Engineering Branch 2, Division of Reactor Safety

FROM: Arthur T. Howell III, Director, Division of Reactor Projects **/RA/**

SUBJECT: SPECIAL INSPECTION CHARTER TO EVALUATE CALLAWAY PLANT COMPONENT COOLING WATER INITIATION TO THE RESIDUAL HEAT REMOVAL HEAT EXCHANGERS DURING THE INITIAL POST-LOCA RECIRCULATION PHASE

A Special Inspection Team is being chartered in response to the discovery that component cooling water (CCW) would not be established to the residual heat removal (RHR) heat exchangers until after the postloss of coolant accident (LOCA) recirculation phase was initiated. This could lead to a failure of the CCW system and a loss of safety injection and other essential loads (such as spent fuel pool cooling). The licensee implemented prompt actions to establish flow to the RHR heat exchangers to restore the safety systems and essential loads to an operable status. You are hereby designated as the Special Inspection Team members. Mr. Dumbacher is designated as the team leader.

A. <u>Basis</u>

On March 30, 2006, the Callaway Plant reported (CAR 200602565) that, during a simulator exercise on March 20, 2006, an operator raised a concern regarding the timeliness of initiation of the CCW flow to the RHR heat exchangers during post-LOCA (large break) recirculation from the containment safety injection sumps. The licensee identified that the sequence of establishing CCW flow, and the delays in its initiation because of the sequence in the emergency operating procedures, could result in the potential to exceed the CCW design temperature during a large LOCA when containment recirculation is first initiated. The licensee found during a simulator exercise that CCW flow to the RHR heat exchangers was not initiated until 4-6 minutes after containment recirculation flow was first established through the RHR heat exchangers. The Final Safety Analysis Report describes that CCW is placed in service prior to refueling water storage tank lo-lo 1 level being reached and the swapover occurring. The licensee had previously established, through the emergency operating procedures, that CCW would be initiated through the RHR heat exchanger following the swapover to containment recirculation. The licensee's identification that the CCW system may not actually be aligned in sufficient time to ensure adequate cooling of the RHR heat exchanger resulted in the licensee questioning their ability to meet design basis requirements. The license's immediate corrective action included aligning and running the CCW system continuously to ensure that adequate cooling water was available to the RHR heat exchanger in the event of a design basis LOCA event.

This Special Inspection Team is chartered to compare the as-found conditions to the licensing basis for containment recirculation; determine if there are generic safety implications associated with the timing of CCW initiation post-LOCA through the RHR heat exchangers; review the identification, evaluation, and determination whether the CCW system and associated safety injection systems were inoperable for the postrecirculation phase; review the licensee's compensatory measures following discovery of the condition; and review the licensee's calculations regarding the impact of the timing of CCW initiation to the RHR heat exchangers as provided in their emergency operating procedures.

B. Scope

The team is expected to address the following:

1. Develop a complete sequence of events related to the discovery of the CCW timing concern for post-LOCA safety injection and the followup actions taken by the licensee.

2. Compare operating experience involving post-LOCA emergency core cooling system (ECCS) cooling requirements to actions implemented at the Callaway Plant. Review prior opportunities to have addressed EOP and/or design considerations associated with ECCS recirculation cooling requirements, including the effectiveness of those actions. Determine if there are any generic issues related to the design and operating practices associated with post-LOCA recirculation and ECCS cooling. Promptly communicate any potential generic issues to regional management.

3. Review the extent of condition determination for this condition and whether the licensee's actions are comprehensive. This should include potential for other EOP validation issues as well as potential ECCS recirculation timing issues.

4. Review the licensee's determination of the cause of any procedural design deficiencies and/or operating practices that allowed the potential for CCW system design temperature to be exceeded. Independently verify key assumptions and facts. If available, determine if the licensee's root cause analysis and corrective actions have addressed the extent of condition for problems with CCW cooling to the safety systems.

5. Determine if the Technical Specifications were met for the ECCS and CCW systems following the implementation of compensatory measures.

6. Determine if the supporting analyses for the licensee's compensatory measures were made in accordance with 10 CFR 50.59.

7. Review the calculations the licensee is developing to evaluate the CCW initiation sequence for post-LOCA ECCS and CCW operability.

8. Collect data necessary to support a risk analysis. Specifically obtain information associated with the degree to which the ECCS and CCW systems would be affected during post-LOCA recirculation, the break sizes that are affected, the containment response, the ability to recover failed pumps and other components, and the dominant accident sequences.

C. Guidance

Inspection Procedure 93812, "Special Inspection," provides additional guidance to be used by the Special Inspection Team. Your duties will be as described in Inspection Procedure 93812. The inspection should emphasize fact-finding in its review of the circumstances surrounding the event. It is not the responsibility of the team to examine the regulatory process. Safety concerns identified that are not directly related to the event should be reported to the Region IV office for appropriate action.

The Team will report to the site, conduct an entrance, and begin inspection no later than April 11, 2006. While on site, you will provide daily status briefings to Region IV management, who will coordinate with the Office of Nuclear Reactor Regulation, to ensure that all other parties are kept informed. A report documenting the results of the inspection should be issued within 30 days of the completion of the inspection.

This Charter may be modified should the team develop significant new information that warrants review. Should you have any questions concerning this Charter, contact me at (817) 860-8248.

cc via E-mail:

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W. Maier	

SUNSI Review Completed: _WBJ_____ ADAMS: / Yes ☐ No Initials: _WBJ
/ Publicly Available ☐ Non-Publicly Available ☐ Sensitive / Non-Sensitive

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4/10/06	4/10/06	4/10/06	4/10/06	

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