UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

APR 25 1995

Entergy Operations, Inc.
ATTN: John R. McGaha, Vice President -
Operations, River Bend Station
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: LICENSEE SELF-ASSESSMENTS RELATED TO SAFETY ISSUES INSPECTIONS

I am enclosing for your information a copy of two inspection reports. The reports document the results of NRC's review of the self-assessment performed by Omaha Public Power District (OPPD) at its Fort Calhoun Station, as well as NRC's special inspection of a significant safety issue identified by the OPPD self-assessment. The self-assessment was performed by OPPD using NRC guidance contained in Temporary Instruction 2515/118, "Service Water System Operational Performance Inspection." This self-assessment was performed by OPPD as an alternative to NRC performing a major team inspection using the same guidance.

NRC used Inspection Procedure 40501, "Licensee Self-Assessments Related to Safety Issues Inspections," to evaluate the adequacy of the self-assessment (in progress) and to review the self-assessment results. I am also enclosing for your information a copy of that inspection procedure. This was the first use of Inspection Procedure 40501 in Region IV and I am encouraged by the results.

The performance of licensee self-assessments as an alternative to a major NRC team inspection, as described in Inspection Procedure 40501, is a voluntary licensee activity limited to those licensees who meet the criteria described in the procedure.

This information is forwarded for your consideration. I would be pleased to discuss any questions you may have concerning this matter.

Sincerely,

Thomas P. Gwynn, Director
Division of Reactor Safety

Enclosures: As stated

Docket: 50-458
License: NPF-47

cc w/enclosures: (see next page)

JE51

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bcc to DMB (IE51)

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Project Engineer (DRP/D)
Senior Resident Inspector (Grand Gulf)
Branch Chief (DRP/TSS)
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

ENCLOSURE 1

MAR 31 1995

Omaha Public Power District
ATTN: T. L. Patterson, Division Manager
Nuclear Operations
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Hwy. 75 - North of Fort Calhoun
Fort Calhoun, Nebraska 68023-0399

SUBJECT: NRC INSPECTION REPORT 50-285/94-04

This refers to the inspection conducted by Elmo E. Collins and others of this office on October 31 through December 16, 1994. The inspection included a review of activities authorized for your Fort Calhoun Station facility. At the conclusion of the inspection, the findings were discussed with you and those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress. The results of this inspection are documented on page one, of the enclosed report.

We were pleased by the thorough, comprehensive self-assessment of the service water systems by your staff at Fort Calhoun Station. This was the first use of NRC Inspection Manual Chapter 40501, Licensee Self-Assessments Related to Safety Issues Inspections, to complete a major Service Water System Operational Performance Inspection in Region IV. Your self-critical assessment identified a number of deficiencies that were pending action at the conclusion of the inspection. Of particular note was the finding by your staff, discussed in NRC Special Inspection Report 50-285/94-24, regarding the inoperability of the control room air conditioning units. Overall, we believe that the results of this self-assessment have enhanced the safety of the Fort Calhoun Station and reflected positively on safety performance at the plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

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Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Thomas P. Gwynn, Director
Division of Reactor Safety

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report
50-285/94-04 w/Attachment

cc w/enclosure:
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-285/94-04

License: DPR-40

Licensee: Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Hwy. 75 - North of Fort Calhoun
Fort Calhoun, Nebraska

Facility Name: Fort Calhoun Station

Inspection At: Blair, Nebraska

Inspection Conducted: October 31 through December 16, 1994

Inspectors: Elmo E. Collins, Region IV Team Leader
Greg Werner, Region IV Reactor Engineer
Michael Shlyamberg, Contractor

Approved: Thomas J. Westerman 3-30-95
Thomas Westerman, Chief, Engineering Branch Date

Inspection Summary

Areas Inspected: Routine, announced inspection of the licensee's service water systems self-assessment.

Results:

- The licensee's self-assessment team had the necessary experience to conduct an effective assessment.

- The licensee performed, overall, a thorough and comprehensive self-assessment, addressing the required areas of Temporary Instruction 2515/118. The licensee concluded that the service water systems at Fort Calhoun Station were in good condition, and that the design requirements were met operationally.

- One safety significant item was identified by the licensee, which was reviewed in NRC Inspection Report 94-24.

- NRC inspectors concluded that the service water systems at Fort Calhoun Station were operable at the time of the inspection.

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- At the time of the inspection, corrective actions to address most items identified were not formulated or implemented; consequently, the inspectors could not assess the effectiveness of the corrective actions.

<u>Attachments:</u>

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 INTRODUCTION

NRC Inspection Manual, Inspection Procedure 40501, "Licensee Self-Assessments Related to Safety Issues Inspections," provides guidance on the NRC pilot program to evaluate a licensee's self-assessment effort as an alternative to an extensive NRC safety-issues team inspection. Under this program, the NRC inspection will be conducted in two phases: 1) an in-process inspection in which the NRC will evaluate the capability of the licensee's team and the depth of review by monitoring the conduct of the licensee's in-process assessment, and 2) a final inspection in which the NRC will perform a technical inspection of the licensee's completed self-assessment when the licensee issues its final report.

At Fort Calhoun Station (FCS), the licensee performed their self-assessment of the service water systems from October 24, 1994 through November 11, 1994. The NRC performed the in-process inspection from October 31, 1994, through November 4, 1994, and the final inspection from December 12-16, 1994. The licensee issued their Service Water System Operational Performance Self-Assessment Report on December 7, 1994. Section 2 of this report discusses the in-process inspection, and Sections 4-7 discuss the final inspection.

2 IN-PROCESS REVIEW (Temporary Instruction 2515/118)

From October 31 through November 4, 1994, the inspectors monitored the implementation of the self-assessment, evaluated the capability of the licensee's self-assessment team, and evaluated the scope of the licensee's planned effort. The inspectors found that the licensee's Service Water System Operational Performance Inspection Assessment Plan adequately covered the scope of Temporary Instruction 2515/118, "Service Water System Operational Performance Inspection." The inspectors concluded, on the basis of the experience and background of the self-assessment team, that the team had the necessary capability to perform the self-assessment.

The inspectors observed, to the extent possible, the implementation of the self-assessment, including the self-assessment team de-briefings, management de-briefings and interviews, and reviewed the list of questions that were being pursued by the self-assessment team. The inspectors found that the progress and depth of review were good. The licensee's self-assessment team exhibited a questioning attitude.

Initially, because of the way the licensee classified issues identified by their self-assessment team, all questions that were being pursued by the Omaha Public Power District (OPPD) response team were classified as Priority 3, the lowest level. Licensee management found that this method of classifying issues did not establish significance. During the conduct of the self-assessment, the licensee began classifying issues as significant, potentially significant, and not significant. The inspectors found that the later method of classification of issues better communicated the potential significance of items that were being pursued.

3 SYSTEM DESCRIPTIONS (Temporary Instruction 2515/118)

At Fort Calhoun Station (FCS), the following systems were used to cool the safety-related loads: (1) raw water (RW) and (2) closed cooling water (CCW) systems.

3.1 Raw Water System

The RW system is a once-through system that supplied cooling water to the safety-related CCW heat exchangers and provided backup water supply (also referred to as direct cooling) to the following safety-related loads: shutdown cooling heat exchangers, containment cooling coils, the seal and lubrication cooling for the engineered safeguards pumps, and the control room air conditioning units. These loads were normally cooled by the CCW system. The backup water supply was isolated by locked closed air operated valves (interface valves). The system consisted of common piping network with redundant components (e.g. all RW pumps were connected to a common header) instead of the "two independent trains". There were four RW pumps and four CCW heat exchangers. The minimum required number of pumps and heat exchangers was dependent on the RW temperature and other limiting conditions.

The ultimate heat sink for the RW system is the Missouri River. The intake structure consisted of three bays and was designed to minimize the occurrence and the consequences of a barge impact.

3.2 Closed Cooling Water

The CCW system is a closed loop system that provided cooling water to the safety-related heat exchangers described in the RW system description and to nonsafety-related loads. The majority of the nonsafety-related loads were not automatically isolated during accident conditions due to the loss of the nonsafety-related instrument air supply. However, these loads were connected to the CCW system by seismically qualified piping. Similarly to the RW system, the CCW system was also comprised of the common piping network instead of the "two independent trains". The CCW system relied on the RW system to provide a backup cooling to safety-related loads. The CCW system was equipped with a surge tank, which was located on the same level as the CCW pumps and was pressurized by nitrogen to 25 psig. In addition to providing the expansion and contraction capability, this tank also provided the static head on the CCW pumps to assure adequate $NPSH_A$. Two out of three CCW pumps were required to provide the safety-related flow in the event of an accident.

4 DESIGN REVIEW (Temporary Instruction 2515/118)

The inspectors reviewed the RW and CCW system design bases, design assumptions, calculations, analyses, boundary conditions, and models against licensing commitments and regulatory requirements. The inspectors also assessed the single failure impact on the ability of the CCW system to perform its required safety function and the capability of the systems to meet the thermal and hydraulic performance specifications. The inspectors' review of the above items was comprised of the examination of the licensee's self-assessment findings and independent inspection.

4.1 Impact of the Maximum Safeguards on CCW system

The licensee identified a condition which had the potential for rendering both control room air conditioning units inoperable. This condition was documented in Licensee Event Report 94-010. The NRC review of this item is documented in Inspection Report 285/94-24.

4.2 RW/CCW Interface Valves

The inspectors found that the licensee performed an in-depth and critical evaluation of the issues related to the RW/CCW interface valves. Specifically, the licensee identified that not all interface valves had been tested (stroked open and closed). The inspectors considered that this evaluation showed a questioning attitude and was a strength.

For design basis accident mitigation, FCS was licensed, considering that alternative cooling must be provided in the event of failure of the CCW system (including pipe rupture) during long-term cooling. This function was questionable because the CCW isolation valves for the shutdown heat exchangers (Valves HCV-480, -484, -481 and -485) were not designed to provide an isolation function, introducing the possibility that RW would leak through the isolation valves into the CCW system, and then out the break. Omaha Public Power District Memorandum PED-FC-94-1173, dated September 28, 1994, addressed the potential flooding concern if direct RW cooling was required due to the CCW pressure boundary failure. The inspectors reviewed this memorandum and found that the memorandum did not consider the break location. The proposed actions relied on the installation of a patch in the auxiliary building.

Engineering Evaluation EA-FC-91-014

The original design requirements for all of the interface valves was to provide the capability for the remote manual opening and to assure that they will fail in the open position in the event of the loss of instrument air. These requirements were modified when these valves were locked closed (Licensee Event Report 90-025). This design change led to a loss of the ability to provide the backup cooling for the engineered safeguards pumps' seals and coolers since the valves would not be accessible during accident conditions. An Engineering Evaluation EA-FC-91-014 was performed to justify that one-time operation of the emergency core cooling systems pumps without cooling under the design basis accident conditions was acceptable.

The engineering evaluation assumed that pump critical components were in a new or nearly new condition. While the inspectors found that the components appeared to be new or nearly new, the licensee had not implemented additional testing or controls to assure the basis of the engineering evaluation was maintained.

The sump temperature profile used in the evaluation was derived from the containment pressure/temperature analysis. The assumptions were to maximize the energy retained in containment volume and minimize the energy transferred to the sump. The assumptions, therefore, did not maximize sump temperature. The inspectors questioned the licensee regarding the amount of cooling necessary for the pumps and the licensee estimated that sump temperature could be about 20°F higher. The licensee considered that the short duration increase of the peak value would not impact pump performance.

4.3 Review of Calculations

The self-assessment review consisted of an in-depth and critical evaluation of the design calculations. Several weaknesses were identified by the licensee with Calculation FC06273, Revision 0, "RW Flows to CCW Heat Exchangers On Raw Water Chosen Pump Performance." The inspectors considered this evaluation to be very good.

The inspectors performed additional independent review and observed similar issues. The Technical Specification requirement that only one RW pump was required when river temperature was less that 60°F was based on the ability to deliver 6,000 gpm of river water to the CCW heat exchangers. This was based on the ability of one pump to deliver up to 6,660 gpm and closure of the air-operated pump discharge valves to prevent the backflow through the other pump discharge check valves. Factors that could negatively affect this were that the current design calculation predicted flows below 6,000 gpm and that the strainer backwash and check valve back flow were not taken into account. The inspectors considered that the calculation issues discussed above did not require prompt resolution due to the RW temperature being significantly below the design basis value of 90°F. However, addressing of these weaknesses would be necessary to provide assurance of the satisfactory operation of the system at limiting conditions. At the time of the inspection, the licensee was formulating corrective actions to address calculational weaknesses.

5 GENERIC LETTER 89-13 (Temporary Instruction 2515/118)

The inspectors reviewed how the recommended actions of Generic Letter 89-13 were implemented at FCS. The basis for this review were the results of the self-assessment documented in the self-assessment report and a limited independent inspection. The results of this review are provided below.

5.1 Recommended Action I

The major emphasis of this action item was on the biological fouling. The recommendations of Generic Letter 89-13 related to the biofouling appear to be implemented at FCS.

5.2 Recommended Action II

Action II of Generic Letter 89-13 requested that licensees implement a program to periodically verify the heat transfer capability of safety-related heat exchangers cooled by the RW system. The test program should consist of an initial test program and a periodic retest program.

Inspectors found that the FCS test program consisted of both inspecting/cleaning and performance testing of safety-related heat exchangers. The performance testing of the CCW heat exchangers was done as required by the Generic Letter 89-13.

Since all of the heat exchangers were included in the inspection program, the inspectors concluded that the licensee's response to Action II was adequate, and the Generic Letter 89-13 intent was met. However, as documented by the licensee's self-assessment, the performance portion of this program exhibited inconclusive trending results. Also, as documented by the licensee's self-assessment, neither test results nor acceptance criteria fully accounted for the instrument and modeling uncertainties. At the time of the in-process inspection, the licensee was formulating corrective actions to address these weaknesses.

5.3 Recommended Action III

Action III recommended the establishment of routine inspection and maintenance programs in order to ensure that silting will not degrade the safety-related functions of the RW system. Section 6.2.3 of this report discusses the licensee's identification of silt.

5.4 Recommended Action IV

Action IV of Generic Letter 89-13 requested that licensees confirm that the RW and CCW systems will perform their intended function in accordance with the licensing basis for the plant. This confirmation should include a review of the ability to perform required safety functions in the event of a single active component failure.

The inspectors found that the licensee identified weaknesses associated with the consideration of single failures of the RW and CCW systems. The limiting single failures were not identified for the maximum CCW return temperature (see Section 4.1). Also, the existing design calculation did not necessarily provide conservative minimum and maximum operating ranges for the RW and CCW systems.

While the inspectors concluded that the RW and CCW system were operable at the time of the inspection, resolution of these weaknesses would be necessary to assure the ability of the RW and CCW to fulfill their safety-related functions during limiting conditions. At the time of the inspection, the licensee was formulating corrective actions to address these issues.

6 MAINTENANCE AND SURVEILLANCE (Temporary Instruction 2515/118)

6.1 Overview

The inspectors reviewed the licensee's self-assessment report and identified areas of potential weaknesses in the review for microbiological-induced corrosion (MIC); resolution of sanding issues; and operation of plant components by maintenance personnel.

6.2 Maintenance

6.2.1 Maintenance Effectiveness Assessment

The inspectors conducted an independent walkdown of the CCW and RW systems to determine the material conditions of these systems and then compared the results with the self-assessment team findings. The inspectors observed some minor deficiencies. Licensee personnel had also identified minor deficiencies; however, no safety significant deficiencies were found. The licensee was tracking these deficiencies.

In order to assess the adequacy of maintenance being performed, the licensee reviewed previous maintenance work packages, observed ongoing maintenance, reviewed maintenance history, interviewed workers and engineers, scrutinized deferred outage maintenance, and evaluated the effectiveness and scope of the preventive maintenance program.

During interviews and reviews of maintenance history, the licensee self-assessment team members identified repetitive maintenance issues on the RW pump impellers, RW pump shaft seal water, and the intake traveling screens. While observing a maintenance activity, a problem was identified with the manipulation of a valve by maintenance technicians inside a clearance boundary. The self-assessment team did not identify any problems with deferred outage maintenance or with the preventive maintenance program. The licensee indicated that a review of historical maintenance activities against preventive maintenance tasks revealed that the preventive maintenance items were effective. The inspectors independently reviewed the maintenance history and deferred outage maintenance and concluded that the licensee's self-assessment was thorough.

The RW system engineer stated that the repetitive maintenance on the pump impellers (pump lift clearance adjustments) was attributed to pump shaft stretch. Evaluations on different types of shaft material were being conducted. Also, the self-assessment team member stated that an engineering change notice was being processed to change out the pump impeller material. An independent review of the engineering change was not done since it had not been completed.

The licensee planned Modification MR-FC-94-019 to upgrade the RW pump shaft seal water system to safety grade, and this was being credited as the solution for repetitive maintenance associated with the seal system. The system engineer stated that with the changes in system design, current repetitive failures of the seal water system should be eliminated.

Overall, maintenance activities observed by the self-assessment team were determined to be completed in an appropriate manner. However, one exception was noted during the replacement of a protective boot on a piston actuator for CCW Valve HCV-400C (CCW return from containment air cooling and filtering unit coils). The team member identified that the maintenance technicians had worked outside the work scope by manipulating the valve without appropriate work instructions. Incident Report 940381 was written to document this observation. The incident report was still open at the end of this inspection. Portions of two maintenance activities were observed by the inspectors and no deficiencies were identified. For the portion of the work activities observed, the work documents contained the necessary instructions and post-work testing requirements.

6.2.2 Microbiologically Induced Corrosion

The RW system engineer stated that only two through-wall failures (RW Valves RW-130 and RW-169 vent nipples) had occurred in the RW system since 1991. Incident Report 940165, "Root Cause and Generic Implications Analysis Report," described the failure on RW-130 as "localized pitting due to the existence of stagnant, oxygenated conditions." The system engineer also stated that the failed section was sent to three laboratories for analysis. No evidence of MIC was identified. Additionally, the system engineer stated whenever the RW system piping was opened for maintenance, he performed visual inspections of interior piping, and no evidence of MIC had been observed.

The RW interface piping and valves contain ideal conditions for MIC; however, no failures in this configuration of piping have been identified. Recent ultrasonic measurements taken at the request of the system engineer on RW interface piping elbows to the shutdown cooling heat exchanges indicated no abnormal loss of wall thickness.

The inspector also interviewed the circulating water system engineer. He stated that all through-wall pipe failures have been either attributed to erosion or to exterior corrosion attack. In addition, the inspectors requested a list of all through-wall piping failures attributed to MIC. Licensing personnel stated that no failures have been attributed to MIC.

Based upon the information provided, the inspectors concluded that the licensee had adequately evaluated and addressed pipe through-wall failures, and the inspectors did not identify evidence of MIC related failures.

6.2.3 Sanding and Silting Problems

The licensee self-assessment team performed a historical review of RW system problems caused by sand. The licensee identified that, as early as 1972, the RW pumps would "sand in" and trip upon starting. The recommendations provided at that time were not implemented until 1994. The self-assessment team identified that the resolution of this sanding problem was untimely.

Reviews of documentation by the licensee's self-assessment team led them to the conclusion that sand/silt obstruction of RW/CCW backup lines was not probable. The licensee's conclusion was based on the piping configuration consisting of RW/CCW backup lines tapping off the top of the main header, a rise of over 50 feet before the first component supplied, and no flow in the header. No previous sand/silt obstruction incidents were documented with the exception of finding sand in RW/CCW backup discharge lines for the shutdown cooling heat exchangers (Generic Letter 89-13 inspection). The sand was determined to have accumulated since original plant construction.

On December 2, 1994, while replacing the soft seat liner for RW Backup Valve HCV-401E, "RW/CCW Interface Valve to Containment Air Cooler VA-1B," a soft sand/silt plug was discovered in the supply piping. Incident Report 940413 was initiated to document the condition. On December 20, another soft sand plug was discovered while reworking Valve HCV-403E, "RW/CCW Interface Valve to Containment Air Cooler VA-8B." The inspectors performed a physical walkdown of the RW/CCW backup piping and confirmed the configuration used to support the licensee's previous conclusions that sanding was not probable in the RW backup lines actually existed in the plant.

The RW backup supply to the emergency core cooling systems pumps tapped off the header just downstream of where sand/silt was found in the containment cooler supply. The low pressure safety injection pumps and the shutdown cooling heat exchangers with cooling provided by RW enabled the plant to be cooled to cold shutdown in the event of a fire in the CCW pump room (Appendix R requirement). The inspectors questioned the flow path for RW backup to the components necessary for an Appendix R shutdown. The RW system engineer and his supervisor stated that the RW backup flow path to the low pressure safety injection pumps and shutdown cooling heat exchangers were operable.

The basis for operability of the shutdown cooling heat exchangers was based on an inspection of the RW/CCW interface valves conducted during the 1990 refueling outage as part of Generic Letter 89-13 inspections. Sand was found in the discharge piping when HCV-482B and HCV-483B were removed. No sand was found in the supply RW backup piping. The sand was not hardened and since high flow rates would be encountered, the sand would be washed away.

The operability of the RW backup for the low pressure safety injection pumps was based upon subjective evidence, since no disassembly of any of the RW backup valves to the emergency core cooling systems pumps had been conducted. System engineering personnel indicated that if sand were present at the valves, the valves may not cycle or would fail to seat completely, and this had not been observed. In addition, licensee personnel stated that if a soft sand plug existed, it would be pushed through the system by water pressure. The system engineer had also been involved with flushing the emergency core cooling systems pumps oil cooling heat exchangers and no evidence of sand had been observed in the heat exchangers.

During the exit meeting, licensee senior management personnel reiterated the operability of the RW backup flow path necessary for Appendix R cold shutdown.

6.3 Surveillance and Testing

6.3.1 Surveillance and Testing Effectiveness Assessment

The licensee self-assessment members performed an extensive review of surveillance testing, inservice testing, and preoperational testing and compared this to the design basis documents and technical specification requirements. Interviews with the personnel responsible for these areas were conducted. The licensee's self-assessment team members also observed surveillance testing activities.

In order to evaluate the licensee self-assessment effort, the inspectors observed interviews conducted by the licensee team members and found them to be probing. The inspectors interviewed the team members responsible for the surveillance and testing area and concluded that the licensee was addressing the areas outlined in their inspection plan and that the scope was appropriate to fulfill the related requirements of Temporary Instruction 2515/118.

During the in-process inspection, the inspectors reviewed Modification MR-FC-88-61, "Air Check Valve for RW Discharge Valves," to determine if changes made to the system had been incorporated into the surveillance or inservice testing program. Surveillance Test IC-ST-IA-3003, "Raw Water Instrument Air Accumulator Check Valve Operability," did test the check valves. The licensee had identified a deficiency in the testing methodology since it did not take into account instrument inaccuracies which could effect the volume of air required to close the valves. The licensee was tracking this item.

The licensee self-assessment team identified that during October 1994, air sparging of the RW/CCW heat exchangers was not being done on an equal basis. Reviews of control room logs for the month of October determined that the daily shift surveillance requirement for RW/CCW heat exchanger air sparging was being completed as required.

Selected items identified in the licensee report and Q&A list were reviewed. All of the items identified were to be corrected and were entered into either the corrective action process or administrative tracking system; however, no assessment as to the completeness or effectiveness of the corrective actions was performed since the licensee had not completed the corrective actions.

6.3.2 RW/CCW Interface Valve Testing

The licensee's self-assessment team identified the failure to test the RW/CCW interface valves. Significant CAR 94-220 documented this condition. During the second week of the NRC inspection, the inspectors reviewed the details surrounding the failure to test the RW/CCW interface valves.

During the 1992 outage, the licensee declared all RW/CCW interface valves inoperable and did not test the valves. This decision appeared to have been based on information contained within the USAR which specified no requirements for RW backup to safety-related components during any USAR described accident. In August 1993, additional information indicated the RW backup function was necessary under certain conditions, and all RW backup valves, with the exception of four valves, were tested. Three RW backup valves to the containment air coolers and one RW backup to Shutdown Cooling Heat Exchanger B were not tested. Raw water backup to the containment air coolers was determined not to be necessary for any accident conditions.

The RW backup to the shutdown cooling heat exchangers was a requirement for an Appendix R cold shutdown. This was identified as a concern to the licensee. American Society of Mechanical Engineers, Section XI, "Inservice Testing of Valves in Light-Water Reactor Power Plants," allowed valves not to be tested if inoperable. However, Section 3.2 also stated that inservice testing shall be done on valves required to be operable to fulfill their required function.

Licensee management stated the requirements of Appendix R were satisfied by the testing completed on the RW backup valves to Shutdown Heat Exchanger A and the availability of Heat Exchanger A. This satisfied the questions concerning the operability of components necessary to complete an Appendix R cold shutdown.

6.4 Conclusion

The inspectors concluded the self-assessment of maintenance, surveillance, and testing associated with the service water system operational inspection was thorough. A thorough review of programs, procedures, implementation, and personnel performance allowed licensee personnel to appropriately evaluate the performance of the raw water and component cooling water systems.

7 LICENSEE SELF-ASSESSMENT CAPABILITY (Temporary Instruction 2515/118)

Overall, the inspectors concluded that the licensee demonstrated a good self-assessment capability. The licensee identified weaknesses in calculations, the absence of basis for the maximum CCW design temperature, untimely corrective actions for RW/CCW interface valve testing, weaknesses in the previously performed safety system functional reviews, weaknesses in the single failure analysis for the CCW system, and untimely corrective actions on sanding problems in the RW system.

At the time of the NRC final inspection, the licensee was in the process of formulating the corrective actions to address the performance problems identified by their self-assessment, consequently the inspectors were not able to evaluate the scope or effectiveness of the licensee's corrective actions. The inspectors did not identify any immediate operability issues.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

R. Andrews, Division Manager, Nuclear Services
J. Chase, Manager, Fort Calhoun Station
G. Cook, Supervisor, Station Licensing
J. Gasper, Manager, Training
G. Gates, Vice President, Nuclear
R. Jaworski, Manager, Station Engineering
L. Kusek, Manager, Nuclear Safety Review Group
D. Lakin, Nuclear Review Safety Group Specialist
E. Matzke, Station Licensing Engineer
W. Orr, Manager, Quality Assurance and Quality Control
T. Patterson, Division Manager, Nuclear Operations
R. Phelps, Acting Manager, Production Engineering
J. Skiles, Acting Manager, Production Engineering Department
M. Tesar, Manager, Corrective Actions
D. Trausch, Manager, Nuclear Licensing
B. Van Sant, Engineer, Production Engineering Department

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

2 EXIT MEETING

An exit meeting was conducted on December 16, 1994. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee acknowledged the items presented by the inspectors. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

ENCLOSURE 2

February 15, 1995

EA 94-267

Omaha Public Power District
ATTN: T. L. Patterson, Division Manager
Nuclear Operations
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Hwy. 75 - North of Fort Calhoun
Fort Calhoun, Nebraska 68023-0399

SUBJECT: NOTICE OF VIOLATION AND ENFORCEMENT DISCRETION
(NRC Inspection Report 50-285/94-24)

This refers to the special inspection conducted by Mr. R. Mullikin and the inspectors identified in the enclosed report, of this office, on November 14 through December 27, 1994. The inspection included a review of activities authorized for your Fort Calhoun Station facility. At the conclusion of the inspection, the findings were discussed with you and those members of your staff identified in the enclosed report.

This special inspection was conducted to review the immediate safety concerns and corrective actions associated with your staff's identification of a significant design deficiency with the control room air conditioning units. On November 14, 1994, Omaha Public Power District determined that both control room air conditioning units would have been rendered inoperable during certain design basis accident conditions. Specifically, increases in component cooling water (CCW) temperature following a main steam line break in the containment building or a large break loss of coolant accident would result in the control room air conditioning units' compressors shutting down and the possible release of the Freon to the control room. With the loss of control room air conditioning, certain engineered safeguards features (ESF) equipment may have been rendered inoperable, as described in the report. This was identified by your staff and documented in a letter dated November 29 to the Region IV Regional Administrator and in Licensee Event Report 94-10, "Potential Accident Scenario Involving a Loss of Control Room Air Conditioners." The special inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress and provided a determination whether activities authorized by the license were conducted safely and in accordance with NRC requirements.

Based on the results of this inspection, certain licensed activities appeared to be in violation of NRC requirements as described in the enclosed Notice of Violation (Notice) and details to this report. The first violation involved a failure to assure that the control room air conditioning unit design modification correctly translated the design basis specifications for assuring system operability during certain design basis accidents. As a result, the control room air conditioning units that were purchased and installed in 1988

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were not capable of operating within the component cooling water maximum temperatures following a postulated main steam line break inside the containment or a large break loss of coolant accident. The second violation involved a failure to implement the established procedures for the documentation and evaluation of the design deficiency.

These violations are being cited in the aggregate as a Severity Level III problem because of the high potential for rendering the control room air conditioning units inoperable during certain design basis accidents and the probability that this in turn would have rendered certain ESF equipment inoperable. The subsequent failure of plant personnel to appropriately document and evaluate this design deficiency resulted in the plant operating at 100 percent power for an additional month outside its design basis. Normally, a civil penalty is considered for a Severity Level III problem. However, I have been authorized after consultation with the Director, Office of Enforcement, not to propose a civil penalty for this problem. This determination was based on our finding that this problem and your actions in addressing it satisfied the provisions of 10 CFR Part 2, Appendix C, section VII.B(4) for old design issues. Specifically, (1) the basic violation existed since the air conditioning units were purchased in 1988 and is appropriately characterized as an old design issue; (2) the problem was discovered by the licensee during a voluntary, formal review and was promptly reported once the ramifications of the problem were recognized; (3) the problem was not likely to be identified through routine inspection, surveillance, or QA activities, and (4) once the problem was recognized, the licensee developed appropriate short-term compensatory measures and comprehensive long-term corrective action. Although section VII.B(4) also provides that the NRC may refrain from issuing a Notice of Violation for old design issues under certain circumstances, we are issuing a Notice of Violation in this case because as indicated in the NOV, your staff did not respond appropriately to indications of this problem in October 1994 that should have resulted in a prompt operability determination and would have lead to earlier identification and correction.

A management meeting has been scheduled for March 2, 1995, in the Region IV Public Meeting Room at 1 p.m. The purpose of this meeting is to discuss your staff's actions to improve the corrective action process and the concerns identified in its implementation following the identification of the control room air conditioning design deficiency in October 1994. This meeting is open to public observation.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice of Violation (Notice) when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. After reviewing your response to this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96.511.

Sincerely,

org signed by

L. J. Callan
Regional Administrator

Docket: 50-285
License: DPR-40

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 50-285/94-24

cc w/enclosures:
LeBoeuf, Lamb, Greene & MacRae
ATTN: Mr. Michael F. McBride
1875 Connecticut Avenue, N.W.
Suite 1200
Washington, D.C. 20009-5728

Washington County Board of Supervisors
ATTN: Jack Jensen, Chairman
Blair, Nebraska 68008

Combustion Engineering, Inc.
ATTN: Charles B. Brinkman, Manager
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Nebraska Department of Health
ATTN: Harold Borchert, Director
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Lincoln, Nebraska 68509-5007

Nebraska Department of Health
ATTN: Dr. Mark B. Horton, M.S.P.H.
Director
P.O. Box 950070
Lincoln, Nebraska 68509-5007

Fort Calhoun Station
ATTN: James W. Chase, Manager
P.O. Box 399
Fort Calhoun, Nebraska 68023

NOTICE OF VIOLATION

Omaha Public Power District
Fort Calhoun Station

Docket: 50-285
License: DPR-40
EA 94-267

During an NRC inspection conducted November 14 through December 27, 1994, two violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violations are listed below:

A. Criterion III, Appendix B of 10 CFR Part 50, states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, on November 14, 1994, the licensee determined that a 1988 control room air conditioning unit design modification had not correctly translated the design basis specifications for assuring system operability during certain design basis accidents. As a result, the control room air conditioning units that were purchased and installed were not capable of operating within the component cooling water maximum temperatures following a postulated main steam line break inside the containment or a large break loss of coolant accident. This condition has existed since 1988. (01013)

B. Criterion V, Appendix B of 10 CFR Part 50, states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings . . . . and shall be accomplished in accordance with these instructions, procedures, or drawings.

The Fort Calhoun Quality Assurance Plan, Revision 3, Section 2.1, paragraph 4.2.1, states, in part, that activities affecting quality shall be prescribed by documented instructions or procedures and shall be accomplished in accordance with these instructions or procedures.

Standing Order SO-R-4, Station Incident Reports, Revision 4[illegible], Section 2.4.5 requires an incident report for violations of established system design bases (Updated Safety Analysis Report [USAR] or Design Bases Document).

Nuclear Operations Division Procedure NOD-QP-31, Revision 8, Section 6.16, requires that nonroutine events and conditions which may require operability and reportability determinations be documented by an incident report per Standing Order SO-R-4. Section 7.1 states that determinations of operability and reportability will be performed using the guidance in this procedure and will be implemented through Standing Order SO-R-4, Station Incident Reports.

Contrary to the above, on October 5, 1994, an incident report was not initiated, after the licensee was notified, for a violation of the established control room air conditioner design basis or to perform the

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required operability and reportability determinations. The required incident report was subsequently initiated on November 15, 1994, following the licensee's determination that the control room air conditioning units would be rendered inoperable during a large break loss of coolant accident or a main steam line break inside the containment. (01023)

This is a Severity Level III problem (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, the Omaha Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Dated at Arlington, Texas,
this 15th day of February 1995

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/94-24

Operating License: DPR-40

Licensee: Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 399, Hwy. 75 - North of Fort Calhoun
Fort Calhoun, Nebraska

Facility Name: Fort Calhoun Station

Inspection At: Blair, Nebraska

Inspec8tion Conducted: November 14 through December 27, 1994

Inspectors: R. Mullikin, Senior Resident Inspector
E. Collins, Team Leader
L. Smith, Senior Resident Inspector, Arkansas Nuclear One
G. Werner, Reactor Inspector
M. Schlyamberg, NRC Contractor

Approved: ______________________________ __________
William D. Johnson, Chief, Project Branch A Date

Inspection Summary:

Areas Inspected: Special inspection to review the circumstances surrounding the licensee's identification that component cooling water (CCW) temperature could =8exceed the maximum design value during accident conditions and that the control room air conditioning (A/C) units would likely be rendered inoperable.

Results:

Operations

- The operators appropriately assessed the operability determination provided by engineering and initiated the Technical Specification required plant shutdown. The Notification of Unusual Event was declared as required by the emergency plan emergency action level for the required plant shutdown (Section 2.1.1).

Engineering

- The licensee identified a significant design basis deficiency which would result in both control room A/C units being rendered inoperable during a main steam line break (MSLB) inside the containment building or a large break loss of coolant accident (LOCA). The failure would result from a CCW temperature in excess of the A/C units' design operating limit. This design deficiency was identified as a result of their initiative to perform a Service Water System Operational Performance Inspection (Section 2).

- A violation was identified for the failure to assure that the control room air conditioning unit design modification correctly translated the design basis specifications for assuring system operability during certain design basis accidents. As a result, the control room air conditioning units that were purchased and installed were not capable of operating within the component cooling water maximum temperatures following a postulated main steam line break inside the containment or a large break loss of coolant accident. This condition has existed since 1988 (Section 2.1.8).

- A second violation was identified for a failure to implement the established procedures for the documentation and evaluation of the design deficiency. Plant personnel responsible for addressing the control room A/C units' operability concern did not notify operations when the concern was first raised and did not independently ensure that a prompt operability determination was performed. This resulted in the plant operating at 100 percent power for an additional month outside its design basis (Section 3).

- A comprehensive safety analysis for operability was developed based on an excellent coordination of operations and engineering inputs to provide compensatory measures to ensure a control room A/C unit would remain operable following a DBA. The analysis was well supported by their plant specific probablistic risk assessment (Section 2).

Management Oversight

- Management was proactive in implementing the Service Water Operational Performance Inspection. Appropriate resources were dedicated to the preparation and conduct of the activity (Section 3).

- Management was not effective in assuring that the operability concerns were promptly reviewed and acted upon in accordance with the established corrective action process, once they were notified on October 17, 1994 (Section 3).

Summary of Inspection Findings:

- Violation 285/9424-01 was opened (Section 2.1.8).

- Violation 285/9424-02 was opened (Section 3).

DETAILS

1 INTRODUCTION

On November 14, 1994, the licensee notified the NRC that CCW temperatures could exceed the maximum limit described in the Updated Safety Analysis Report and the Design Basis Document. This report contains the results of two inspection efforts which assessed immediate safety concerns and the review of the actions prior to the licensee's determination of the problem.

2 IMMEDIATE SAFETY CONCERNS

On November 14, 1994, the inspectors assessed the immediate safety concerns and the licensee's actions to justify continued operation following the determination that CCW temperatures could exceed the maximum described limit following a DBA. The licensee determined that the higher CCW temperatures would render the control room A/C units inoperable.

2.1 Inoperable Control Room A/C Units

2.1.1 Description of Initial Event

On November 14, 1994, the licensee's engineering organization determined that the control room A/C units would be rendered inoperable during a large break LOCA or an MSLB inside containment. The loss of the A/C units would be caused by increased CCW temperature through the A/C units' water cooled condenser and waterside economizer which would result in the rupture discs on the Freon system actuating, emptying the system of Freon. The major increase in CCW temperature would result from the heat transfer from the containment cooling units following a DBA. After A/C failure, the control room temperature would continue to rise until the offsite power low signal (OPLS) relays, lockouts, and sequencers would become inoperable at 105°F room temperature, which would correspond to an internal cabinet temperature of 120°F. The OPLS is an engineered safety feature (ESF) which load sheds the safety-related 4160-volt busses, starts the emergency diesel generators, and energizes the busses following a loss of offsite power with a safety injection actuation signal present.

The licensee entered Technical Specification 2.15(3), which required that the reactor be placed in hot shutdown within 12 hours with the OPLS inoperable. Technical Specification 2.15(3) was entered at 1:49 p.m.. A Notification of Unusual Event was declared at 2:55 p.m. as required by the licensee's emergency plan for a plant shutdown required by the Technical Specifications. At 4:30 p.m., a reactor power decrease was initiated at a rate of 1 percent per hour. Concurrently, the licensee was reviewing the design deficiency and working toward mitigating its significance. Engineering and operations subsequently developed compensatory actions that could be taken to mitigate the significance of the design deficiency and would permit continued plant operations until the refueling outage. The Plant Review Committee met to consider the proposed alternatives to the shutdown. The Plant Review Committee approved the compensatory measures based on an engineering analysis and Operations Memorandum 94-07. These actions provided for placing one of the two control room A/C units (VA-46A) in "STOP" and tagging closed the CCW valves to the unit. This would preserve the one unit for control room cooling. The memorandum required shutting down the operating control room A/C unit after a plant trip. However, if no DBA had occurred and CCW temperature remained below 90°F and was not rising, then this unit could be restarted. If a DBA had occurred, then the following actions were to be taken by operators:

- Maximize raw water (RW) flow through available heat exchangers.

- Have only two CCW pumps operating to reduce CCW flow rate and enhance heat transfer.

- Reduce the containment cooler heat load on the CCW system when containment pressure is reduced to an operator judged acceptable level.

- Continue controlling CCW heat loads until CCW temperature is reduced to less than 90°F. Then the protected control room A/C unit can be placed into service to maintain control room temperature.

- In the event that CCW temperature cannot be controlled, operators would then enter the appropriate abnormal operating procedure (AOP) for either controlling temperature or establishing a raw water backup. The appropriate AOP's would be AOP-11, "Loss of Component Cooling Water," and AOP-13, "Loss of Control Room Air Conditioning."

The licensee also imposed more restrictive equipment operability requirements than specified in the Technical Specifications for the control room ventilation, raw water pumps, and CCW heat exchangers. The following actions were defined:

- The loss of the operating control room A/C unit (VA-46B) would require entering a 24-hour Limiting Condition for Operation (LCO). If the unit could not be repaired within 24 hours, then the licensee would enter Technical Specification 2.15(3) and be in hot shutdown within 12 hours. The Technical Specification did not have an action statement on either A/C Unit VA-46. However, Technical Specification 2.12.1 stated that, if the temperature in the control room cabinets exceeded 120°F, and could not be reduced below 120°F in 4 hours, then the reactor must be put into hot shutdown within 6 hours.

- An inoperable CCW/RW heat exchanger would require entering a 14-day LCO. The Technical Specification did not require any action for one inoperable heat exchanger.

- An inoperable RW pump would require entering a 24-hour shutdown LCO. The Technical Specification would require entering a 7-day shutdown LCO.

The operations memorandum was supported by an operability evaluation and a 10 CFR 50.59 review. The 10 CFR 50.59 review concluded that an unreviewed safety question existed without the actions defined in the operations memorandum.

At 7:28 p.m., the licensee stopped the power reduction at 97.5 percent. The Notification of Unusual Event was terminated at 7:50 p.m.

2.1.2 System Description

The CCW was a closed system of demineralized water which provided the cooling medium for many plant heat loads using three CCW pumps. The major heat loads following a recirculation actuation signal were the shutdown cooling heat exchangers (AC-4A and -4B), containment air cooling units (VA-7C and -7D), and containment air cooling and filtering units (VA-3A and -3B). CCW was cooled by RW through four RW/CCW heat exchangers using four RW pumps. During normal operation, one CCW pump, one RW pump, and one heat exchanger would be in service.

The control room was cooled by two redundant A/C units (VA-46A and -46B). Normally only one unit would be in operation. There were also two control room filtering units (VA-64A and -64B).

2.1.3 Description of Concern

The containment cooling units and the containment spray system provided the capability to reduce containment temperature and pressure after a large break LOCA or an MSLB inside containment. The containment cooling units were only required during an MSLB. A containment isolation actuation signal would have been received following either of these two accidents. The containment cooling and filtering units would have automatically started and CCW would have supplied these two units. In addition, both control room A/C units and fans would have automatically started, including the CCW supply to the A/C Unit VA-46 Freon condensers.

To evaluate the CCW temperature transient, the licensee performed Calculation FC-06304 and determined that within 3 minutes of a large break LOCA or MSLB, concurrent with a loss of instrument air, the CCW temperature would exceed 130°F for certain initial configurations of the RW and CCW systems. The compressors for the control room A/C Units VA-46A and -46B would trip at 106°F and, at a CCW temperature of 130°F, Freon pressure would actuate the system rupture disk, releasing the Freon into the control room complex. The licensee determined that the Freon would not pose a habitability concern. The loss of the Freon, however, would disable the A/C units. Without cooling, the control room temperature would increase within 30 minutes to the Technical Specification limit of 105°F. The licensee would enter AOP-13 to attempt to reduce the control room temperature. Some of the methods outlined in the procedure would be to shut down unnecessary control room heat loads and open the control room doors. However, opening the control room doors would have potentially created a radiological hazard during a LOCA.

The 105°F control room temperature would not have created a severe habitability concern by itself for up to 2 hours, but would have affected ESF equipment. The licensee determined that the first to be affected would have been the OPLS. The OPLS equipment would have been affected by an internal cabinet temperature of 120°F, which corresponds to a 105°F control room temperature. If a loss of offsite power occurred at the beginning of the accident, OPLS would perform its functions. However, if the loss of offsite power were to have occurred at 30 minutes after the loss of control room cooling, then the OPLS might not have performed its design function.

2.1.4 Engineering Calculations

The following licensee engineering calculations were used in the resolution of this concern:

- Calculation FC-06236, "Post-DBA Heat Load on CCW System as a Function of CCW Return Temperature"

  The objective of this calculation was to determine the maximum post-DBA heat load on the CCW system as a function of CCW return temperature, which is the temperature exiting the CCW/RW heat exchangers. This calculation showed how the maximum heat load on the CCW system changes if the CCW system heats up in a post-DBA situation.

- Calculation FC-06304, "RW and CCW Initial Temperature Rise for LOCA - Maximum Safeguards"

  The objective of this calculation was to estimate the RW and CCW temperatures during the early stages of a large break LOCA with full safeguards actuation.

- Calculation FC-06308, "CCW Return Temperature Maintainable at RAS with 50°F River Temperature"

  The objective of this calculation was to determine whether the RW system could support a CCW temperature of 90°F or less during the post-RAS period of a LOCA with river temperature of 50°F or less. The calculation considered one RW pump feeding four RW/CCW heat exchangers, which simulates one inoperable RW pump and a failure of an emergency diesel generator. In addition, the calculation assumed three RW pumps feeding three RW/CCW heat exchangers, which simulates the one RW pump inoperable and the failure of a RW isolation valve to open on one RW/CCW heat exchanger.

- Calculation FC-06311, "Control Room Heat Gain Without Air Conditioner VA-46A or -B Cooling"

The objective of this calculation was to determine the control room heatup rate between the present time and the spring 1995 refueling outage. The following assumptions were made:

- Both A/C Unit VA-46 compressors are shut off at time 0.
- One of the A/C Unit VA-46 fans is running at time 0 but is shut off at time 20 minutes and not restored.
- The A/C Unit VA-64 charcoal filter has a 9-KW heating element that is energized the entire time.
- The control room temperature is 80°F, with 50 percent relative humidity at time 0.
- The outside air temperature is 60°F maximum with 50 percent relative humidity.

2.1.5 Safety Analysis for Operability

On November 18, 1994, the licensee approved Safety Analysis for Operability (SAO) 94-02, "Control Room Air Conditioners VA-46A/B, ESF Lock-outs, Relays, and Sequencers." The SAO concluded, based upon engineering calculations, that the control room temperature would not exceed 105°F if outside ambient air temperature remained less than or equal to 60°F, with no control room A/C units in service. All control room equipment would remain operable under this scenario. Additional conditions were established in the event that outside air temperature were to exceed 60°F before the 1995 refueling outage, scheduled to begin on March 11. The licensee used the results of probabilistic risk assessment (PRA) models to conclude that control room equipment would be operable for outside temperatures greater than 60°F when the duration of these temperatures was less than or equal to 250 cumulative hours. This assumed that the operating VA-46 unit fan would be stopped at or before 20 minutes after a DBA since the fans also provided a heat load. In addition, if the Missouri River temperature was less than or equal to 50°F the licensee concluded that the CCW temperature could be reduced sufficiently to allow starting of the protected VA-46 unit.

The licensee also evaluated whether elevated CCW temperatures would affect other safety-related equipment cooled by CCW. The conclusion was that these pieces of equipment would be operable with river temperature less than 60°F, three RW pumps available, and three RW/CCW heat exchangers available.

The SAO provided operational restrictions during normal plant operation. These restrictions were:

- no more than 250 total hours that outside air temperatures reach 60°F. This would be logged and tracked in the control room;

- one RW pump may be inoperable indefinitely. One RW/CCW heat exchanger may be inoperable for 14 days. However, the RW pump and heat exchanger could not be inoperable at the same time;

- two RW pumps may be inoperable for 24 hours;

- RW flow must be maintained through three RW/CCW heat exchangers;

- river temperature must be less than or equal to 50°F;

- one A/C Unit VA-46 must be protected;

- the A/C Unit VA-46 in service may be inoperable for 24 hours as long as control room temperature was below 80°F; and

- control room temperature must be below 80°F.

The SAO stated that, if any of the above conditions were not met, Technical Specification 2.15(3) would be entered, which required a shutdown of the plant. In addition, the SAO also provided for actions to take if a large break LOCA or an MSLB inside containment were to occur. These actions involved stopping operating A/C Unit VA-46 at or before 20 minutes into the accident and, if outside air temperature exceeds 60°F, then restarting of protected A/C Unit VA-46 would be desired. In order to bring this unit into service, the following actions were defined:

- maximize both the number of RW pumps and RW/CCW heat exchangers in service;

- ensure a maximum of two operating CCW pumps in order to maximize heat transfer capability;

- when sufficient containment spray flow was verified, then stop all containment cooling and filtering units except for one filtering unit (VA-3A or -3B). The remaining filtering unit could be stopped if CCW temperature did not decrease to less than 90°F;

- start protected A/C Unit VA-46 if CCW temperature was below 90°F and decreasing; and

- if the control room temperature reached 100°F and CCW temperature had not decreased to less than 90°F then the operators should consider entering either AOP-11 or -13.

On November 29, the licensee submitted a letter to the NRC Region IV Regional Administrator detailing the operational requirements that they imposed on themselves.

2.1.6 Reportability Process

NOD-QP-31, "Operability and Reportability Determinations," is the procedure that provides overall guidance for determining operability and reportability of nonroutine events and conditions. Section 7.3.2 stated that an initial operability determination shall be made within 24 hours even though complete information may not be available. The licensee determined that both control room A/C units would be inoperable following a large LOCA or MSLB inside containment. In addition, since the A/C units are support equipment for ESF components, the OPLS was also declared inoperable. However, the operability determination stated that the plant could continue to operate with the recommendations previously described.

The inspectors questioned the licensee as to when an SAO would be written and approved since this is the licensee's document for justifying continued operation. Quality Procedure NOD-QP-22, "Preparation and Approval of a Safety Analysis for Operability (SAO)," states that the SAO is a comprehensive and documented evaluation of a safety or operability concern which established the bases for the continued safe operation of the plant. The procedure did not require that the SAO be approved within a certain time frame, but a prior revision of this procedure had established a time requirement of 48 hours.

2.1.7 Use of PRA

The licensee used PRA models to determine the risk involved by taking interim actions prior to the 1995 refueling outage. The models used were the following:

- Change in core damage probability versus numbers of hours that outside ambient air temperature is greater than 60°F. This resulted in a change of approximately 1.00E-06 for up to 250 cumulative hours of outside temperature greater than 60°F.

- Change in core damage probability versus time that outside ambient temperature is greater than 60°F with standby A/C Unit VA-46 available. This resulted in a change of approximately 5.00E-07 for up to 250 cumulative hours of outside temperature greater than 60°F.

- Same as above but with A/C Unit VA-46 unavailable. This resulted in a change of approximately 1.00E-05 for up to 250 cumulative hours of outside temperature greater than 60°F.

2.1.8 Technical Review

The inspectors reviewed the licensee's calculations and evaluation of the effects of CCW temperature exceeding the design value of 120°F. The inspectors found that the original CCW maximum design temperature apparently did not consider the consequences of the maximum heat rejection from containment during an accident and nonconservatively bounded the heat rejection from the CCW system through the RW/CCW heat exchangers. Also, the licensee did not establish the basis for the maximum design CCW temperature (120°F) during the design basis reconstitution program. This situation was further exacerbated by the use of 90°F as a maximum cooling (CCW) water temperature in the purchase specification for the replacement of the original A/C units (Modification MR-FC-81-51), which took place in 1988. This was not consistent with the design parameters of the CCW system.

During the service water self-assessment, the licensee identified weaknesses related to the single failure analysis which had previously been performed on the CCW system. The inspectors concluded that the lack of a thorough and rigorous single failure review factored in not identifying the maximum safeguards condition earlier.

The inspectors questioned the impact of this condition on the CCW pump net positive suction head ($NPSH_A$). The $NPSH_A$ was of particular concern, since it is inversely related to CCW temperature. The inspectors' concern about the $NPSH_A$ was compounded by the uncertainty of the ability to maintain nitrogen overpressure in the CCW surge tank. The nitrogen supply was not single failure proof and some of the pressure retaining components connected to the tank were not safety related. The licensee responded that the $NPSH_A$ would be adequate within the RW and air temperature limits specified in LER 94-010, "Potential Accident Scenario Involving Loss of Control Room Air Conditioners." The licensee indicated that a more comprehensive evaluation of the $NPSH_A$, including loss of the overpressure, as well as the other potential impacts on the CCW system, would be addressed during the detailed CCW transient analysis. The inspectors concluded that the most limiting impact of high CCW temperatures was the control room A/C failure and that there was no immediate safety concern with the continued operation of the plant. The inspectors noted that neither the Updated Safety Analysis Report nor Design Bases Document had considered the impact of the maximum safeguards equipment temperatures on the CCW system.

Criterion III, Appendix B of 10 CFR Part 50 requires that measures shall be established to assure that applicable regulatory requirements and the design basis for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. In addition, Criterion IV, requires that measures shall be established to assure that applicable regulatory requirements, the design basis, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services. The failure to appropriately translate the design basis for the CCW system into the control room A/C unit design modification specifications is a violation (298/9424-01). As a result, the control room A/C units that were purchased and installed in 1988 were not capable of operating within the CCW system design basis temperatures following a postulated main steam line break inside the containment or a large break loss of coolant accident.

3 PROBLEM IDENTIFICATION AND EVALUATION

The inspectors reviewed the sequence of events leading to the licensee's identification of the maximum safeguards equipment temperature concerns and the associated documentation and evaluation.

The inspectors found that, in the spring of 1994, the licensee assembled a "preinspection" team in anticipation of an NRC service water operational performance inspection team. In about April, it was decided that Fort Calhoun Station would perform a self-assessment in lieu of a team inspection and the NRC's inspection was postponed. As part of the preinspection team, the licensee questioned the maximum CCW design temperature of 120°F. No calculation could be found to support the number. In March 1994, the licensee had requested that the architect engineer evaluate the consequences of CCW temperature exceeding 120°F. The results of this evaluation were documented in Stone and Webster Calculation 16472.8030-PM-1. Revision 1 of this calculation, dated October 3 (available onsite about October 5) included information that the control room A/C units would trip at a cooling water temperature of approximately 102°F. The licensee continued to pursue the problem, but the incident report operability/reportability processes were not entered at this time.

On October 17, senior licensee management was briefed of a potential problem with CCW temperature. On October 26, the reportability procedure was entered with the drafting of a memorandum to plant operations. No incident report was generated.

Licensee engineers apparently focused on the postaccident CCW transient temperature and could not find any analysis indicating what the temperature would do. The licensee used iterative hand calculations and discovered that the temperature would rise very quickly and that the control room A/C unit Freon rupture discs would likely actuate. On November 14, the control room A/C units were declared inoperable.

Stone and Webster Calculation 16472.8030-PM-1, Revision 1, was onsite October 5 and indicated that the A/C units would trip at a temperature of approximately 102°F. This information indicated a design deficiency since the maximum design CCW temperature was 120°F. This design deficiency was not entered into the corrective action system and no prompt operability determination was performed. Inspectors reviewed Standing Order SO-R-4 and found that Section 2.4.5 stated that violations of established system design bases (Updated Safety Analysis Report or Design Bases Document) or externally imposed regulations are types of incidents requiring an incident report.

The inspectors also reviewed Nuclear Operations Division Procedure NOD-QP-31, Revision 8, and found that Section 6.7 assigned the responsibility for prompt operability and reportability determinations of events or conditions to the Shift Supervisor. Section 6.16 stated that all "NOD, PED, and NSD" personnel are responsible for identification to appropriate personnel of nonroutine events and conditions, including nonconformances, which may require operability and reportability determinations, by initiating an incident report per Standing Order R-4. Section 7.1 stated that determinations of operability and reportability will be performed using the guidance in this procedure and will be implemented through Standing Order R-4, "Station Incident Reports," and Standing Order R-11, "Notification of Significant Events." However, certain situations may meet prompt 10 CFR 50.72 reporting requirements or other regulations; for this reason, the shift supervisor must be informed of these events concurrent with or ahead of the incident report generation. Cognizant personnel shall provide information and recommendations necessary for the shift supervisor to determine prompt reportability.

The inspectors concluded that the design deficiency indicated by Calculation 16472.8030-PM-1, Revision 1, was a condition that violated established system design basis and warranted an Incident Report on October 5, under the requirements of Standing Order R-4. Also, Procedure NOD-QP-31 directed the initiation of an incident report for conditions which may require an operability determination. As a result of not initiating the incident report, the shift supervisor was not notified of the condition and a prompt operability determination was not performed for conditions where it was known or extremely likely that the control room A/C units would trip under accident conditions. In addition, potential problems were informally evaluated. An incident report was written on November 15 after the systems were declared inoperable. This is a violation (285/9424-02).

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

*#+R. Andrews, Division Manager, Nuclear Services
+B. Blome, Supervisor, Corporate Quality Assurance
+C. Boughter, Acting Manager, Station Engineering
*G. Cavanaugh, Licensing Engineer
#+J. Chase, Manager, Fort Calhoun Station
*#+G. Cook, Supervisor, Station Licensing
+R. Eurich, Supervisor, Design Engineering
*#J. Gasper, Manager, Training
*#W. Gates, Vice President, Nuclear
*#R. Jaworski, Manager, Station Engineering
*#+L. Kusek, Manager, Nuclear Safety Review Group
#D. Lakin, Nuclear Safety Review Group Specialist
*#W. Orr, Manager, Quality Assurance and Quality Control
*#+T. Patterson, Division Manager, Nuclear Operations
*#+R. Phelps, Acting Manager, Production Engineering
*M. Sandhoefner, Shift Supervisor
*+J. Sefick, Manager, Security Services
*#+J. Skiles, Acting Manager, Design Engineering
#M. Tesar. Manager, Corrective Actions
+J. Tesarek, Supervisor, Simulator Services
*J. Tills, Assistant Plant Manager, Operations
*#+D. Trausch, Manager, Nuclear Licensing and Industry Affairs
#B. Van Sant, Mechanical Engineer, Design Engineering

* Denotes personnel that attended the exit meeting on December 13, 1994.

# Denotes personnel that attended the exit meeting on December 16, 1994.

+ Denotes personnel that attended the exit meeting on December 27, 1994.

In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on December 27, 1994. During this meeting, the inspector reviewed the scope and findings of the report. The licensee acknowledged the inspection findings. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ENCLOSURE 3

NRC INSPECTION MANUAL

DRIL

INSPECTION PROCEDURE 40501

LICENSEE SELF-ASSESSMENTS RELATED TO SAFETY ISSUES INSPECTIONS

PROGRAM APPLICABILITY: 2515

SALP FUNCTIONAL AREA: ENGINEERING

40501-01 INSPECTION OBJECTIVE

To provide guidance on the pilot program to evaluate a licensee's self-assessment effort as an alternative to an extensive NRC safety issues team inspection.

40501-02 INSPECTION REQUIREMENTS

02.01 Regional management will determine if licensee's past performance is sufficient to warrant reduced NRC inspection.

02.02 Licensee's Proposed Self-Assessment

a. Organization. Evaluate the capability of the licensee's organization to manage the self-assessment, whether it is conducted by some element of the licensee organization or an independent organization.

b. Assessment Team. Ensure that the licensee's assessment team has the necessary credentials and experience to perform a technically creditable self-assessment.

c. Scope of Effort. Ensure that the scope and depth of the licensee's program are at least equivalent to those specified in the temporary instruction (TI) for the safety issues inspection, or that the licensee provides an acceptable basis for reducing scope or depth.

d. Timing. Ensure that the timing of the licensee's proposal and assessment is such that NRC can adequately adjust its planning for the safety issues inspection and can review the licensee's assessment planning and in-process implementation.

02.03 Implementation of the Licensee's Self-Assessment

a. In-Process Inspection. Evaluate the capability of the licensee's team and the depth of review by monitoring the conduct of the licensee's in-process assessment.

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b. <u>Final Inspection</u>. Perform a technical inspection of the licensee's completed self-assessment when the licensee issues its final report.

40501-03 INSPECTION GUIDANCE

General Guidance

Under this pilot program, the NRC will recognize a licensee's good performance and quality self-assessment by reducing the scope of NRC inspection. This reduced inspection applies to safety issues team inspections, which are resource intensive (require more than 15 inspector-days of onsite inspection). This procedure outlines the process for monitoring the licensee's self-assessment effort and for conducting a limited-scope, indepth inspection as an alternative to a full-scope NRC inspection as specified in the related TI.

The NRC has safety issues inspections such as Service Water System Operational Performance Inspection (TI 2515/118, Revision 1) or other required inspections such as Safety-Related Motor-Operated Valve Testing and Surveillance (TI 2515/109) which could be conducted at each plant. This procedure is independent of Inspection Manual Chapter (IMC) 2515, Section 06.03, which provides that the regional administrator may for good cause elect not to perform a generic safety issues team inspection.

The NRC acceptance process should be initiated by a licensee's formal submittal of its proposed self-assessment as a basis for reduced NRC inspection, including the schedule, scope, level of effort, and team qualifications. The region should formally respond to all licensee requests for reduced inspection. If reduced NRC inspection is conditionally approved, the letter should include any stipulations (e.g., revised self-assessment scope) on which the approval is conditioned. The letter should state that the option of permitting licensees to conduct a self-assessment in lieu of an NRC safety issues inspection is a pilot NRC program aimed at minimizing regulatory impact and utilizing NRC resources more efficiently. Copies of all correspondence to the licensee regarding reduced inspection effort will be sent to the Chief, Inspection and Regulatory Criteria Branch, NRR.

Notwithstanding the licensee's self-assessment, the region may elect to not reduce its normal safety issues inspection scope if there are indications that the licensee's effort was not implemented in accordance with the licensee's approved plan or was deficient for some other reason.

Any issues identified as a result of a licensee's self-assessment should be considered to be licensee identified, notwithstanding that licensee effort was initiated as a result of NRC selecting the area to inspect. However, any enforcement action involving an "old design issue" should be referred to the Office of Enforcement.

The requirements in the TI pertaining to reporting requirements, completion schedule, expiration, and technical contact should be followed. For MIPS and RITS purposes, the inspection effort involved in performing this IP is to be charged against the TI, and the TI closed with a status code of "C" when NRC inspection activities are completed.

Specific Guidance

03.01 Regional management should consider licensee performance during prior major safety issues inspections, relevant NRC inspections, periodic plant performance reviews, and SALP ratings. Licensees recognized by NRC senior management as good performers should be considered as automatic candidates. Requests from other licensees should be evaluated on a case-by-case basis. Licensees placed on the "watch list" would not be eligible for reduced NRC inspection.

03.02 Licensee's Proposed Self-Assessment

a. Organization. The NRC review should consider if the backlog of activities for the licensee's organization could preclude effective management of the self-assessment. The technical qualifications of the involved licensee personnel should be evaluated.

b. Assessment Team. The design reviewers should have hands-on design experience, such as that of a supervisory engineer in an architect-engineer organization. The reviewers in other than design areas should have appropriate technical and plant experience.

c. Scope of Effort. Deviations from the scope of the safety issues inspection will be treated on a case basis with appropriate justification. For example, it may be appropriate for the licensee's assessment not to include areas evaluated during recent NRC inspections.

d. Timing. Licensee self-assessments should be completed in a timeframe that is sufficient to allow the NRC to complete a final inspection of the licensee's effort before the expiration date of the TI. It is desirable that licensee proposals be made sufficiently in advance of both the implementation of the self-assessment and the planned NRC safety issues inspection (generally at least 90 days). This is to ensure that the region has sufficient time to evaluate the proposal (Section 02.02) and align its resources as appropriate to monitor in-process conduct of the licensee's self-assessment (Section 02.03.a). Although the intent is not to penalize a licensee who does not give the region sufficient lead time before the self-assessment, it is obviously more efficient for the licensee to resolve NRC comments on the inspection scope, depth, and so forth before initiating the self-assessment. In addition, it is impractical for NRC to be notified of a licensee request for reduced NRC inspection under this IP after the NRC inspection has proceeded substantially in the planning stage.

   To ensure timely self-assessment proposals, regions should provide advance notification of safety issues inspection by as much as 9 months, consistent with the latitude in IMC 0300, "Announced and Unannounced Inspections." This advance notification should advise the licensee of the self-assessment option and of the possibility of obtaining a copy of this inspection procedure if requested.

03.03 Implementation of the Licensee's Self-Assessment

a. In-Process Inspection. Generally, a regional team leader and/or senior inspector qualified for conducting the specific safety issues inspection should conduct the in-process inspection. Generally, this will require more than one individual because of the several disciplines involved, but not as many as required for the final inspection of the licensee's effort. The in-process inspection should last up to 5 days and be implemented when the self-assessment is approximately 50 percent complete. During this inspection, the scope and depth of the licensee's self-assessment, including the objectivity and independence of the self-assessment team should be evaluated. The evaluation should also include a review of the process for addressing operability concerns and the process for developing corrective actions and ensuring their appropriateness. If the licensee's organization primarily responsible for the areas of the self-assessment has demonstrated good performance (e.g., SALP Category 1) and the licensee's implementation of its self-assessment appears to be effective, NRC effort during the final inspection may be reduced.

b. Final Inspection. This inspection should be performed when the licensee has completed its self-assessment, specified corrective actions to be implemented, and issued a final summary report. The regional technical inspection of the completed self-assessment should be led by a regional team leader and include appropriate multi-discipline expertise, including contractor specialists. Wherever possible, personnel who participated in the in-process inspection (Section 03.03.a above) should be on the team. The team should parallel the disciplines for the normal safety issues inspection, with appropriate reductions made in areas where one inspector can cover several disciplines. The inspection scope and team size should be predicated on the TI, findings from other inspections, results from the NRC in-process inspection, site-specific characteristics, and the licensee's past performance.

The NRC inspection scope should include areas covered by the self-assessment to evaluate the completeness of the licensee's reviews, but minimal resources should be expended in areas where the licensee had adequately addressed significant findings. For areas within the scope of the TI reviewed during the self-assessment on a sampling basis, the NRC inspection should include items both reviewed and not reviewed during the licensee's self-assessment. For example, if the self-assessment did not include a review of the thermal performance of all service water system heat exchangers, the NRC inspectors would need to reach a conclusion on whether the self-assessment sample provided a reasonable basis to support a licensee conclusion regarding the performance of the heat exchangers not reviewed. The NRC scope should also include any significant areas of the TI not addressed by the self-assessment.

During the NRC inspection, the corrective actions proposed by the licensee for the more significant assessment findings as well as for generic findings and the licensee's handling of any operability concerns should be evaluated. It may be appropriate for the region to follow up on the licensee's implementation of the corrective actions for significant findings.

40501-04 RESOURCE ESTIMATE

The goal for the total NRC oversight effort should be no more than 25 percent of the normal preparation and inspection effort for the safety issues inspection. Inspection duration and scope and team size and composition will be determined by the regional office on a case-by-case basis. Normally, the 25 percent goal can be achieved by reducing both team size and inspection duration to half the normal level.

END