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

July 26, 2002

R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, Nebraska 68023-0550

SUBJECT: NRC INSPECTION REPORT 50-285/02-02

Dear Mr. Ridenoure:

On June 29, 2002, the NRC completed an inspection at your Fort Calhoun Station. The enclosed report documents the inspection findings which were discussed on July 2, 2002, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified a finding that involved nonconservative processes that could affect the availability of mitigating systems during plant heatup conditions. This issue was evaluated under the risk significance determination process as having very low safety significance (Green).

The NRC has increased security requirements at the Fort Calhoun Station in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/ by Jeffrey A. Clark acting for

Claude E. Johnson, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report
50-285/02-02

cc w/enclosure:
Mark T. Frans, Manager
Nuclear Licensing
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, Nebraska 68023-0550

James W. Chase, Division Manager
Nuclear Assessments
Fort Calhoun Station
P.O. Box 550
Fort Calhoun, Nebraska 68023-0550

David J. Bannister, Manager - Fort Calhoun Station
Omaha Public Power District
Fort Calhoun Station FC-1-1 Plant
P.O. Box 550
Fort Calhoun, Nebraska 68023-0550

James R. Curtiss
Winston & Strawn
1400 L. Street, N.W.
Washington, D.C. 20005-3502

Chairman
Washington County Board of Supervisors
Washington County Courthouse
P.O. Box 466
Blair, Nebraska 68008

Sue Semerena, Section Administrator
Nebraska Health and Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P.O. Box 95007
Lincoln, Nebraska 68509-5007

Daniel K. McGhee
Bureau of Radiological Health
Iowa Department of Public Health
401 SW 7th Street, Suite D
Des Moines, Iowa 50309

Technical Services Branch Chief
FEMA Region VII
2323 Grand Boulevard, Suite 900
Kansas City, Missouri 64108-2670

Electronic distribution by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(EEC)**
Senior Resident Inspector **(JGK)**
Branch Chief, DRP/C **(CEJ1)**
Senior Project Engineer, DRP/C **(WCW)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Jim Isom, Pilot Plant Program **(JAI)**
RidsNrrDipmLipb
Scott Morris **(SAM1)**
FCS Site Secretary **(NJC)**
Dale Thatcher **(DFT)**
W. A. Maier, RSLO **(WAM)**

RIV:SRI:DRP/C	RI:DRP/C	C:DRS/EMB	C:DRS/PSB	C:DRP/C
JGKramer:nc	LMWilloughby	CSMarschall	GMGood	CE Johnson
T - JAClark	***T - JAClark***	***/RA/***	***JBNicholas for***	***JAClark for***
7/25/02	7/25/02	7/25/02	7/25/02	7/26/02

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-285
License:	DPR-40
Report:	50-285/02-02
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm., P.O. Box 399, Hwy. 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	March 31 through June 29, 2002
Inspectors:	J. Kramer, Senior Resident Inspector L. Willoughby, Resident Inspector W. Walker, Senior Project Engineer B. Baca, Health Physicist, L. Ellershaw, Senior Reactor Inspector
Accompanying Personnel:	I. Barnes, Consultant
Approved By:	Claude E. Johnson, Chief, Project Branch C

SUMMARY OF FINDINGS

Fort Calhoun Station
NRC Inspection Report 50-285/02-02

IR 05000285-02-02; 03/31-06/29/2002; Omaha Public Power District, Fort Calhoun Station, Integrated Resident & Regional Report; Refueling and Outage Activities

The inspection was conducted by resident and regional inspectors. This inspection identified one Green finding. The significance of the issues is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process.”

Cornerstone: Mitigating Systems

- Green. The licensee exercised a nonconservative decision making process when controlling foreign materials in containment and eliminating the potential for blocking the emergency core cooling system suction strainers. As a result, approximately 20 55-gallon drums with paper taped onto the lid on the same elevation as the sumps and several hundred other pieces of tape remained in containment during initial plant heatup following a refueling outage.

  This finding was of very low safety significance because the containment emergency sumps remained available (Section 20).

Report Details

Summary of Plant Status

The unit began the inspection period at approximately 98 percent power. On May 4, 2002, the unit was shutdown for Refueling Outage 20 and entered cold shutdown. Upon completion of the outage activities, the unit entered hot shutdown on May 30. On June 1, the reactor was made critical and the unit was placed on-line the following day. On June 7, the unit achieved approximately 100 power and remained at the power level for the remainder of the inspection period.

1. **REACTOR SAFETY**
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial equipment walkdowns. The inspectors checked portions of the equipment to identify any discrepancies between the existing and proper alignment as determined by system piping and instrumentation drawings or plant procedures:

- Shutdown Cooling Purification System using Procedure OI-SC-5, “Shutdown Cooling Purification,” Revision 16, on May 8, 2002.

- Emergency Diesel Generator 2 locally at the engine and switch alignment in the control room on June 12, 2002.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed selected plant areas to determine if the licensee appropriately implemented the fire protection program. Specifically, the licensee adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capabilities, and maintained passive fire protection features in good material condition. The inspectors checked the following areas:

- Diesel Generator 2 room on June 12, 2002

- Electrical switch gear room on June 17, 2002

- East battery room on June 17, 2002

- Air compressor and auxiliary feedwater pumps room on June 27, 2002

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other than Steam Generator Tube Inspections

Performance of Nondestructive Examination (NDE) Activities

The Fort Calhoun Station inservice inspection program is committed to the ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 1989 Edition, no Addenda for the third 10-year interval. The third 10-year interval will end following the next refueling outage, currently scheduled for the fall of 2003.

a. Inspection Scope

The inspectors observed portions of the following inservice inspection examinations:

System	Component/Weld Identification	Examination Method
Safety Injection	Elbow-to-Pipe, 4-CH-12, Weld 08	Ultrasonic Examination
Component Cooling Water	Pipe-to-Pipe, 10-AC-2004, Weld 25	Ultrasonic Examination
Component Cooling Water	Elbow-to-Pipe, 10-AC-2003, Weld 23	Ultrasonic Examination
Main Steam	Pipe-to-Elbow, 28-MS, Weld 04	Ultrasonic Examination

During the performance of each examination, the inspectors verified that the correct NDE procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation and equipment were properly calibrated. The inspectors reviewed the NDE certification packages of the observed contractor personnel and verified that they had been properly certified in accordance with ASME Code requirements. The inspectors also verified that indications revealed by the examinations were compared against the ASME Code-specified acceptance standards and appropriately dispositioned.

The inspectors reviewed the licensee’s NDE records for certain examinations that were performed during the current outage to verify that either required or committed NDE activities were performed in accordance with ASME Code requirements, and indications and defects, if present, were appropriately dispositioned. These included ultrasonic examinations of the weld overlays in reactor vessel head Heated Junction Thermocouple Housings 7 and 11 and liquid penetrant examinations of the three butt welds in peripheral Control Element Drive Mechanism Housing 37.

In addition, the inspectors also observed eddy current examinations of the “J” welds in four of the eight control element drive mechanism upper seal housings that the licensee committed to perform.

The inspectors determined, with respect to reactor vessel head control element drive mechanism nozzles, that the licensee had completed the ASME Code required number of examinations specified for the third 10-year interval.

Finally, the inspectors determined that the licensee, as allowed by the ASME Code, deferred the volumetric examinations required for Category BA full penetration reactor vessel nozzle welds until the next refueling outage.

b. Findings

No findings of significance were identified.

.2 ASME Code Repair and Replacement Activities

a. Inspection Scope

The inspectors reviewed Work Order Packages 76794 and 91590 for ASME Code Section XI repair and replacement of Safety Injection Check Valve SI-194 and Hand Control Valve HCV-331 and reviewed Safety Injection Check Valve SI-197, respectively.

The inspectors observed the fit up and welding of these components and verified that these activities were in accordance with the specified welding procedure specifications. The welding procedure specifications were verified, by review of the procedure qualification records, to be appropriately qualified. All welding material used on these work orders was properly identified and controlled. The inspectors also observed the subsequent fiber optic visual examination of the root passes and radiographic setups and shoots of the completed welds. The inspectors reviewed all of the applicable radiographic film and reader sheets. All indications were appropriately dispositioned, and identified defects were removed, repaired, and reexamined.

The inspectors observed and verified that controls were in place to assure that welding materials were properly stored, identified, certified, and distributed.

b. Findings

No findings of significance were identified

.3 Steam Generator Tube 2R15 Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee’s in-situ screening criteria to verify that the criteria were in accordance with industry guidelines. The estimated size and number of tube wear flaws identified up to the date of the inspection were compared to the operational assessment predictions from the previous outage. The inspectors also reviewed the eddy current examination scope and expansion criteria to determine if the Technical Specifications, industry guidelines, and commitments to the NRC were being met.

The inspectors reviewed the areas of potential degradation (based on site-specific and industry experience) to verify that such areas were being inspected. The eddy current probes and equipment were reviewed to ascertain if they were properly qualified for the expected types of tube degradation.

The inspectors observed the collection and analysis of eddy current data by licensee personnel to verify that the eddy current procedure was being followed and that indications were being appropriately dispositioned.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalifications (71111.11)

a. Inspection Scope

The inspectors observed licensed operator requalification training activities, including the licensed operators' performance and the evaluators' critique. The inspectors compared performance in the simulator on June 17, 2002, with performance observed in the control room during this inspection period.

The inspectors placed an emphasis on high-risk licensed operator actions, operator activities associated with the emergency plan, and previous lessons learned items. These items were evaluated to ensure that operator performance was consistent with protection of the reactor core during postulated accidents.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

During the inspection period, the inspectors reviewed licensee implementation of the Maintenance Rule. The inspectors verified structure and component scoping, characterization, safety significance, performance criteria, and the appropriateness of goals and corrective actions. The inspectors compared the licensee's implementation of the Maintenance Rule to the requirements outlined in 10 CFR 50.65. The inspectors reviewed the following components:

- Pressurizer Spray Control Valves PCV-103-1 and PCV-103-2

- Inconel Pressurizer Nozzles
- Fire Suppression System Deluge Valves FP-526, FP-798, FP-749, FP-513, FP-708, FP-210, and FP-211

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's risk assessments for equipment outages as a result of planned and emergent maintenance to evaluate the licensee's effectiveness in assessing risk for planned and emergent activities. The inspectors compared the licensee's risk assessment and risk management activities against requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors also discussed the planned and emergent work activities with planning and maintenance personnel. The inspectors reviewed and observed emergent work on the following:

- Heated Junction Thermocouples venting while draining to hot midloop on May 7, 2002
- Safety Injection Tank Level Instrumentation calibration while draining to hot midloop on May 7, 2002
- Electrical Bus Maintenance Activities planned to occur while in hot midloop on May 10, 2002
- Reactor Coolant Pump seal replacement while in cold midloop on May 28, 2002.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of operability evaluations to verify that they provided adequate justification that the equipment could still meet its Technical Specification, Updated Safety Analysis Report, and design bases requirements. The following evaluations were reviewed:

- Diesel generator operability with fuel oil particulate contamination results being out of specification high (Condition Report 200200792)
- Diesel Generator operability after a pipe used in the sampling process fell into Fuel Oil Tank 10 (Condition Report 200200838)
- Auxiliary Spray Valve HCV-249 operability with the close stroke time greater than the acceptable value (Condition Report 200207735)
- Shutdown Cooling System Piping operability with Snubber SIS-134 being removed for repairs and fuel in the reactor (Condition Report 200201248)
- Containment Penetration M-39 Isolation Valve HCV-425B exceeding administrative leak limits (Condition Report 200201592)
- Containment Penetration M-45 Isolation Valves HCV-2504A and HCV-2504B exceeding administrative leak limits (Condition Report 200201912)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors observed the installation of a modification to the Auxiliary Feedwater System. Specifically, a 4-inch gate valve was installed in the system to allow full flow testing of Auxiliary Feedwater Pumps AFW-6 and AFW-10 without stationing a dedicated operator at main and auxiliary feedwater crossconnect Valve HCV-1384. The inspectors reviewed Work Order 00117552, Engineering Change 30191, and Quality Control Reports 20021299 and 20021308.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors verified that postmaintenance tests were adequate to verify system operability and functional capabilities. The inspectors verified that testing met design and licensing bases requirements, Technical Specifications, the Updated Safety Analysis Report, inservice testing, and licensee administrative procedures. The inspectors verified test results for the following components:

- Closed indication for Pressurizer Vent Stop Valve HCV-178 repairs on May 29, 2002
- Air actuator NCV-6680B-2-0 repairs on May 29, 2002
- Work Order 00109544, Emergency Diesel Generator anti-freeze replacement on June 12, 2002

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors periodically observed plant conditions to verify that safety systems and support systems, including electrical distribution, were properly aligned with the shutdown operations protection plan. In addition, the inspectors observed the following evolutions: reactor shutdown, midloop operations, core reload from containment and the spent fuel building, and plant heatup. The inspectors performed a containment cleanliness and material readiness tour prior to the unit's return to power.

b. Findings

The licensee exercised a nonconservative decision making process when controlling foreign materials in containment and eliminating the potential for blocking the containment emergency sumps. As a result, approximately 20 55-gallon drums with paper taped onto the lid on the same elevation as the sumps and several hundred other pieces of tape remained in containment during initial plant heatup following a refueling outage. This was determined to be of very low safety significance using the Significance Determination Process.

On May 29, 2002, the inspectors performed a walkdown of the containment building to verify cleanliness and material readiness prior to the unit returning to power. The inspectors observed approximately 20 55-gallon drums with paper taped onto the lid on the same elevation as the sumps and several hundred other pieces of tape throughout containment. The unit, at that time, was at 300 degrees and pressurized to 600 psi with a heatup in progress. Licensee Technical Specification 2.3, "Emergency Core Cooling," states in part that all equipment required to function during accident conditions are to be operable prior to the reactor being made critical. Licensee Standing Order SO-M-10, "Foreign Material Exclusion," specifies that the intent of the procedure is to eliminate the potential for restricting flow to the emergency core cooling system suction strainers. The inspectors reviewed this and the revision of Procedure OI-CO-1, "Containment Closeout." The inspectors noted that neither procedure addressed keeping the emergency core cooling system suction strainers free of debris during plant heatup. The decision by the licensee to only consider the operational condition of the suction strainers, with respect to debris in containment, just prior to criticality and not during plant heatup, was nonconservative. The inspectors informed outage management about the observations. The unit, at that time, was at 300 degrees and pressurized to 600 psi with a heat-up in progress. The licensee dispatched personnel into containment to address the inspectors concerns. The inspectors reviewed Procedures OI-CO-1, “Containment Closeout,” Revision 23, and SO-M-10, “Foreign Material Exclusion,” Revision 22. The inspectors noted that neither procedure addressed keeping the emergency sumps free of debris during plant heat-up. The licensee initiated Condition Report 200202326 to address the inspectors’ concerns.

The inspectors evaluated the significance of the issue. The inspectors determined this issue had a credible impact on safety because of the potential for plugging the containment emergency sumps (Group 1 question answered yes). The inspectors determined that the issue could credibly affect the availability of a mitigating system (Group 2 question answered yes). However, Using the significance determination process Phase 1 worksheet, the inspectors answered “no” to all questions under the mitigating systems cornerstone and therefore the issue screened out as Green (FIN 285/2002002-01).

This finding is in the licensee's corrective action program as Condition Report 200202326.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed/reviewed the following surveillance tests to ensure that the systems tested were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors verified that the following surveillance test met Technical Specifications, ASME Section XI test requirements, the Updated Safety Analysis Report, and licensee procedural requirements:

- Channel B safety injection, containment spray and recirculation actuation signal test on April 19, 2002
- Shutdown cooling pump refueling leakage test on April 24, 2002
- Personnel access lock O-ring seal test on May 28, 2002
- Control element assemblies drive system interlocks check on May 30, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 30027, "Restrain or Disable Refueling System Fuel Handling Machine (FH-1) in the Retracted Position" to verify that the safety functions of the system were not affected.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA (As Low as Reasonably Achievable) Planning and Controls (71121.02)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel to determine if low dose waiting areas were utilized, personnel were maintaining doses ALARA, and radiation workers were receiving appropriate job supervision and radiation protection coverage. The inspectors reviewed a summary of ALARA and radiological worker performance condition reports written since November 2001. The following condition reports were reviewed in detail: 200103694, 200103725, 200200268, 200200379, 200200288, and 200200392.

The inspectors reviewed nine radiation work permit packages. These packages included five pre-outage and four spent resin transfer recovery packages for work activities with the highest estimated personnel collective exposures and selected total effective dose equivalent ALARA evaluations. The following radiation work permits were reviewed: 02-2510, “Steam Generator Support in High Radiation Areas”; 02-3007, “Investigate Cause and Extent of Radiological Conditions Following Dewatering of the Spent Resin Storage Tank”; 02-3008, “Drain Vent Header in Rooms AI-100, 7, 13 and 16”; 02-3009, “Dewater Spent Resin Storage Tank (SRST)”; 02-3011, “Restore Radiological Conditions Pre-SRST Overfill”; 02-3509, “Primary Side Steam Generator Work Requiring Full Jumps”; 02-3510, “Primary Side Steam Generator Support Work”; 02-3529, “Upper Guide Structure Lift Rig Repair”; 02-3534, “Inspection of Nuclear Well Cooling System Insulation.”

The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposures ALARA:

- ALARA program procedures

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information

- Use of engineering and administrative controls to achieve dose reductions, including four temporary shielding request packages (TSR 02-034, TSR 02-035, TSR 02-037, TSR 02-038) and scheduling of work activities

- Individual exposures of selected work groups (steam fitter mechanics, mechanical maintenance, radiological operations, and radwaste operations)

- Five preoutage ALARA packages for work activities with the highest estimated personnel collective exposures (02-08, "Fuel Movement"; 02-10, "A/B Steam Generator Primary Side Services"; 02-11, "A/B Steam Generator Secondary Side Services"; 02-12, "Reactor Head Removal and Replacement Tasks"; 02-18, "Replacement of RC-3C Rotating Assembly")

- Draft 2002 ALARA Program One Year and Five Year Plan, Nuclear Safety Review Group Observation 02-QUA-018 and selected quality assurance surveillance observations from November 2001 to February 2002 (2001-480, 481, 576, and 592 and 2002-21, 49, 126, 158, 161, and 166)

- Hot spot and point source tracking and reduction program

- Overall facility source term reduction plan

- Radiological work planning and interfaces between various departments

- ALARA Committee Meeting minutes since November 2001

There were no declared pregnant workers since November 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA5 Other

Temporary Instruction 2515/145: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

a. Inspection Scope

The inspectors observed and reviewed licensee activities in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head

Penetration Nozzles," issued on August 3, 2001, in response to identified circumferential cracking in control element drive mechanism nozzles at other facilities.

The licensee performed a 100 percent visual inspection of the reactor pressure vessel head nozzle penetrations using a contractor developed procedure approved by the licensee. The licensee used a robotic device to perform a 360 degree inspection around each nozzle penetration. The penetrations near the reactor pressure vessel head outer edge could not be inspected by the robotic device due to the head and insulation configuration. These areas were inspected using a boroscope that was attached to the robotic device.

The licensee detected no evidence of boric acid deposits as described in NRC Bulletin 2001-001. However, the licensee observed boric acid stains in some locations on the reactor pressure vessel head and on some nozzles that were associated with flange leakage. The vessel head contained small debris particles located around nozzles where they penetrate the reactor vessel head and a few small mechanical fasteners. The debris was easily blown away from the nozzle area with air. All deficiencies identified were dispositioned.

The licensee's quality control personnel involved with the inspection were VT-2 qualified. They received additional training from the contractor on how the robotic inspection would take place. The training consisted of how the robot was operated, the limitations, and practical experience on a reactor pressure vessel head mockup.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

.1 The inspectors presented the inspection results to Mr. D. Bannister, Plant Manager, and other members of licensee management at the conclusion of the ALARA planning and controls inspection on April 12, 2002. The licensee acknowledged the findings presented. No proprietary information was identified.

.2 The inspectors presented the inspection results to Mr. R. Ridenoure, Division Manager, Nuclear Operations, at the conclusion of the inservice inspection program inspection on May 22, 2002. The licensee acknowledged the findings presented.

The inspectors noted that, while proprietary information was reviewed, none would be included in this report.

.3 The inspectors presented the inspection results to Mr. R. Ridenoure, Division Manager, Nuclear Operations, at the conclusion of the resident inspectors' inspection on July 2, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

KEY POINTS OF CONTACT

Licensee

D. Bannister, Plant Manager
C. Bloyd, Component Testing Supervisor
G. Cavanaugh, Acting Manager, Nuclear Licensing
J. Chase, Division Manager, Alliance
S. Coufal, ALARA Technician, Radiation Protection
D. Dryden, Engineer, Station Licensing
M. Core, Manager, System Engineering
J. Goodell, Manager, Operations
P. Hamer, Component Testing, Inservice Inspection Program Coordinator
B. Lisowyj, Design Engineer
J. McBride, ALARA Technician, Radiation Protection
J. McKinley, Acting Manager, Maintenance
E. Matzke, Licensing Engineer
R. Phelps, Division Manager, Nuclear Engineering Division
M. Puckett, Manager, Radiation Protection
R. Reno, ALARA and Radiological Equipment Supervisor, Radiation Protection
R. Ridenoure, Division Manager, Nuclear Operations
K. Steele, Operations Supervisor, Radiation Protection
C. Williams, ALARA Technician, Radiation Protection
K. Woods, Senior Nuclear Design Engineer

Others:

R. Maurer, Westinghouse Corporate NDE Level III
K. Rajan, Westinghouse Steam Generator Programs

ITEMS OPENED AND CLOSED

50/285/02-01	FIN	Nonconservative processes for Controlling Containment Cleanliness (Section 1R20)

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
CEP-ISI-004	Arkansas Nuclear One Unit 2 Inservice Inspection Plan	1
	Operational Assessment of ANO-2 Steam Generator Tubing for Cycle 15 (ER 981220 E207)	2/03/01
	Steam Generator Pre-Outage Degradation Assessment and Repair Criteria for 2R15 (ER-ANO-1997-4855-004)	1
Engineering Change EC 28012	“Replacement of Check Valves SI-194 and SI-197”	0
LIC-01-0075	“Response to NRC Bulletin 2001-01 ‘Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles’”	August 31, 2001
LIC-02-0034	“Response to NRC Bulletin 2002-01 ‘Reactor Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity’”	April 1, 2002
LIC-95-0126	“Response to Generic Letter 95-03, Circumferential Cracking of Steam Generator Tubes”	June 23, 1995
LIC-98-0031	“Response to Generic Letter 97-05, Steam Generator Tube Inspection Techniques”	March 16, 1998
LIC-98-0039	“Response to Generic Letter 97-06, Degradation of Steam Generator Internals”	March 25, 1998
NEI 97-06	Nuclear Energy Institute Document, “Steam Generator Program Guidelines”	1
TR-107569-V1R5	EPRI Document, “PWR Steam Generator Examination Guidelines”	5, Volume 1
TR-104788-R2	EPRI Document, “PWR Primary-To-Secondary Leak Guidelines”	2

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
TR-107621-R1	EPRI Document, “Steam Generator Integrity Assessment Guidelines”	1
TR-107620-R1	EPRI Document, “Steam Generator In Situ Pressure Test Guidelines”	1
Licensee Report SG-SGDA-02-9	“Fort Calhoun Station Steam Generator Degradation Assessment, 02RFO Pre-Outage Planning Report.”	0
Westinghouse Report CSE-01-054	“Steam Generator Secondary Side Visual Inspection”	September 21, 2001
PED-SEI-42	“Steam Generator Program.”	2
Standing Order SO-G-105	“Steam Generator Tube Leakage”	9
	“Fort Calhoun Station Data Analyst Indoctrination”	May 2000
Westinghouse letter	“Reconciliation of FCS Data Analysis Procedure and EPRI ETSS”	May 9, 2002
ABB Combustion Engineering Document	“Comparison of Detectability for Tube wall Degradations (OD-SCC and OD-IGA) When Testing Nuclear Steam Generators Tubing with Downsized Bobbin Type Probes”	October 14, 1996
ABB Combustion Engineering Document	“Comparison of Detectability for Pit-Like Degradations When Testing Nuclear Steam Generators Tubing with Downsized Bobbin Type Probes”	October 12, 1996
ABB Combustion Engineering Report 9410-CH-RPT-98-001	“Comparison of Eddy Current Data Using Various Cable Lengths”	0

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
	Summary listing of steam generator-related Condition Reports that were issued for the time period of March 16, 2001, to May 16, 2000	
Westinghouse Report CSE-01-038	"Safety Evaluation for Steam Generator Secondary Side Foreign Objects"	0
Westinghouse Field Service Traveler CFTC1-SG-004	"In Situ Pressure Test Using the Computerized Data Acquisition System"	00
Data Sheet 5.2	In Situ Pressure Test Details, Tube - Row 78, Column 59	
ETSS # 20409.1	EPRI Examination Technique Specification Sheet	2
ETSS # 96005.2	EPRI Examination Technique Specification Sheet	6
ETSS # 96008.1	EPRI Examination Technique Specification Sheet	9
	Acquisition Technique Sheet No. B1, (Bobbin Coil).	02
	Acquisition Technique Sheet No. R3 (Rotating Coil/2-coil solid body probe).	02
	Analysis Technique Sheet No. B3	02
	Analysis Technique Sheet No. R3	05

Procedures

NUMBER	TITLE	REVISION
CFTC1-ISI-351	Automated Ultrasonic Procedure for the Inspection of CEDM Upper Housing Welds From the Outside Diameter	0

Procedures

NUMBER	TITLE	REVISION
83-0041	Reactor Vessel Head Penetration Remote Visual Inspection for Fort Calhoun	0
OPPD-UT-89-8	Manual Ultrasonic Examination of Ferritic Piping Welds	0, Add. 1
WPS 08-08-TS-001	Welding Procedure Specification	2
WPS 8-MC-GTAW	Welding Procedure Specification	4
20.A.100-1992	Radiographic Examination of Welds (General Requirements)	3
20.A.131-1992	Radiographic Examination of Welds	1
OPPD-400-002	"Analysis of Steam Generator Tube Eddy Current Data"	4 thru Change Form 3
SE-PM-MS-0100	"Visual Inspection of Steam Generator Secondary Side"	0
SE-RR-RC-0100	"Steam Generator Tube Plugging and Sleeving"	2
SE-ST-RC-0003	"Inservice Testing of Steam Generator Tubes"	1
CH-AD-0007	"Primary to Secondary Leak Rate Determination"	13
STD-400-024	ABB Combustion Engineering Procedure, "Procedure for the Eddy Current Examination of the CEDM Seal Housing"	3
STD-400-173	Westinghouse Field Service Procedure, "Checkout and Operation of the Steam Generator Tube In Situ Pressure Test System"	03

Drawings

E 232-417-11	Closure Head Housing Details	0
E 23866-164-005	Nozzle Requirements - Closure Head	5
E 2927	Upper Housing Assembly	5

E 2520	Fabrication Skewtch For SI-194 and HCV-331	7
E 23866-163-010	Control Element Drive Mechanism Outline and Installation Data	7

Condition Report

200101382, dated April 11, 2001

Work Orders

00103922-01, UT of Upper Housing RC-10-20
00103873-01, UT of Upper Housing RC-10-21
00103925-01, UT of Upper Housing RC-10-18
00103868-01, UT of Upper Housing YE-116B (HJTC)
00102910-01, UT of Upper Housing YE-116A (HJTC)
00103924-01, UT of Upper Housing RC-10-19
0091590, Replace LPSI to RC Loop 2B Check Valve SI-197
0091589, Replace LPSI to RC Loop 2A Check Valve SI-194