ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-382/95-08

License: NPF-38

Licensee: Entergy Operations, Incorporated
P.O. Box B
Killona, Louisiana 70066

Facility Name: Waterford Steam Electric Station, Unit 3

Inspection At: Waterford 3

Inspection Conducted: August 20 through September 30, 1995

Inspectors: Troy W. Pruett, Resident Inspector
David P. Loveless, Senior Resident Inspector, South Texas Project
Jim Melfi, Resident Inspector, Arkansas Nuclear One

Approved: P. H. Harrell, Acting Chief, Project Branch D   11-22-95 Date

Inspection Summary

Areas Inspected: Routine, unannounced inspection of plant operations, maintenance and surveillance observations, onsite engineering, and plant support activities.

Results:

Plant Operations

- The inspectors noted that the control room operations staffs were distracted from their assigned duties by excessive personnel traffic during periods prior to the start of the outage (Section 2.1).

- The formation of a shift support center, at the beginning of the refueling outage, appeared to minimize the number of control room distractions and improved control room decorum (Section 2.1).

- The shift supervisor emphasized the importance of effective communications and provided excellent coordination of multiple tasks during the September 22, 1995, reactor and steam plant shutdown (Section 2.5).

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- There were several recent examples of licensee and inspector identified corrective action program deficiencies that contributed to the repeated lifting of shutdown cooling heat exchanger (SDCHX) relief valves and the auxiliary component cooling water (ACCW)/component cooling water (CCW) hydraulic transient. These deficiencies included: (1) untimely initiation of condition reports (CR) for repetitive occurrences, (2) untimely review of action items in CRs, (3) incomplete evaluation of contributing factors, and (4) untimely closure and implementation of corrective actions. The licensee was in the process of addressing these issues at the end of this inspection period. Additional reviews will be performed and this issue will be tracked as an inspection followup item (Sections 5.2 and 5.3).

Maintenance

- The failure of a procedure to identify the appropriate measuring and test equipment (M&TE) prior to testing the Emergency Feedwater Pump A Breaker 62-2 relay resulted in the inability to complete the test. The maintenance procedure referenced test equipment that was no longer used by the technicians. This issue was cited as an example of Violation 382/9508-01 (Section 3).

- The inspectors identified that the prerequisites for testing main steam safety valves had been inadvertently, because of miscommunications, signed as completed even though the calibration of the M&TE did not meet the requirements of the testing procedure. The failure to ensure prerequisite requirements were met prior to initiation of a test is cited as an example of Violation 382/9508-01 (Section 4).

- The inspectors noted that a local equipment operator, utilizing three-way communications with the control room, corrected a misunderstood pressure reading. This was an example of good communications during testing of the main steam safety valves (Section 4).

Engineering

- A modification was implemented for the removal of a support used to secure a valve locking device and, as a result, the valve was not locked as required by procedure. NRC identified an unresolved item to review the licensee's locked valve program. Additionally, engineering personnel had not notified operations (Section 2.3).

Plant Support

- Continued inattention to detail by plant personnel with respect to the storage of loose items, which was previously identified in NRC Inspection Reports 50-382/94-20, -94-21, -95-06, and -95-07, indicated the requirements for storage of loose items were not being met and that actions taken to correct this ongoing problem had been ineffective. Improper storage of loose items constitutes the third example of Violation 382/9508-01 (Section 2.2).

- The licensee adequately posted radiation areas and established effective temporary radiological boundaries to ensure radioactive materials being staged for Refueling Outage (RFO) 7 were properly controlled (Section 6).

Management Oversight

- The Safety Review Committee maintained adequate oversight of the corrective action program and appeared to be fully implementing the requirements specified in licensee procedures and the Quality Assurance Program Manual (Section 2.4).

Summary of Inspection Findings:

New Items

Violation 382/9508-01: Failure to follow a procedure for performance of safety-related activities (Sections 2.2, 3, and 4).

Unresolved Item 382/9508-02: Review licensee's program for control of locked valves (Section 2.3).

Inspection Followup Item 382/9508-03: Review of the use of CRs to identify plant and equipment deficiencies (Section 5.3).

Closed Items

None

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

The plant operated at 100 percent power during this inspection period until it was shut down, on September 22, 1995, to commence RFO 7.

2 PLANT OPERATIONS (71707)

2.1 Control Room Observations

On September 1, 1995, the inspectors observed, for a 30-minute period, the various activities of the operations crew in the control room. During these observations, the inspectors noted that the control room supervisor and two plant operators were discussing technical issues with members of the maintenance and quality assurance (QA) departments and not monitoring the control panels for a period of approximately 5 minutes. The shift supervisor also noted that the operators had allowed themselves to be distracted from monitoring the control room panels and admonished the operating crew. The inspectors observed that the shift supervisor demonstrated effective oversight by recognizing the distractions and demonstrated initiative by admonishing the operating crew, the instrumentation and control supervisor, and the individual from the QA department.

On September 11, the inspectors again observed a different operating crew being intermittently distracted from monitoring control room panels for approximately 10 minutes, as evidenced by: (1) while the secondary plant operator was monitoring a trainee during performance of an emergency diesel generator surveillance test, the primary plant operator went behind the control room panels to obtain a stepladder and the control room supervisor went to the administrative area, (2) both plant operators and the control room supervisor were distracted on three occasions by technicians either needing to obtain permission to commence work or to discuss the status of ongoing maintenance, and (3) five maintenance technicians were involved with activities in the control room that distracted the plant operators and control room supervisor. In addition, the inspectors noted that informal communications were used when an instrumentation and control technician used a hand gesture and stated, "Its that time again," to indicate that the plant operator needed to manipulate a control switch on the turbine control panel, which the operator did without confirming through a formal communication process which valve needed to be operated.

During control room observations, the inspectors also noted that a sign, installed at the entrance of the control room to limit personnel access to the control room, was having minimal effect. The sign requested that personnel not enter the control room until acknowledged by an operator. On numerous occasions, the inspectors observed personnel walk directly to the operators and engage them in conversation regardless of ongoing activities. The inspectors discussed these observations with the shift supervisor, who stated that he had noted the distractions but did not discuss them with the operators since they were intermittent and of short duration.

The inspectors discussed the examples of a lack of control room formality provided above with the operations superintendent on September 12. The operations superintendent stated that the distractions in the control room were not what was expected by management and that the issue of control room formality and limiting the number of distractions and personnel in the control room would be discussed during the weekly meeting of operations department supervisors on September 13. Subsequently, during two plan-of-the-day meetings, the inspectors noted that the an operations representative requested that department supervisors minimize the number of distractions in the control room by informing subordinates not to enter unless authorized by an operator. On September 15 and 19, the inspectors performed 30-minute observations of control room activities and noted that the operators were limiting the number of personnel in the control room and that personnel were waiting until an operator gave permission to enter.

On September 20, a visiting inspector observed that the control room operators did not monitor plant parameters for approximately 10 minutes. During this 10-minute period, the shift supervisor, control room supervisor, and two trainees were reviewing documentation for the outage, the secondary plant operator was involved with document review and phone communications, and the primary plant operator was reviewing a surveillance procedure in the rear of the control room. After the control room staff became aware that the individual observing the activities was an inspector, the extraneous traffic dissipated and the primary plant operator began monitoring the control panels.

On September 26, 4 days after the outage commenced, the inspectors noted that the number of personnel entering the control room was maintained at a minimum. The operations superintendent informed the inspectors that, for the first time, a shift support center had been established for the control of work to reduce distractions in the control room. The shift support center was established to review and approve maintenance activities and to provide information to personnel that was typically obtained directly from the control room. The inspectors concluded that the establishment of the shift support center was effective in minimizing the number of control room distractions and was considered a strength.

2.2 Storage of Loose Items

During routine tours of plant spaces on August 25 and September 5, 1995, several examples of poor storage of loose items were noted. The inspectors observed that several ladders were left unattended, unrestrained and/or not placed flat, and either in contact with or within falling distance of safety-related equipment. Specifically, ladders had been improperly stowed adjacent to high pressure safety injection (HPSI) Train A flow control valves, SDCHX Train B, SDCHX A CCW inlet pressure Root Valve CCW-950A, and switchgear ventilation components in the +7-foot heating and ventilation room. The inspectors also observed a tool belt with tools draped across HPSI Flow Control Valve CCW-225-A and loose scaffold boards and poles adjacent to HPSI Train A flow control valves.

Procedure UNT-007-006, "Housekeeping," Section 3.1 stated, in part, that housekeeping involves prevention of loose items from interacting with safety-related components during a seismic event. Section 3.9 defined a loose item as an item that was not restrained by means of bolting, welding, or other means. These items included, but are not limited to, portable carts, tool boxes, ladders, welding equipment, gas cylinders, storage cabinets, and other miscellaneous items. Section 5.4.1.4 stated, in part, that all departments should perform checks during the course of work activities in accordance with Attachment 6.6, "Housekeeping Requirements to Prevent Seismic Interactions With Operable Safety-Related Equipment." Attachment 6.6, Item 7, stated, in part, that standing ladders and similar items that upon falling could potentially damage operable safety-related equipment will be restrained or placed flat when unattended. Attachment 6.6, Item 10 stated, in part, that items weighing less than 20 pounds did not require restraints unless stored in a location where the falling impact can affect operation of operable safety-related equipment. The failure to properly store loose items that could potentially impact the function of a safety-related system is the first example of a violation of Technical Specification (TS) 6.8.1.a (382/9508-01).

The inspectors were concerned about the continued inattention to detail by plant personnel with respect to storage of loose items since numerous other concerns had been previously identified in NRC Inspection Reports 50-382/94-20, -94-21, -95-06, and -95-07. The inspectors discussed the licensee's corrective actions with the maintenance superintendent. The maintenance superintendent stated that training had been provided during shop meetings and that additional guidance would be provided to employees through a company newsletter. The inspectors concluded that the corrective actions implemented as a result of previous observations had been ineffective in minimizing the number of instances of poor storage practices.

2.3 Improperly Installed Locking Device

During a tour of Boric Acid Pump Room A, on August 25, 1995, the inspectors noted an improperly installed valve locking device for the Refueling Water Storage Pool-to-Charging Pumps Valve CVC-504. The inspectors determined that the valve was required to be, and was found, in the full-open position; however, the valve was not locked in the full-open position, as required by licensee procedures.

The inspectors notified the control room supervisor, who initiated a locked valve deviation sheet and dispatched an operator to properly secure the valve locking device. In addition, the control room supervisor initiated a CR to determine why personnel involved with implementation of a modification, which removed the support piece that the locking device was attached to, did not inform operations that the valve would need to be locked in another manner.

This issue will be tracked as an unresolved item pending the NRC's review of the locked valve program (382/9508-02).

The valve locking configuration was installed such that the cable locking device for the valve was secured to a nearby support that was attached to the wall. A modification was implemented that specified that a portion of the support be removed, which resulted in there being no location to attach the locking cable.

The inspectors discussed the removal of the support with the maintenance and construction department superintendents. As a result of these discussions, the licensee implemented interim corrective actions, which involved training for construction and modification group personnel and temporary contractors used for RFO 7. The construction superintendent stated that an analysis would be performed to identify the root causes and that the deficiency would be included in department standards to ensure that this item would be discussed during continuing training.

2.4 Safety Review Committee (SRC)

On September 22, 1995, the inspectors performed a review of portions of the Quality Assurance Program Manual. Section 4.5 stated that the SRC was responsible for maintaining oversight and assessing the effectiveness of the corrective action program. The SRC assessment shall include, as a minimum, a review of trend reports and significant adverse conditions.

Site Directive W2.303, Revision 4, "Safety Review Committee Charter," Section 6.1.3.4.b, established an audit subcommittee of the SRC to ensure, in part, that the required audits of the Quality Assurance Program Manual were being performed. Section 6.1.9.1.a also stated that the Quality Assurance Program Manual delineate the required frequency at which the audits were to be performed.

The typical meeting agenda provided in the procedure indicated that the QA manager will report on the audits performed since the last meeting. The discussion should include the status of the corrective action program, including adverse trends, significant CRs, corrective action timeliness concerns, and any audit findings open greater than 120 days.

The inspectors reviewed the agenda package, prepared on July 5, 1995, to support SRC Meeting 95-05. The package contained a corrective action program report that included: (1) equipment adverse trends not declared because of a program inadequacy, (2) a review of the plant trending program, (3) a discussion of each significant CR generated during the second quarter, (4) adverse trends identified and the status of previously identified adverse trends, and (5) the status of open CRs.

The inspectors reviewed the meeting minutes for SRC Meeting 95-05 and noted that the committee discussed the significance of the adverse trends and appeared to ask questions appropriate to the circumstances. The inspectors concluded that the SRC was maintaining an adequate oversight of the corrective action program and appeared to be implementing the requirements of Site Directive W2.303.

2.5 Reactor and Steam Plant Shutdown

On September 22, 1995, the inspectors observed the performance of a reactor and steam plant shutdown. The inspectors noted that the shift supervisor provided effective coordination of the tasks and stressed professional communications during the performance of various plant evolutions.

3 MAINTENANCE OBSERVATION (62703)

On September 6, 1995, the inspectors observed the performance of testing on the Emergency Feedwater Pump A Breaker 62-2 relay in accordance with Procedure ME-007-030, "G. E. Auxiliary Relay Model 12HGA17C." During testing of the relay dropout time in accordance with Step 8.2.2, the inspectors observed that the technicians were unable to obtain the elapsed dropout time during numerous unsuccessful attempts. Because they were unable to perform the test, the technicians requested the assistance of an individual more knowledgeable of the relay. The technicians were informed that the relay dropout time could not be obtained unless two pieces of test equipment (Doble and Multi-Amp SST) were installed in parallel.

The inspectors reviewed Procedure ME-007-030 and determined that there was no guidance to direct the technicians to use multiple pieces of equipment to test the relay. Procedure ME-007-030, Section 6.2, "Test Equipment," indicated that the relay test system, Multi-Amp SR76 or equivalent, was to be used to perform the procedure. The electrical supervisor stated that the Multi-Amp SR76 was no longer used and that technicians had been using the Doble as substitute M&TE.

Procedure MD-001-021, "M&TE Accountability Procedure," Section 5.7, stated that substitute M&TE may be used in place of specific M&TE designated in a procedure if the substitute has been shown to be equivalent by evaluation for the specific use. The electrical department M&TE equivalency document specified that Procedure ME-007-030 required the use of the Multi-Amp SST if using the Doble. The inspectors concluded that a contributing factor to the failure to use the correct M&TE was the reference in Procedure ME-007-030 to test equipment that was no longer used by the technicians. The inspectors also concluded that the failure to verify that appropriate M&TE was used could potentially affect the testing program and is a second example of a violation of TS 6.8.1.a (382/9508-01).

This example of the improper use of test equipment became evident only because the test equipment did not work. However, the potential exists for technicians to improperly calibrate instruments and components in the plant because the improper equivalent measuring and test equipment was used.

As discussed below, the licensee took effective actions to address this violation. For this reason, no response to this violation is required. In response to this problem, the licensee initiated a CR to document the inability of the technicians to calibrate the relay in accordance with the procedure. The licensee's corrective actions included a procedure change to specify that the Multi-Amp SST and Doble were needed to calibrate the Breaker 62-2 relay, a review of relay procedures to determine if test equipment needed to be specified, and training of technicians to ensure the departmental M&TE equivalency guide was referenced when substitute M&TE was utilized. The inspectors concluded that the licensee's corrective actions were timely and should be effective in preventing recurrence.

4 SURVEILLANCE OBSERVATION (61726)

On September 21, 1995, the inspectors observed the testing of Main Steam Safety Valve MS MVAAA-112B in accordance with Work Authorization (WA) 01138718. The WA directed that the testing be performed in accordance with Maintenance Manual Procedure MM-007-015, Revision 5, "Main Steam Safety Valve Test."

Procedure MM-007-015, Step 3.1.6, required that the test performer ensure that the pressure and ring gauges were calibrated within 7 days before the test and had not been used prior to performance of the testing. In addition. Step 3.1.7 required that the performer ensure calibration records furnished by a qualified testing laboratory, in accordance with ANSI 45.2-77, were part of the WA package.

The inspectors noted that the prerequisites and limitations sections of the procedure had been signed as completed by the test engineer even though Pressure Test Gauge MMPT 319.018, installed as the main steam line pressure gauge, had been calibrated 14 days (September 7) prior to the test. The inspectors also noted that the test line ring pressure gauge (MMPT 319.006) had been calibrated 15 days prior to the test (September 6). Additionally, the calibration reports for the gauges from the testing laboratory were not included in the WA package. The inspectors concluded that the failure to ensure prerequisites were complete prior to the commencement of testing is a third example of a violation of TS 6.8.1.a (382/9508-01).

The inspectors identified to the test engineer that the prerequisites were not properly met. The test engineer stated that the gauges would be calibrated prior to continuing the testing and that the appropriate documentation would be included in the package.

Following this discussion, the test engineer initiated CR 95-0817 to document that, in preparation for testing of the main steam safety valves, the test gauges had not been calibrated in accordance with the procedural requirements. During a subsequent review of the circumstances by the inspectors, regarding the issue of the engineer signing that the prerequisites were completed, it was determined that there was a miscommunication between the engineer and the lead mechanic. It appeared that each thought the other had verified the gauge calibration status. The inspectors concluded that signing the prerequisites as having been completed was an unintentional oversight and a lack of attention to details.

After the gauges were replaced, the licensee commenced testing of the valves. The inspectors noted that, on one occasion during the testing, the local equipment operator utilized three-way communications with the control room and a misunderstood pressure reading had been corrected. The inspectors concluded that the communications techniques used by the equipment operator during testing of the main steam safety valves were very good.

Step 8.3.6 required the performers to record the line pressure indicated on the gauge and compare with crossover line pressure reading in the control room. Step 8.3.7 stated that, if the line pressure on the gauge is not within ±15 psig of control room reading, then request engineering input for action to be taken. The test engineer and another engineer present discussed the results and decided that the steam generator pressure was closer to the test gauge pressure than the crossover line and decided to use the steam generator pressure indication for the test.

The inspectors questioned the in-field decision to utilize a pressure indication other than that designated by the procedure. In a meeting the following day, the test engineer's supervisor informed the inspectors that the engineer's actions had been determined to be appropriate. The inspectors concluded that the test engineer's decision to use steam generator pressure instead of crossover line pressure was appropriate.

5 ONSITE ENGINEERING (37551)

5.1 Lifting of SDCHX Relief Valves

On September 5, 1995, during the performance of Procedure OP-903-118, "Primary Auxiliaries Quarterly IST Valves Test," Section 7.3, stroke time testing of CCW Train A Valves CCW-200A and -727 (A to AB supply and return valves) resulted in the SDCHX Train A relief valve lifting. In addition, during stroke time testing of CCW Train B Valves CCW-200B and -563 (B to AB supply and return valves), the SDCHX Train B and CCW HX relief valves lifted. The lifting of the relief valves resulted in a continued level decrease in the CCW surge tank and an increase in waste tank level of approximately 2,000 gallons. Operations personnel reseated the relief valves by temporarily isolating and depressurizing the affected portions of the CCW system.

The inspectors reviewed CR 93-209 to determine what corrective actions had been initiated to prevent lifting of SDCHX relief valves during surveillance testing. The licensee determined the root cause to be an inadequate procedure, which did not account for hydraulic surges caused by isolating the nonsafety-related header from Trains A or B. A contributing cause was the small margin between the relief valve set pressure of 125 psig and system pressure of 95 psig.

The initial corrective actions, dated April 8, 1994, which was 5 months after initiation of the CR, included recommendations for operations to revise procedures to ensure the SDCHX reliefs did not lift during testing and design engineering to make recommendations to system engineering concerning the design and application of the relief valves. The QA department reviewed and approved the recommendations on June 1, 1994, and considered the corrective actions acceptable, with a stipulation that they would be verified at a later date.

The inspectors expressed a concern to the QA corrective action supervisor that QA approved corrective actions as acceptable that were not specifically delineated on the CR. The QA supervisor stated that the review and approval of CRs with recommendations for departments to develop corrective actions was a past poor practice and that the current QA practice would not approve CRs without specific corrective actions.

On July 19, 1994, which was 8 months from the initiation of the CR, QA acknowledged that the operations department determined that the original proposed corrective actions to revise procedures had been canceled. On July 20, 1994, the QA department acknowledged that engineering had decided to upgrade the system pressure from 125 psig to 150 psig and increase the set pressure of the relief valves.

On October 18, 1994, the CR indicated that Root Cause Investigation (RCI) 93-012 required the corrective actions for CR 93-209 to include the preparation of a WA by design engineering to increase the CCW system pressure and for operations to review surveillance procedures for enhancements that could minimize hydraulic transients.

On December 7, 1994, which was 13 months after the initiation of the CR, operations completed a review of surveillance procedures and concluded that the only feasible change that could be made was a change to the level setpoint at which the dry cooling towers isolate and that this change was currently scheduled for completion during RFO 7. The inspectors noted that 13 months had elapsed between the initiation of CR 93-209 and completion of the surveillance procedure review, which was considered to be untimely.

On December 21, 1994, design engineering personnel updated CR 93-209 to indicate that a review had been completed, which concluded that the CCW system pressure could be increased from 125 to 150 psig and that the WA had been developed and forwarded to maintenance to perform field work. The inspectors questioned engineering personnel to determine why the field work had not been completed and were informed that the new relief valves had not been received by the supplier and that the modification should be completed during RFO 7.

The inspectors reviewed RCI 93-012, which discussed the lifting of SDCHX Train B relief valves. The inspectors noted that RCI 93-012 was performed in response to CR 93-209 and that RCI 93-012 only addressed the operation of CCW Train B. RCI 93-012 indicated that no operability concern existed because, during accident conditions, no pressure surge was expected since the net effect of isolating the CCW nonsafety-related header would be offset by the increase in CCW flow through SDCHX B. The increase in flow through the SDCHX results from opening the CCW SDCHX Train B outlet isolation valve on a safety injection actuation signal (SIAS). Engineering determined that the flow increase and resultant head decrease would combine to lower the system pressure sufficiently to cause the SDCHX B relief valve to reseat. Engineering based this conclusion on the results of a special test performed on February 14, 1994, in which the CCW system was aligned to obtain full accident flow through the CCW AB pump discharge check valve (CCW-123AB). During the test, the CCW system was brought to full system flow and when the nonsafety-related portion of CCW was isolated, the SDCHX relief valves did not lift.

The inspectors reviewed the CCW system operating description and determined that the CCW SDCHX Train B outlet isolation valve receives an open signal on an SIAS, but the CCW SDCHX Train A outlet isolation valve remains closed on an SIAS and opens on a containment spray actuation signal. The inspectors questioned engineering to determine why lifting of the CCW Train A relief valve had not been included in RCI 93-012. Engineering stated that they had evaluated the operation of CCW Train A during their review but did not include the results in the report. Engineering also stated that the evaluation would have been more complete had it documented the results of the CCW Train A evaluation. Based on discussions with engineering, the inspectors concluded that the CCW Train A would remain functional under accident conditions since the increased flow rate through the emergency diesel generator and containment fan coolers would lower the pressure of the system and increase the total flow. Additionally, the automatic fill of the CCW surge tank from the CCW make-up pump (600 gpm) was sufficient to compensate for the capacity of the relief valves (11 gpm each) if they were to open.

The inspectors discussed the evolution, which occurred on September 5, 1995, with the shift supervisor and determined that there had been multiple occurrences of SDCHX relief valves lifting during performance of surveillance testing before and after the initiation of CR 93-209. The inspectors questioned two shift supervisors to determine why CRs had not been initiated for each occurrence and were informed that they had been requested not to initiate additional CRs for similar instances of the SDCHX reliefs lifting because CR 93-209 had been developed to correct the problem.

The inspectors discussed the initiation of repetitive CRs with the licensee on September 19 and 20, 1995, and determined that management expected initiation of a CR for each occurrence and that the QA department would administratively close the item to an existing CR to ensure that trend analysis for repeat occurrences was performed. Based on the discussions with the shift supervisors, the inspectors concluded that management's expectations for initiating CRs for repeat occurrences had not been satisfied. The inspectors discussed the initiation of CRs with the QA group and were informed that CR 95-705 had already been initiated on August 23, 1995, to review timely initiation of CRs.

During the plan-of-the-day meeting following the lifting of the relief valves, the plant manager indicated that the repeated lifting of the relief valves during testing over the extended period was unacceptable and directed the QA department to initiate a significant CR concerning untimely closure of CRs.

5.2 Water Hammer Event During Testing of ACCW and CCW Isolation Valves

On September 5, 1995, during stroke-time testing of CCW and ACCW Valves CC-301A(B), CC-322A(B), ACC-122A(B), and ACC-139A(B), in accordance with Procedure OP-903-118, "Primary Auxiliaries Quarterly IST Valves Test," Section 7.3, Change 4, a water hammer event occurred in the CCW/ACCW piping for Essential Chiller A, when transferring from the wet to the dry cooling tower. The initial walkdown of the CCW/ACCW systems by operations personnel did not identify any damage. However, the walkdown performed by engineering identified that a fire seal had been degraded and a fire impairment was initiated.

The inspectors discussed the water hammer event with the shift supervisor and determined that CR 95-0567 had been initiated, on July 4, 1995, to document that a potential existed for a water hammer event when placing the CCW cooling mode switch from the wet to dry tower position during surveillance testing. Prior to Change 4 of Procedure OP-903-118, the ACCW pump was running when switching the CCW cooling mode switch from the wet to the dry tower. With the ACCW pump running, the shared portion of piping between the CCW and ACCW systems remained pressurized. However, with the ACCW pump running, cross-contamination between the CCW and ACCW systems occurred during surveillance testing, which resulted in either tritiated water entering the ACCW system or chlorides entering the CCW system. Because of past cross-contamination problems, the chemistry department requested that a change to the surveillance procedure be made.

CR 95-0567 indicated that the root cause for the potential water hammer event was an inadequate procedure, in that Change 4 to Procedure OP-903-118 established conditions that led to a void in the CCW piping when Valve ACC-139A(B) (header return from essential chiller isolation) was opened. Opening Valve ACC-139A(B) drained the water from system piping between the +64-foot 6-inch level to the wet cooling towers located at the +22-foot level.

The corrective actions for CR 95-0567 recommended a revision to Procedure OP-903-118 to establish a test condition that would not cause a water hammer. However, implementation of a revision to the procedure did not occur prior to the performance of the surveillance on September 5, 1995. Because of the untimely revision, the operations crew performed the surveillance in accordance with a deficient procedure that was known to create a water hammer event.

The inspectors discussed the lack of the development of a procedure revision following the July 1995 water hammer with personnel in the operations procedures group and operations planning and scheduling group. Based on the discussions, the inspectors determined that interim corrective actions were not implemented to inform the operating shifts of the potential for a water hammer event during the surveillance test. Additionally, recommended procedure revisions were not routinely discussed between the procedures and planning groups. In response to the water hammer event that occurred on September 5, 1995, the operations procedure group initiated interim corrective actions to notify the operations planning and scheduling group of procedure revisions in progress. The planning group then identified the tasks associated with the procedure and did not schedule the activity until a revision was issued. Procedure OP-903-118 was revised to provide instructions for properly conducting the test.

The inspectors noted that CR 95-0567 did not identify, as a contributing factor, any discrepancies noted during the development and technical review of Change 4 to Procedure OP-903-118. The inspectors noted that both operations and engineering personnel reviewing the procedure change identified the potential for a water hammer event on the supply header and modified the procedure to include shutting the ACCW pump discharge valve to prevent draining the system. However, the reviewers failed to consider the effect the procedure change would have on the return header. The inspectors concluded that the licensee's failure to determine why an adequate technical review was not performed was an example of a poor determination of all contributing factors.

Following the second water hammer event, the plant manager admonished the staff at the plan-of-the-day meeting held on September 5 and directed that QA initiate significant CR 95-0750, on September 6, 1995, to determine the root causes of untimely initiation of CRs, implementation of ineffective interim corrective actions, and untimely closure of CRs. This action taken by the plant manager was in response to the failure to implement prompt and effective corrective actions to prevent recurrence of the water hammer event, which occurred on September 5, 1995, in the ACCW and CCW systems.

The licensee's preliminary review of the corrective action program identified three root causes. First, the corrective action program did not specify the need for timely documentation of conditions adverse to quality and management had not developed expectations for performance and accountability. Second, the corrective action program did not contain a graded approach related to the significance of CRs. Third, the corrective action process did not prompt the responsible organization to take interim corrective actions.

The licensee's preliminary corrective actions included proposed revisions to Procedure UNT-006-011, "Condition Reports," communication of management expectations to personnel, modifying CR status reports to include a highlight of overdue items, and a notification to all organizations that open CRs be reviewed to determine if interim actions were necessary.

The inspectors concluded that the licensee had initiated apparent appropriate actions to address the identified deficiencies in the corrective action program. This issue involving the use of the CR program to identify deficiencies will be tracked as an inspection followup item pending review of the licensee's activities by the inspectors (382/9508-03).

6 PLANT SUPPORT ACTIVITIES (71750)

On September 11, 1995, the inspectors performed independent radiation surveys of radioactive material storage containers located within the protected area and assessed the adequacy of radiological postings during a routine tour of outside spaces. The radiological surveys and evaluation of postings were performed in response to an increase in the storage of radioactive materials due to staging activities for RFO 7. Based on the results of the radiation surveys, the inspectors concluded that the licensee had appropriately posted radiation areas and established effective temporary radiological boundaries located within the protected area.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

R. G. Azzarello, Director, Design Engineering
R. F. Burski, Director, Nuclear Safety
G. L. Fey, Corrective Action Supervisor, Quality Assurance
T. J. Gaudet, Supervisor, Licensing
J. B. Houghtaling, Technical Services Manager
D. R. Keuter, General Manager, Plant Operations
D. C. Matheny, Operations Superintendent
W. H. Pendergras, Shift Supervisor, Licensing

1.2 NRC Personnel

K. M. Kennedy, Senior Resident Inspector, Arkansas Nuclear One

The above personnel attended the exit meeting. In addition, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on October 3, 1995. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee expressed a position that the violation involving the use of the gauges that were not calibrated in accordance with the procedure (see Section 4) had minimal safety significance and should be a noncited violation. The licensee was informed that the violation was cited because it represented an instance of personnel involved in testing safety-related equipment not following a procedure.

The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.