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

March 28, 2007

R. T. Ridenoure
Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION - NRC INSPECTION REPORT 05000285/2006006

Dear Mr. Ridenoure:

On February 12, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. The enclosed report documents the inspection findings, which were discussed on February 12, 2007, with Mr. Reinhart and other members of your staff.

The 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. This inspection covers steam generator, pressurizer and reactor vessel head replacement activities.

Based on the results of this inspection, no findings of significance were identified.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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).

Sincerely,

/RA/

William B. Jones, Chief
Engineering Branch 1
Division of Reactor Safety

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2006006
w/Attachments: 1. Supplemental Information
2. Official Use Only - Security-Related Information

cc w/o Attachment 2:
Joe I. McManis, Manager - Licensing
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Electronic distribution by RIV **w/o Attachment 2**:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
DRS Deputy Director **(RJC1)**
Senior Resident Inspector **(JDH1)**
Resident Inspector **(LMW1)**
Branch Chief, DRP/E **(JAC)**
Senior Project Engineer, DRP/E **(JCK3)**
Team Leader, DRP/TSS **(FLB2)**
RITS Coordinator **(MSH3)**
DRS STA **(DAP)**
D. Cullison, OEDO RIV Coordinator **(DGC)**
ROPreports
FCS Site Secretary **(BMM)**

SUNSI Review Completed: ___Y_____ ADAMS: √ Yes ☐ No Initials: __WBJ____
√ Publicly Available ☐ Non-Publicly Available ☐ Sensitive √ Non-Sensitive

Security-Related **ATTACHMENT 2** - OUO:
SUNSI Review Completed: __Y____ ADAMS: √ Yes Initials: ____WBJ__
√ Non-Publicly Available √ Sensitive

RI:EB1	HP:PSB	RI:EB1	RI:NSPDP	RI:NSPDP	SRI:PBE	SPSI:PSB
JPAdams/lmb	BDBaca	GAGeorge	STGraves	JRGroom	JHanna	DAHolman
/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
3/9/07	3/12/07	3/13/07	3/12/07	3/12/07	3/10/07	3/13/07

PE:RPB6:RII	HP:PSB	RI:PBE	C:EB1	C:PSB	C:PBE	C:EB1
TNazerio	DLStearns	LWilloughby	WBJones	MPShannon	JAClark	WBJones
/RA/	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
3/13/07	3/12/07	3/12/07	3/27/07	3/27/07	3/27/07	3/28/07

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2006006
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm. P.O. Box 399, Highway 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	March 6, 2006, through February 12, 2007
Inspectors:	J. Adams, Reactor Inspector, Engineering Branch 1 B. Baca, Health Physicist, Plant Support Branch G. George, Reactor Inspector, Engineering Branch 1 S. Graves, Reactor Inspector (NSPDP) J. Groom, Reactor Inspector (NSPDP) J. Hanna, Senior Resident Inspector, Projects Branch E D. Holman, Senior Physical Security Inspector, Plant Support Branch T. Nazerio, Project Engineer, Reactor Projects Branch 6, Region II D. Stearns, Health Physicist, Plant Support Branch L. Willoughby, Resident Inspector, Projects Branch E
Approved By:	William B. Jones, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000285/2006006; 3/6/06 - 2/12/07; Fort Calhoun Station; Integrated Resident and Regional Report of Steam Generator, Pressurizer and Reactor Vessel Closure Head Replacement Activities.

This report covered a 11-month period of inspections by ten resident and regional inspectors. No findings of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management's review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

4OA5 Other Activities

.1 Steam Generator, Pressurizer and Reactor Vessel Head Replacement Projects (50001, 50003, 71007)

a. Inspection Scope

This inspection report discusses NRC inspection activities related to the Fort Calhoun steam generator, reactor vessel head, and pressurizer replacement projects. These three inspection procedures are presented together because, in most instances, the inspection procedures required the same inspection tasks and there was considerable overlap between the inspections. In the few instances where different component specific inspections were performed, they are provided later under the "Component Specific Inspections."

These three inspection activities are not part of the normal baseline inspection program but are performed on an as-needed basis. Therefore, no sample size is specified. The inspectors completed the entire procedure for each project, with the exception of the disposal radiological plans. This inspection will be performed at a later date.

Inspection Activities Common to All Three Projects:

A significant portion of these inspection efforts were completed and documented in prior NRC inspection reports. Each of the following past inspections were related to all three projects:

- 50.59 and modification inspections as well as emergenct work risk assessments, NRC Inspection Report 05000285/2006004, Sections 1R02, 1R13, and 1R17 respectively

- Radiation protection program controls, NRC Inspection Reports 05000285/2006004 and 05000285/2006005, Sections 2OS1 and 2OS2 (both reports)

- Storage radiological safety plans, NRC Inspection Reports 05000285/2006004 and 05000285/2006005, Sections 2OS1 and 2OS2 (both reports)

- Welding and nondestructive (NDE) examinations, NRC Inspection Report 05000285/2006005, Sections 1R08 and 1R22

- Outage operating conditions, NRC Inspection Reports 05000285/2006004 and 05000285/2006005, Sections 1R20, 2OS1, and 2OS2 (all sections in both reports)

- Surveillance and postmaintenance testing, NRC Inspection Report 05000285/2006005, Sections 1R19 and 1R22

- Fire protection and mitigation, NRC Inspection Report 05000/2006005, Section 1R05

In addition to the above, as part of this current inspection effort, the inspectors reviewed additional activities to ensure proper (1) design and planning; (2) removal and replacement; and (3) post-installation verification and testing of the projects, as provided below:

Engineering and Technical Support. The inspectors reviewed engineering and technical support activities prior to, and during, the replacement outage. The inspectors reviewed the following temporary modifications:

- Temporary Outage Transformer
- Temporary Power to the Containment Palfinger Crane
- The temporary containment opening modification.

The inspectors verified the technical adequacy of the temporary modifications, including 10 CFR 50.59 related documents.

Lifting and Rigging. The inspectors reviewed engineering design and analysis associated with major component lifting and rigging projects. This included: (1) crane and rigging equipment, (2) steam generator, pressurizer, and reactor vessel head component drop analysis, (3) safe load paths, and (4) load lay-down areas. The inspectors verified that the licensee had properly evaluated the potential impact of load handling on the reactor core and spent fuel, including cooling and support systems. This review also included the potential impact to underground electrical lines and fluid piping that were traversed by the heavy haul route.

The inspectors conducted reviews of the preparations and procedures for: (1) crane and rigging inspections; (2) testing; (3) equipment modifications; (3) lay-down area preparations; and (4) training. The related activities and equipment included:

- Area preparation for the outside systems

- Outside lift system

- Hatch transfer system

- Reactor cavity decking

- Inside lifting device

- Upending device

- Steam generator, pressurizer, and reactor vessel head removal, placement of new components, and transport of the old units to the storage facility

The inspectors directly observed the movement of the installed Steam Generator A out of containment, as well as transfer of the replacement Steam Generator A into containment. The inspectors also observed portions of Steam Generator B movement. In addition, the inspectors observed the removal of the "old" reactor vessel upper head, and the insertion into containment of the new reactor vessel head. The inspectors observed a sample of various other activities including the initial lifts, heavy haul routes, and use of the transfer equipment.

Radiation Protection Program Controls. The inspectors reviewed the following additional areas as part of this inspection: (1) contamination controls, (2) emergency contingencies, and (3) project staffing and training plans. The inspectors used the requirements in 10 CFR Part 20 and the licensee's technical specifications and procedures as criteria for determining compliance.

Disposal Radiological Safety Plans. Disposal activities will be addressed in a future NRC inspection report, under Inspection Procedures 71122.02, "Radioactive Material Processing and Transportation," and 71122.03, "Radiological Environmental Monitoring Program (REMP) and Radioactive Material Control Program."

Security Related Activities. Additional activities performed are documented in Attachment 2, which is designated and marked as "Official Use Only - Security-Related Information."

Major Structural Modifications. The inspectors reviewed documentation related to structural modifications to facilitate steam generator, pressurizer and reactor vessel head replacement, including the structural supports for the steam generators and pressurizer, the temporary reactor coolant system piping structural supports, and all attached piping during all phases of removal and installation.

The inspectors reviewed the major structural modification to reroute the 161kV transmission lines to allow clearance for the major components to move through the temporary equipment penetration made in containment. The inspectors also completed a review of modifications to the reactor coolant gas system piping inside containment.

Containment Access and Integrity. The inspectors reviewed the following activities:

- The licensee's evaluation of containment concrete voiding issues, found during the temporary equipment opening excavation

- The licensee's actions associated with a liner plate weld repair and the subsequent retest

- Procedures for installing reinforcing steel, Cadweld splices, and control of concrete placement

- Concrete pours

- Material test results (cement, fine and coarse aggregate, water, and admixtures)

- Concrete mix data, to ensure that selected trial mix met concrete design strength requirements

- Acceptance criteria for the plastic concrete

- Concrete batch plant and mixer inspection results

The inspectors verified that the licensee's actions were consistent with the applicable Codes -ACI 318-63, Part IV-B, *Building Code Requirements for Reinforced Concrete Institute*, 1963; AWS D1.4-98, *Structural Welding Code-Reinforcing Steel*; and American Society of Mechanical Engineers (ASME) Section III, *Rules for construction of Nuclear Power Plant Components*, 1968. The inspectors examined the reinforcing steel to ensure it was installed in accordance with design requirements, observed the concrete forms to ensure tightness and cleanliness, and that reinforcing steel was clean.

The inspectors reviewed activities pertaining to concrete delivery time, free fall, flow distance, layer thickness and concrete consolidation conformed to industry standards established by the American Concrete Institute. Concrete batch tickets were examined to ensure that the specified concrete mix was being delivered to the site. The inspectors also witnessed testing of the plastic concrete for slump, air, and temperature, unit weight, molding and storage of the concrete cylinders for testing. The inspectors performed reviews to ensure concrete testing was performed and the cylinders were molded in accordance with applicable American Society for Testing and Materials (ASTM) requirements. Finally, the inspectors reviewed activities to ensure that concrete testing was performed by qualified personnel from an independent testing company, and that concrete placement activities were continuously monitored by licensee and contractor quality control and quality assurance personnel.

The inspectors verified that concrete batching activities included proper storage and separation of materials, as well as appropriate temperature controls. The inspectors also verified that the contractor's inspection of the trucks and batch plant were performed in accordance with the guidance of the National Ready Mixed Concrete Association and mixer efficiency tests were performed on the truck mixers in accordance with Standard ASTM C-94, *Ready-Mixed Concrete*. The inspectors reviewed the concrete mix data to ensure that mix proportions for delivered concrete were selected based on trial concrete mix results, that quality control acceptance criteria for the plastic concrete were based on the trial mixes, and that the trial mix met concrete strength requirements.

<u>Additional Post-installation Verification and Testing</u>. The inspectors reviewed:

- Implementation of the licensee's post-installation inspections and verifications program, including witnessing the auxiliary feedwater functional test

- Pressurizer performance test and the pressurizer heaters resistance and insulation test.

- Calibration and testing of instrumentation affected by pressurizer replacement - for example, the inspectors reviewed critical functions such as the level control & actuation setpoints, and validation of pressurizer heater performance

- Control element assembly position indication system check and control element assembly group indicating lights and rod drop testing **Note**: The inspectors closely reviewed the connections of the control element assemblies to the rack extensions once radial misalignments were identified by the licensee.

- Reactor vessel seal leakoff monitoring system, and core exit thermocouple performance following reactor startup

- Post-installation testing of the reactor vessel head. The inspectors also reviewed critical functions of the replacement reactor vessel head that potentially might be adversely affected either by improper design or during installation activities

- Procedures required for equipment performance testing to confirm the design and to establish baseline measurements

- Preservice inspection of new steam generator, pressurizer and reactor head welds

The inspectors verified equipment performance was consistent with the proceduralized acceptance criteria and design requirements.

<u>Containment Integrated Leak-Rate Test</u>. The inspectors walked down the installation of test equipment used to pressurize the containment for performing a containment leak check. The inspectors verified that the actual equipment configuration was consistent with installation and test records. The inspectors also verified that the equipment was operating properly during the test and that instrument calibrations were current.

The inspectors verified through observation, records review, and independent calculations that the containment integrated leak rate test was properly conducted. The inspectors observed the initial pressurization of containment including the communications established for the performance of the test. In addition, the inspectors independently verified the acceptability of the test results.

<u>Component Specific Inspections:</u>

<u>Pressurizer</u>.

<u>Foreign Material Controls</u>. The inspectors directly observed the cuts of pressurizer piping and the foreign material exclusion boundaries for the remaining reactor coolant system piping. Particular attention was focused on grinding and other debris-generating work in the area and the possibility of introduction of loose material. The inspectors also performed engineering reviews of the foreign material controls.

Reactor Vessel Head Fabrication Inspections at Licensee Facility. The inspectors performed the following reactor vessel head fabrication inspection activities.

Heat Treatment. The inspectors verified that the ASME, Section III, requirements for reactor vessel head forging heat treatment were correctly reflected in the certified design specification, procurement specification, and the fabrication specification. The inspectors also verified that the heat treatment used by the fabrication vendor was performed in accordance with the specifications. The inspectors reviewed the material test report for the reactor vessel head forging and verified that the heat treatment parameters met the applicable procurement specifications and ASME Code requirements. The inspectors also reviewed the material test record for the finished reactor vessel head and verified that the post-weld heat treatments were conducted in accordance with the ASME Code, including furnace temperature and conditions, thermocouple placement, heating and cooling rates, and documentation requirements.

Nondestructive Examination. The inspectors reviewed the NDE program for the fabrication vendor, including a review of the NDE technician certifications.

Welding. For the cladding, the inspectors reviewed the design specification, design drawings, weld records, NDE records, defect disposition reports, weld procedure specifications, procedure qualification records, welder certifications, and dimensional records and verified that the cladding was performed per specification.

For the control element drive mechanism (CEDM) flange-to-nozzle welds, the penetration nozzle-to-head weld buttering welds, and the CEDM nozzle-to-head J-Groove welds, the inspectors reviewed the certified material test reports, the weld procedure specifications, procedure qualification records, welder certification records, and weld map drawings. The inspectors verified that these welds were performed in accordance with the ASME code and procurement specifications. The inspectors also reviewed the code-required NDE of these welds, including NDE technician certifications, NDE procedures (including NDE solvents for the penetrant testing procedure), and NDE records and verified that these inspections were performed in accordance with authorized procedures and the ASME code.

Weld Repairs. The inspectors reviewed selected weld repairs to ensure they were conducted in accordance with Section IX of the ASME code and procurement specifications. The inspector reviewed the certifications of the welder and NDE technicians, results of the pre- and post-repair NDE examinations, and the records of the repair. The inspectors also reviewed selected nonconformance reports, where reportable indications were found and were dispositioned without repair, to verify that these were performed in accordance with the ASME code and contract requirements. The inspectors concluded that the repairs were done in accordance with applicable codes and specifications.

Code Reconciliation. The inspectors reviewed supplemental examinations, analysis, and ASME Code documentation reconciliation to ensure that the original ASME Code –Stamp remained valid, and that the replacement head complied with appropriate NRC requirements and applicable industry standards. The inspectors also ensured that the design specification was reconciled and a design report was prepared for the reconciliation of the replacement head, verifying that they were certified by professional engineers competent in ASME Code requirements.

Quality Assurance Program. To the extent practicable, the inspectors ensured that machining was carried out under a controlled system of operation, a drawing/document control system was in use in the manufacturing process, and that part identification and traceability was maintained throughout processing and was consistent with the manufacturer's quality assurance program.

4OA6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. J. Reinhart, Site Director, and other members of the licensee's management staff on February 12, 2007 during a telephonic exit meeting. The licensee acknowledged the information presented. Some proprietary information was reviewed during this inspection but no proprietary information was included in this report.

ATTACHMENTS: 1. Supplemental Information
2. Official Use Only - Security-Related Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Cavenaugh, Supervisor, Regulatory Compliance
D. Guinn, Licensing Engineer
E. Matski, Compliance
J. Reinhart, Site Director
J. Cate, Supervisor, System Engineering
J. Spilker, NSSSRP - Engineering
R. Short, Manager, NSSSRP
S. Gambhir, Division Manager, Nuclear Projects
R. Ruhge, Supervisor - QC
R. Bayer, NSSS Installation Manager
J. Bednash, Resident Technical Rep. (MHI)
D. Spear, Mechanical Design Engineer,
D. Cyboron, Reactor Vessel Head/Pressurizer Components Lead
P. Ward, Installation Engineering
D. Pier, Control Room Supervisor

LIST OF DOCUMENTS REVIEWED

In addition to the documents referred to in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Procedures		
Number	Title	Revision/Date
SO-G-23	Surveillance Test Program	53
SO-G-92	Conduct of Infrequently Performed Procedures	10
IC-ST-AE-3139	Type C Local Leakage Rate Test of Penetrations M39 and M53	11
IC-ST-RC-0028	Channel Calibration of Pressurizer Pressure Loop D/P-102	3
IC-ST-RC-0030	Channel Calibration of Pressurizer Safety Valve RC-141 Tailpipe	3
OP-ST–CEA-0006	Refueling Control Element Assembly (CEA) Group Indicating Lights and Rod Drop Test	9

Number	Title	Revision/Date
OP-ST-CEA-0005	Control Element Assemblies CEA Drive System Interlocks Check	15
OP-ST-CEA-0001	Control Element Assembly Position Indication System (ICAPIS) Check	10
RE-CPT-RX-0001	Post Refueling Core Physics Testing and Power Ascension	38
OI-RR-1	Reactor Regulating System Normal Operation	21
OI-RC-2A	RCS Fill and Drain Operations	53
32ST-9RC01	92 Day Pressurizer Heater Capacity Test	8
CP-C-05	Grouting	0
CP-C-01	Concrete Operations	0
MM-RR-RC-0305	Removal of Reactor Vessel Closure Head, old Down Ring, and Upper Guide Structure	23
P-2739-42	Removal of Old Steam Generators	0
MM-RR-RC-0314	Reactor Vessel Closure Head Installation	16
GM-OI-HE-1	Operating Instruction Polar Crane Normal Operation	8
MHI-MT-98-1	Magnetic Particle Examination of Welds and Bolting	0
MHI-PT-98-1	Liquid Penetrant Examination - Solvent Removable, Visible Dye Technique	0
UGS-L5-030105	Penetrant Examination Procedure	0
EC31589-T005	AFW Test	1
EC31589-T011-50	RSG Warranty Test: Steam Generator Water Level Fluctuation - Low Power	0
CP-C-01	Concrete Operations	0
EC31589-T011-80	RSG Warranty Test: Steam Generator Water Level Fluctuation - Medium Power	0
CP-C-05	Grouting	0
EC32447-T001	Pressurizer Performance Test	0
OP-ST-CEA-0006	Refueling Control Element Assembly (CEA) Group Indicating Lights and Rod Drop Test	9
OP-ST-CEA-0005	Control Element Assemblies CEA Drive System Interlocks Check	15

Number	Title	Revision/Date
OP-ST-CEA-0001	Control Element Assembly Position Indication System (ICAPIS) Check	10
OI-RR-1	Reactor Regulating System Normal Operation	21
EE-RR-CEDM-0100	Removal, Inspection and Installation of CEDM Drive Package	6
EC-31589	Post Modification Testing for the Nuclear Steam Supply Replacement Project	May 11, 2006
EC-33089	N1 NSSSRP Containment Modifications	August 25, 2006
EC-33104	Post Modification Testing for Installation of Reactor Coolant System Piping, and Tubing for the Nuclear Steam Supply Replacement Project	October 24, 2005
EC-33105	Post Modification Testing for Installation of Instrumentation Piping, and Tubing for the Pressurizer for the Nuclear Steam Supply Replacement Project	October 24, 2005
EC-33113	Containment Temporary Structures	September 12, 2005
EC-33114	Containment Temporary Utilities	October 25, 2005
EC-33153	Post Modification Testing for Replacement Reactor Vessel Head Installation	May 11, 2006
EC-33629-T001	Turbine Post Modification Test Plan	1
EC-33110	Rigging and Handling	0
EC-33101	Outside Containment Structures and Equipment	0
UGS-L5-030146	Hydrostatic Test Procedure	3
UGS-L5-030142	UT Procedure (Cladding)	0
UGS-L5-030138	PT Procedure (Solvent Removable)	5
N-7624-30	Ultrasonic Examination Procedure for Closure Head Forging	2
N-7624-40	Magnetic Particle Examination Procedure for Closure Head Forging	1
UGS-L5-030140	Magnetic Particle Examination Procedure (Yoke Method)	0

Number	Title	Revision/Date
UGS-L5-030136	Material Verification Procedure	1
EC-39412	Tubing Separation Modifications for Steam Generators	

Calculations		
Number	Title	Revision
06Q4630-CAL-001	Stress Evaluation of Fort Calhoun Containment Liner Considering Concrete Voids	
FC07052	Equipment Hatch Runway System	1
FC07445	Evaluation of Fort Calhoun Containment Shell Concrete Voids	0
FC07446	Re-Analysis of Fort Calhoun Containment Liner Considering Concrete Voids	0
FC07173	Evaluation of Containment Building Liner Plate for Temprary Conditions	0
25036-C-010	NSSS Component Weights	1
C-2739-14	Structural Calculations, Fort Calhoun - NSSS Refurbishment Project Outside Lift Rigging for NSSS Components	0
C-2739-60	Operational Calculations, Fort Calhoun - NSSS Refurbishment Project OSG/RSG Trailer Transport	1
C-2739-30	Structural Calculations, Fort Calhoun - NSSS Refurbishment Project, U/D Cart	1
C-2739-38	Structural Calculations, Fort Calhoun - NSSS Refurbishment Project, Cart Towing System - General Analysis	0
C-2739-36	Structural Calculations, Fort Calhoun - NSSS Refurbishment Project, ‘Towing Calculation	0
C-2739-40	Structural Calculations, Fort Calhoun - NSSS Refurbishment Project , TLD Load Case 1	0
CN-CFTCRPZR-01	Omaha Public Power District, Fort Calhoun Station Replacement Pressurizer Weight and CG Calculation	1

FC07216	Fort Calhoun Station – Weight & Center of Gravity Calculation for Head Lift	1
FC00064	Polar Crane Compliance to CMAA-70	0
32-5046461-01	FCS RSG - Decay Heat Removal Cap. In Nat. Circ. Analysis	9/30/06

Miscellaneous Documents		
Number	Title	Revision
	Certificate of Conformance for Concrete Production Facilities, NRMCA	
WPIR C-COI-186	Containment Opening	0
WPIR R-SLX-190	Liner Plate Removal and Restoration	0
WPIR C-TEN-253	Removal and Reinstallation of pre-outage tendons- work performed at bottom end	0
WPIR C-TEN-232	Removal and Reinstallation of pre-outage tendons- work performed at top end	0
WPIR C-COT-188	Removal and Reinstallation of containment opening tendons- work performed at top end	0
WPIR C-TEN-255	Post tension VT inspections - top end of tendons	0
WPIR C-TEN-256	Post tension VT inspections - bottom end of tendons	0
WPIR C-COT-254	Removal and installation of containment opening tendons- work performed at bottom end	0
TS 25036-C-321 (Q)	TS for Purchase of CQE Ready-Mixed Concrete	3
TS 25036-C-304 (Q)	TS for Installation of CQE Reinforcing Steel	0
TS 25036-C-322 (Q)	TS for placement CQE Concrete	0
ANSI/ANS Standard 56.8-2002	Containment System Leakage Testing Requirements	
GL 81-07	Control of Heavy Loads	
	Report of Concrete Cylinder Test	10/16/06
	Report of Liner Repair Gouge Repair Leak Chase Pressure Test	9/29/06
	Meeting Minutes and Action Plan for Pressurizer Heater Investigation	

Number	Title	Revision
TS 25036-C-322(Q)	Technical Specification for Placement of CQE Concrete	
	Report of the Nuclear Safety Review Group titled "Independent Review of the 2006 Refueling Outage Schedule	
	Briefing Outline for Operations Staff: "Draining the RCS to Mid-Loop,"	09/2006
	Briefing Package for Plant Review Committee regarding containment voids	11/21/06
TDB, Section III.20	Reactor Vessel Elevation View	17
USAR Section 8.5	Initial Cable Installation Design Criteria	9
Purchase Order 00098992	Ready Mix Concrete	
	Kleinfelder Report of Cylinder Test, ASTM C39, sample taken on 10/07/06	
	Letter from Jerome Snyder (Westinghouse) to Richard W. Bradshaw (OPPD), "Fort Calhoun Station – Weight and Center of Gravity of ORVH and RRVH Assemblies for Handling Purposes	3/6/06
	Letter from T. Kusakabe (MHI) to J. M. Cate (OPPD), "Project Fort Calhoun Replacement Steam Generator (Contract No. 00056373) Subject Revised SG Shipping Weight,"	3/31/06
	Bechtel Response to Review of Topical Report TR5-45 'Review of Crane, Lifting, Hoisting, and Rigging Related Events'"	3/3/06
	Fort Calhoun Nuclear Power Station NSSS Refurbishment Project Project No. 24903 Task 03 Rigging and Handling Study	
	Letter BPC/FCS-03-0136, Enclosure 1, "Licensing Issue: Rigging and Handling of Heavy Loads	
	Memorandum Jerome Snyder to Richard W. Bradshaw, "Weight and Center of Gravity of ORVH and RRVH Assemblies for Handling Purposes,"	10/26/05
	Letter SGR-FC/BPC-05-0178, "Rigging Operating Experience Review" from R. W. Short (OPPD) to Mr. A. T. Morrow (Bechtel	

Number	Title	Revision
WPIR 25036	The old steam generator (OSGA) removal from the containment bldg	0
USAR Section 14.24	Safety Analysis Heavy Load Incident	13
	Letter from A. C. Thadani (NRR) to R. L. Andrews (OPPD), "Control of Heavy Loads, NUREG-0612, Update of Phase Two Report,"	12/20/06
	Letter from R. L. Andrews (OPPD) to E. J. Butcher (NRR), "Control of Heavy Loads, NUREG-0612"	11/14/85
L5-04FN313	Fort Calhoun Unit 1 Replacement Steam Generators, Basis for Calculation of Weight and Center of Gravity for the Steam Generators	0
DS-ME-03-4	Design Specification for Replacement Pressurizer for Omaha Public Power District, Fort Calhoun Station	4
Contract 62224, Section "H"	Specification for Replacement Pressurizer for Omaha Public Power District Fort Calhoun Station	0
Contract 58681, Section "H"	Certified Design Specification for Replacement Reactor Vessel Head for Omaha Public Power District Fort Calhoun Station	2
NUREG 0612	Control of Heavy Loads at Nuclear Power Plants	07/1988
V03-008	Quality Assurance Audit Repot of Mitsubishi Heavy Industries, Ltd. June 30 - July 10, 2003	
WO 00175901	Polar Crane Inspection and Preventive Maintenance	1
WO 00208678	HE-1 Polar Crane Inspection and Preventive Maintenance	1
ASME Code III, NB-2000	Materials	1989 Edition, no addenda
N-7624-10	Manufacturing Specification for Closure Head Forging	1
L5-01DQ401	Purchase Specification - Closure Head	1
L5-01DQ402	Purchase Specification - CEDM Penetration Material	1
KBS-20040645	Replacement Reactor Vessel Head for Omaha Public Power District Fort Calhoun Station - Material Properties of ICI/CEDM Penetration	2

Number	Title	Revision
N-474-2	Design Stress Intensities and Yield Strength values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi, Class 1 Section III, Division 1	12/9/1993
V03-008	Nuclear Procurement Issues Committee - Audit Checklist - Steam Generators, Reactor Vessel Head, and components	10
N-525	Design Stress Intensities and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 30 ksi, Class 1 Components, Section III, Division 1	12/9/1993
N-20-4	SB-163, Cold Worked UNS N08800; and SB-163 UNS N0660, UNS N06690, and UNS N08800 to Supplementary requirements S2 of SB-163, Class 1 Section III, Division 1	2/26/1999
KBS-20040645	Replacement Reactor Vessel Head for Omaha Public Power District Fort Calhoun Station Material Properties of ICI/CEDM Penetration	
JQA-04-076	Certified Material Test Report - Reactor Vessel Closure Head for Fort Calhoun Station Unit 1	5/17/04
UES-69-030012	Quality Assurance Program (Project Addenda), Omaha Public Power District Fort Calhoun Nuclear Station Reactor Vessel Head Replacement Project	1
UGW-69-030009	ASME Code Job List of Qualified NDE Personnel	21
SBB-FCVH-M002	Omaha Public Power District Fort Calhoun Station Replacement Reactor Vessel Head Postweld Heat Treatment Procedure	2
WDI-PJF-1303221-FSR-001	Fort Calhoun Unit 1 OPD Replacement Reactor Vessel Closure Head Pre-Service Inspection Final Report	11/2005
OFN-RVH-D-11	Dimensional Verification of RRVH Penetration Position Before Final Machining	1
SBR-FCSG-W002	Fort Calhoun Unit 1 - Replacement Steam Generators - Welding Performance for WPS	8
SBB-FCRVH-M004	Omaha Public Power District Fort Calhoun Station Replacement Reactor Vessel Head Manufacturing Process Flow Diagram	1
UGS-L5-030098	Manufacturing and Inspection Plan	3

Number	Title	Revision
SA-508/SA-508M	Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels	1998
N-460	Alternative Examination Coverage for Class 1 and Class 2 welds; Section XI, Division 1	7/27/88
RFP 00001034	Certified Design Specification For Replacement Steam Generators; Section H Technical Specification	2
SBR-FCSG-W003	List of Qualified Welders and Welding Operators	10
SBR-FCSG-W002	Welding Performance for WPS	6
51-5031755-02	FCS RSG - USAR Events Disposition	5/23/05
	Letter from Alan B. Wang (NRR) to R. T. Ridenoure (OPPD), Fort Calhoun Station, Unit No. 1 - Issuance of Amendment Re: Use of Areva NP Inc. Realistic Large Break Loss-of-Coolant Accident Methodology (TAC NO. MC8946)	11/3/06
MM-RR-RC-0305	Removal of Reactor Vessel Closure Head, Hold Down Ring, and Upper Guide Structure	25
ACI-347R-94	Guide for Formwork for Concrete	3/1/94
ACI-306R-88	Cold Weather Concreting	1989

Drawings		
Number	Title	Revision
R-SLX-190	Liner Plate Leak Chase Channel Weld Map	3
25036-C-008	Buried Utilities Composite Plan	0
25036-C-048	Liner Plate Rigging Sequence - Removal	0
25036-C-031	Temporary construction opening reinforcing restoration details	0
845-J87	NSSS Refurbishment Project seal opening	
L5-01DQ201	Reactor Vessel Closure Head	3
L5-01DQ155	Replacement Reactor Vessel Closure Head Repair Weld Map	0
L5-01DQ110	Replacement Reactor Vessel Closure Head CEDM Nozzle ½	5

Number	Title	Revision
L5-01DQ111	Replacement Reactor Vessel Closure Head CEDM Nozzle 2/2	5
L5-01DQ111	Replacement Reactor Vessel Closure Head Incore Instrumentation Nozzle ½	3
L5-01DQ111	Replacement Reactor Vessel Closure Head Incore Instrumentation Nozzle 2/2	3
L5-01DQ153	Replacement Reactor Vessel Closure Head Weld Map	1
L5-04FJ211	Marking of Tube Sheet	1
L5-04FJ201	Tube Sheet	3
L5-04FJ202	Tube Sheet	3
L5-04FJ402	Tube	2
L5-04FJ203	Tube Sheet	6
L5-04FJ401	Tube	1

Quality Assurance Data Package Sub-Index	
Replacement Reactor Vessel Upper Head	
Number	Title
P2.12	Heat Treatment Records
P3	Certificate of Conformance
P2.7	MICL & CMTR of Base Materials (Safety Related Items)
P2.13	Final NDE Records
P2.16	Dimensional Inspection Record
P2.15	Surface Examination Records after Hydrostatic Test
P2.9	MICL & CMTR of Welding Materials <Safety>
P4	Welding Procedure Specifications (WPS)
P3.4	Procurement Specifications
P2.18	Non-Conformance Report (A) and (B) and Repair Records
P2.14	Hydrostatic Test Records
P2.17	Cleanliness and Foreign Material Inspection Records

Replacement Steam Generators	
Number	Title
P3.1	List of Applicable Codes and Standards, All the Effective Deviation Requests
P2.24	Pre-Service Inspection Reports
P3.3	Design Report
P2.3	MHI Certificate of Conformance
P2.4	Copy of Certificate Holder's Code Data Report
P2.11	Welding Records <Safety Related Items>
P2.12	Heat Treatment Records <Time-Temperature Records>
P2.13	Final NDE Records
P4	Welding Procedure Specifications (WPS)

Condition Reports

OPPD:

2006-01036
2006-01655
200604043
200604073
200604106
200604305
200604327
200604434
200604723
200604945

Palo Verde:

CRDR 2914811

Nonconformance Reports

Bechtel:

06-0051
06-0053
06-0045
06-0026
06-0044
06-0046

OPPD:

UGNR-OFN-RPR-027
UGNR-OF1-SG096
UGNR-OFN-RVH-016
UGNR-OFN-RPH-001
UGNR-OF1-SG065
UGNR-OF1-SG028
UGNR-OF1-SG017
UGNR-OF1-SG005
UGNR-OF1-SG001
UGNR-OF1-SG002
UGNR-OF1-SG003
UGNR-OF1-SG004
UGNR-OF1-SG005
UGNR-OF1-SG009

Field Change Requests

FCR C-0076, Rev. 0	FCR C-0113, Rev 3	FCR C-0168, Rev 4
FCR C-0082, Rev A	FCR C-0152, Rev A	FCR C-0179, Rev A
FCR C-0083, Rev A	FCR C-0156, Rev A	FCR C-0185, Rev 4
FCR C-0085, Rev 0	FCR C-0161, Rev A	FCR C-0189, Rev A
FCR C-0086, Rev A	FCR C-0164, Rev A	FCR C-0190, Rev A
FCR C-0088, Rev A	FCR C-0167, Rev 0	FCR C-0193, Rev A
FCR C-0111, Rev A		

Quality Surveillance Observations

JFE Steel

123	253	428	526	641	660
134	259	429	640	659	598
245					

Japan Steel Works

52	32	130	249	365	329
55	33	169	261	366	330
57	86	208	303	367	331
20	91	213	305	507	393
31	107	230	306	717	394