UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

July 5, 2012

David J. Bannister, Vice President
and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION – NRC POST-APPROVAL LICENSE RENEWAL INSPECTION REPORT 05000285/2012009

Dear Mr. Bannister:

On June 7, 2012, U.S. Nuclear Regulatory Commission inspectors performed a Post-Approval Site Inspection for License Renewal at your Fort Calhoun Station. The enclosed report documents the inspection findings, which were discussed on June 7, 2012, with you 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based upon 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, and its enclosure, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Geoffrey Miller, Chief
Engineering Branch 2
Division of Reactor Safety

Docket: 50-285
License: DPR-40

cc w/Enclosure:
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R:\ _REACTORS\FCS\2012\FCS 2012009 PRI-GAP
ADAMS Accession No.: **ML12187A068**

SUNSI Rev Compl.	■ Yes ☐ No	ADAMS	■ Yes ☐ No	Reviewer Initials	GAP
Publicly Avail	■ Yes ☐ No	Sensitive	☐ Yes ■ No	Sens. Type Initials	GAP

DRS/EB2	DRS/EB2	DRS/EB2	C:DRS/EB	C:DRP/F
GPick	SMakor	CSpeer	GMiller	JClark
/RA/	**/RA/**	**/RA/**	**/RA/**	**/RA/**
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2012009
Applicant:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	P.O. Box 310 Fort Calhoun, NE 68023
Dates:	June 4 – 7, 2012
Inspectors:	G. Pick, Senior Reactor Inspector S. Makor, Reactor Inspector C. Speer, Reactor Inspector
Approved By:	Geoffrey Miller, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000285/2012009; 06/04 – 06/07/2012; Fort Calhoun Station, Post-Approval Site Inspection for License Renewal

The report covers an inspection conducted by regional inspectors in accordance with NRC Manual Chapters 2515 and NRC Inspection Procedure 71003.

The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

None

B. Licensee-Identified Violations

None

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other - Post-Approval Site Inspection for License Renewal (Phase 1) – IP 71003

The inspectors reviewed a sample of license renewal activities scheduled in June 2012 while the plant was in an extended shutdown. The inspectors selected this period because it allowed an opportunity to evaluate inaccessible areas prior to entry into the period of extended operation. The period of extended operation is the additional 20 years beyond the original 40-year licensed term. The period of extended operation begins after midnight on August 9, 2013.

Specific areas walked down during this inspection included:

- Manholes 5 and 31
- Containment steam generator bays
- Chemical and volume control system ion exchanger room
- Chemical and volume control system valve gallery
- Regenerative heat exchanger room
- Shutdown cooling heat exchanger rooms
- Spent fuel pool cooling and heat exchanger rooms
- Review of underwater inspections of the discharge tunnel

NRC inspectors performed this inspection to evaluate whether the licensee: (1) completed the necessary actions to comply with the license condition and commitments that are a part of the renewed operating license; (2) implemented the aging management programs and time-limited aging analyses as described in the updated final safety analysis report; (3) followed the guidance in NEI 99-04, “Guidelines for Managing NRC Commitment Changes,” for changing license renewal commitments; (4) identified, evaluated, and incorporated “newly identified” structures, systems, and components into their aging management programs; (5) implemented programs that agreed with those approved in the safety-evaluation report and described in the updated final safety analysis report; and (6) implemented operating experience review and corrective action programs that account for aging effects.

.01 Review of Commitments

a. Scope

The inspectors evaluated whether the licensee met the commitments listed below, as described in NUREG-1782, “Safety Evaluation Report (SER) Related to the License Renewal of Fort Calhoun Station, Unit 1.” The inspectors verified that the licensee implemented procedures, documented inspection results, initiated corrective action documents, and provided training to implementing personnel.

The inspectors reviewed supporting documents including implementing procedures, work orders, inspection reports, engineering evaluations, and condition reports; conducted interviews with licensee staff; observed in-process outage activities; and performed visual inspection of structures, systems, and components including those not accessible during power operation to verify that the licensee completed the necessary actions to comply with the license conditions stipulated in the renewed facility operating license.

NUREG-1782 and the updated final safety analysis report supplement lists 40 commitments made during the license renewal application process. During this inspection, the inspectors reviewed 9 of the 40 commitments and closed 7 commitments.

Specific documents reviewed are listed in the report attachment.

b. Findings and Observations

1. Commitment 1

Commitment 1 specified, "The Alloy 600 Program is a new program at FCS. With this being the case, inspection methodologies for all of the components in the program have not yet been determined. Some of the components that are in the program are currently part of other programs like the reactor vessel internals inspection program. The activities that occur under the interfacing programs relative to these components will be utilized to help analyze and determine the methodologies to be incorporated within the Alloy 600 program for inspection of its included components. These analyses and determinations will be completed prior to entry into the period of extended operation."

During Refueling Outage 23, the licensee replaced the steam generators, the pressurizer, and the reactor pressure vessel head. All of the associated Alloy 600 nozzles/penetration locations and Alloy 82/182 butt welds (the welds susceptible to primary water stress corrosion cracking) were eliminated on these components since they were manufactured and welded with Alloy 690 material. The inspectors determined that the licensee included the new welded joints/materials in their inservice inspection program. The licensee will examine the welds in accordance with ASME, Section XI, IWB-2500 requirements. After the first operating cycle with the new Alloy 690 reactor vessel head, the licensee performed a general visual examination of the head and over-head components without removing insulation. Additionally, the licensee conducted a visual examination (VT-2) of the reactor vessel head-to-flange joint.

During Refueling Outage 24, the licensee determined that the reactor coolant system hot leg and cold leg nozzles contained Alloy 82/182 nozzle-to-safe-end butt welds. The licensee included these components and welds in the Alloy 600 materials program. The licensee performed Materials Reliability Program 139, "Primary System Piping Butt Weld Inspection and Evaluation Guideline," baseline inspections on the dissimilar metal butt welds in November 2009 (Refueling Outage 25). The licensee did not identify any indications or cracking during these examinations. The inspectors verified that the

Alloy 600 program met the requirements of the NRC-accepted Materials Reliability Program 139. The inspectors verified that NRC inspectors had reviewed licensee inservice inspection activities of the reactor vessel head and steam generators in 2008 and had reviewed the dissimilar butt weld inspection records in 2009. NRC inspectors review these inservice inspection activities as part of the baseline inspection activities specified in Inspection Procedure 71111.08.

Based on review of the timeliness and adequacy of the actions implemented, the inspectors determined that the licensee met the conditions of Commitment 1.

2. Commitment 2

Commitment 2 specified, "There is to be a plant-specific program, the Alloy 600 program, for the aging management of Inconel 182 welds. The details of this program are still in development but will be completed prior to the period of extended operation."

The inspectors interviewed the Alloy 600 program owner and reviewed the program documents. The inspectors verified that the licensee established an Alloy 600 program that met the requirements of Materials Reliability Program 139. The program included participation in industry programs to determine an appropriate aging management program for stress corrosion cracking of Alloy 600 materials and primary water stress corrosion cracking of Inconel 182 welds. The inspectors concluded that the Alloy 600 program will effectively manage aging in the components, and systems will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

Based on review of the timeliness and adequacy of the actions implemented, the inspectors determined that the licensee met the conditions of Commitment 2.

3. Commitment 3

Commitment 3 specified, "The flow skirt is one of those components currently included under the scope of the reactor vessel internals inspection program and the Alloy 600 program. Exactly how the flow skirt is to be managed under the Alloy 600 program is yet to be determined; however, that determination will be made before entry into the period of extended operation."

The inspectors reviewed program procedures, the program bases documents related to the Alloy 600 program and the reactor vessel internals. Both program procedures discussed and included the flow skirt in their scope. The licensee divided the examination of the reactor vessel internals into three major sections (1) the core support barrel (including the lower core support structure, the core shroud and the thermal shield), (2) the upper guide structure (including the control element assembly shrouds and the in-core instrumentation guide tubes) and (3) the flow skirt. The reactor vessel flow skirt consists of a ring and flange constructed of Alloy 600 plate welded to the bottom head of the vessel. The flow skirt is a perforated cylinder designed to provide a uniform flow distribution and to minimize the pressure loads.

Although the licensee added the flow skirt to the scope of the Alloy 600 program, the licensee conducts the examinations of the flow skirt as part of the Reactor Vessel Internals Inspection program. The Reactor Vessel Internals Inspection program conducts the reactor vessel internals examination activities related to Materials Reliability Program-227, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The Materials Reliability Program-227 augmented inspection requirements are applicable to Fort Calhoun and will become part of the Reactor Vessel Internals Inspection Program. Corrective actions for augmented inspections will be developed and will use repair and replacement procedures equivalent to those requirements specified in Section XI of the ASME Boiler and Pressure Vessel Code.

The inspectors verified that these evaluations will continue to be reviewed as part of baseline inspection activities as specified in Inspection Procedure 71111.08.

Based on review of the timeliness and adequacy of the actions implemented, the inspectors determined that the licensee met the conditions of Commitment 3.

4. <u>Commitment 4</u>

Commitment 4 specified, "Develop the Alloy 600 program which reflects the program elements of GALL AMP XI.M11, and other commitments in response to the NRC staff's review. An assessment of Alloy 600 and Alloy 82/182 components has been performed and incorporated into the Alloy 600 program basis document. The assessment provided conclusions and recommendations to address the specified components...These recommendations will be evaluated as part of the Alloy 600 program and implemented as necessary to ensure the reliability of the Alloy 600 and Alloy 82/182 components. The applicant will incorporate appropriate information from its responses to Generic Letter 97-01, *"Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997*, and NRC Bulletins 2001-01, '*Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,' dated August 3, 2001;* 2002-01, *'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,' dated March 18, 2002*; and 2002-02, '*Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs,' dated August 9, 2002*."

The inspectors interviewed the Alloy 600 program to discuss monitoring of industry operating experience and implementation of program enhancements as necessary. The inspectors determined the licensee proactively processed operating experience related to primary water stress corrosion cracking of Alloy 600. Specifically, the licensee

- Evaluated the applications of Alloy 600, Alloy 82 and Alloy 182 in the reactor coolant system in response to NRC Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600;"

- Participated in the industry integrated inspection program used to respond to Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetration;" and

- Continued to follow industry developments related to circumferential cracking in control rod drive mechanisms, as described in NRC Bulleting 2001-01.

The inspectors verified that the licensee revised, developed, and updated multiple procedures consistent with the requirements of the Generic Aging Lessons Learned Report, Section XI.M11, "Nickel-Alloy Nozzles and Penetrations." The inspectors determined that the licensee reviewed fabrication records of Alloy 82/182 welds and Alloy 600 components for evidence of rework in response to industry experience.

The inspectors verified that the licensee revised the updated final safety analysis report as their site program changed. For example, the licensee submitted a request letter that deleted the J-Weld inspection. The inspectors determined that these activities were reviewed as part of baseline inspection activities as specified in Inspection Procedure 71111.08.

Based on review of the timeliness and adequacy of the actions implemented, the inspectors determined that the licensee met the conditions of Commitment 4.

5. Commitment 5

Commitment 5 specified, "OPPD's response to RAI B.3.1-1 also states that the FCS Alloy 600 Program currently includes a requirement to monitor industry operating experience and implement program enhancements as necessary. By making this a requirement of the Alloy 600 Program, OPPD has committed to incorporating industry activity recommendations or mandates as applicable."

The inspectors reviewed the actions implemented for this commitment and determined that the actions taken for Commitment 4 also addressed Commitment 5.

Based on review of the timeliness and adequacy of the actions implemented, the inspectors determined that the licensee met the conditions of Commitment 5.

6. Commitment 17

Commitment 17 specified "OPPD has incorporated an augmented inspection of the thermal shield bolting or pins within the Reactor Vessel Internals Inspection Program. OPPD continues to monitor thermal shield vibrations as a task within the Reactor Vessel Internals Inspection Program (B.2.8)."

The inspectors reviewed the program basis document, implementing procedures, and engineering analyses related to reactor vessel internals vibration.

The inspectors verified that the licensee used visual and ultrasonic tests to identify whether any age-related degradation occurred in the reactor vessel internals. In addition, the licensee collected vibration data quarterly and had a third party analyze the data for the presence of abnormal vibrations.

This was an existing program that had previously been successful in identifying wear in the thermal shield positioning pins. The licensee identified abnormal vibration data from 1988 through 1990 that indicated loosening of thermal sleeve positioning. During the 1992 refueling outage, the licensee visually inspected the support lugs and the positioning pins and analyzed the preload on 11 of the 16 lower positioning pins. Based on the results, the licensee replaced seven lower and four upper pins, which returned the vibrations to normal.

Based on review of the timeliness and adequacy of the actions implemented, the inspectors determined that the licensee met the conditions of Commitment 17.

7. Commitment 28

Commitment 28 required that the licensee add the following specific criteria to procedures and instructions related to the overhead cranes:

> "Specific guidance will be added to applicable inspection procedures to inspect for degradation of expansion anchors and surrounding concrete.
>
> Specific guidance will be added to applicable inspection procedures to identify acceptance criteria for general corrosion and degradation of expansion anchors and surrounding concrete.
>
> Specific guidance will be added to applicable inspection procedures to initiate FCS corrective action documentation if excessive general corrosion or cracking of concrete around expansion anchors is identified."

The inspectors verified that the licensee had included the appropriate cranes for monitoring the effects of aging. The inspectors reviewed the procedures and maintenance instructions related to the overhead cranes and determined that the licensee had not incorporated the requirements into all of the procedures. The inspectors discussed the deficiencies related to each procedure or instruction with the mechanics that implemented the inspections and developed the instructions. The licensee documented the deficiencies and initiated Condition Report 2012-05389.

The inspectors will complete the review of this area and verify the corrective actions during a future inspection.

8. Commitment 32

Commitment 32 specified that "OPPD will perform a one-time inspection of the circulating water discharge tunnel per the structures monitoring program (B.2.10). The circulating water discharge tunnel will be included within the scope of license renewal as part of the intake structure." The safety evaluation report and updated final safety analysis report identified that the inspections would occur prior to entering the period of extended operation.

The inspectors reviewed this area by reviewing the underwater video of the inspection, conducting interviews with personnel, and reviewing documents.

The licensee indicated that they planned to drain the discharge canal and perform a 100 percent one-time inspection of the structure. Because of time needed for planning and logistics, the licensee decided to conduct the inspection after entering the period of extended operation. The licensee had performed a 10 CFR 50.59 evaluation that justified allowing the inspection to occur until after they entered the period of extended operation. The inspectors questioned the licensee about performing the inspection during the ongoing extended shutdown.

Following questioning by the inspector, the recovery team, who was methodically reviewing previously completed 10 CFR 50.59 evaluations, selected this 10 CFR 50.59 evaluation for independent review. The licensee determined that the 10 CFR 50.59 failed to recognize that License Condition 3.E required that the inspection be completed prior to entering the period of extended operation. Any change that did not meet this time frame would require NRC approval for the change. Consequently, the licensee performed underwater inspections prior to entering the period of extended operation. Divers evaluated the condition of approximately 10 percent of the interior surface area of the discharge tunnel. The divers reviewed low flow areas, areas that had direct impingement of flow, and one area that had the expected flow midway down one wall.

The inspectors reviewed the areas selected and identified no concerns. The inspectors determined that underwater inspections of similar structures at other facilities had been performed as a representative sample. At the end of this inspection, the licensee was evaluating whether to conduct periodic inspections of the discharge tunnel as they do other structural inspections or to perform a 100 percent inspection after entering the period of extended operation.

The licensee documented the inadequate 10 CFR 50.59 evaluation in Condition Report 2012-03113. The inspectors reviewed the examples in the Enforcement Manual and determined that no violation of 10 CFR 50.59 or other regulatory requirements resulted because the licensee completed the task prior to entering the period of extended operation, as committed.

Follow-up and evaluation of the resolution for conducting inspections of the discharge tunnel is an unresolved item: Unresolved Item 05000285/2012009-01, Evaluation of Future Discharge Tunnel Inspection Plans

9. Commitment 40

Commitment 40 specified “OPPD will submit to the NRC a license amendment request containing the fracture mechanics evaluation of the small-bore instrument nozzle J-weld region at the repaired instrument nozzle in the side of the pressurizer lower shell. This evaluation will include bounding of the flaw size by the size of the j-weld itself, and addressing the possibility of corrosion in the presence of a flaw.”

The licensee replaced the pressurizer in 2006. The inspectors verified that the new pressurizer had stainless steel welds instead of Alloy 600 weld metal. Consequently, the inspectors determined that Commitment 40 no longer applied. The licensee described the plant conditions and withdrew the commitment in Letter LIC-10-0083, "Deletion of License Renewal Commitment Pertaining to the Fracture Mechanics Evaluation of Pressurizer Lower Shell, Small-Bore Instrument Nozzle J-weld Region."

Based on the revised material configuration of the pressurizer and the commitment withdrawal letter, the inspectors determined that the aging effect no longer applied and the licensee took appropriate actions in withdrawing Commitment 40.

.02 Evaluation of Aging Management Programs and Time-Limited Aging Analyses

a. Scope

As part of the review of aging management programs and time-limited aging analyses approved by the NRC in the safety evaluation report, the inspectors reviewed their updated final safety analysis report descriptions to confirm the implemented programs were consistent with the safety evaluation report. The inspectors evaluated whether the aging management programs addressed any prior inspection items and included the information required by the commitments. The inspectors determined whether the licensee established an appropriate basis to add or remove structures, systems, or components from the aging management programs.

The inspectors evaluated whether the licensee properly included any information in the updated final safety analysis report supplement since the date the renewed license was issued.

The inspectors reviewed supporting documents including implementing procedures, work orders, inspection reports, engineering evaluations, and condition reports; conducted interviews with licensee staff; observed in-process outage activities; and performed visual inspection of structures, systems, and components including those not accessible during power operation to verify that the licensee completed the necessary actions to comply with the license conditions stipulated in the renewed facility operating license.

b. Observations and Findings

1. B.1.7 Reactor Vessel Integrity

The licensee established this aging management program, consistent with the GALL Report, to monitor the embrittlement of the reactor pressure vessel resulting from neutron irradiation. The affected components include the reactor vessel plate material and the reactor vessel welds. The program consists of a revised withdraw schedule for the reactor vessel surveillance capsules, an integrated surveillance program including the surveillance results from other commercial reactor vessels made of similar materials, and revised pressure-temperature limits.

The team reviewed the reactor surveillance capsule schedule change request, the upper-shelf energy evaluation analysis, the pressure-temperature limit request, and supplemental documentation. The team also interviewed the responsible program engineer.

The team concluded that the applicant had implemented this aging management program in accordance with their license renewal application and the updated final safety analysis report supplement.

2. B.2.10 Structures Monitoring

The inspectors performed the walk downs of inaccessible areas looking for signs of aging such as corrosion on piping and supports, corrosion of cable trays, water intrusion, cracking and spalling of concrete.

Manholes 5 and 31

During the inspection of Manhole 31, the inspectors determined that the cables were routed in conduit. From discussions with the licensee, Manhole 31 is designed to drain to Manhole 5. The inspectors determined that each cable conduit dipped as it passed through the manhole and determined that any water trapped in the conduit had no way to flow into Manhole 5 as designed. The inspectors determined that when Manhole 5 floods an open path exists for water to travel into the conduit and become trapped in the conduit low spots in Manhole 31. The licensee documented this deficiency in Condition Report 2012-04997. Licensee records indicated that Manhole 5 had filled three times in 14 years (since 1998) since they began to keep records.

The inspectors noted that corrective action documents described that the cables could be wet. The inspectors will review the specific qualifications of these cables and confirm whether the cables were qualified to be submerged. In addition, the inspectors determined that the licensee specified they would address water intrusion into the manholes prior to startup from the extended shutdown in response to Notice of Violation 05000285/2011006-02.

The inspectors will review corrective actions implemented by the licensee and evaluate the impact on the cables resulting from wetting during the commitment inspection. This is considered an unresolved item: Unresolved Item 05000285/2012009-002, Impact of Wetting on Safety-Related Cables.

Steam Generator Bays A and B

During containment walk downs in the steam generator bays, the inspectors verified the presence of a significant amount of boric acid on the bioshield walls in both steam generator bays outside of the reactor compartment near both the hot and cold legs. Because of the presence of insulation, the inspectors could not identify the condition of the piping and structures inside the bioshield. The licensee indicated the boric acid came from the refueling cavity liner when flooded up during refueling outages. The

licensee had initiated a task team to resolve this condition and documented various deficiencies in roll-up Condition Report 2012-00116. The licensee initiated Condition Report 2011-05763 that documented the presence of an eight foot long horizontal crack above the reactor cavity liner leak-off line in the reactor cavity wall. The inspectors noted that the licensee had identified a similar deficiency in 2003, as documented in Condition Report 2003-04261.

The licensee concluded the leakage through the crack did not affect the wall structurally based on no evidence of degraded reinforcing steel, the amount of reinforcement, and the thickness of the wall. The inspectors confirmed that Part 9900, "Operability Determinations and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety," Section C.13 specified that degraded structures remain operable so long as the degradation does not result in exceeding any acceptance limits specified in the codes and standards specified in the design basis. The licensee also compared the trends of reactor cavity liner leak-off line flow rates to previous outages. The review of trends confirmed that leakage remained constant and low for the last three outages. The inspectors verified no brown discoloration that would indicate wastage of reinforcing steel from the crack.

The inspectors questioned the origin of a wetted brown stain on the reactor cavity wall below the insulation around the Steam Generator B hot leg. Consequently, the licensee initiated Condition Report 2012-05067 to determine the cause and take corrective actions. The licensee had previously identified this condition in February 2012, as documented in Condition Report 2012-01339, but had not taken any corrective actions.

The inspectors will review the licensee actions to identify and address reactor cavity liner leakage and its impact on the containment structures during a future inspection. This is considered an unresolved item: Unresolved Item 05000285/2012009-003, Assess Corrective Actions and Determine Structural Effect of Water Intrusion and Boric Acid on Interior Containment Walls.

.03 Operating Experience and Corrective Action Programs

a. Inspection Scope

The inspectors reviewed the operating experience program to determine whether the licensee updated aging management programs to account for operating experience issued since the licensee had received the renewed license and any changes to the Generic Aging Lessons Learned Report or other approved topical reports.

The inspectors reviewed the corrective action program to evaluate whether the applicant established a method to evaluate the effects of aging and to identify deficiencies that may have resulted from aging effects.

The inspectors sampled corrective action documents, interviewed personnel, evaluated corrective actions implemented, and reviewed process documents during this inspection.

b. <u>Observations and Findings</u>

The licensee initiated a gap analysis to identify the actual differences that affected their aging management programs as a result of the differences between GALL Report, Revision 0 and Revision 2. At the time of this inspection the licensee had ongoing reviews to identify recommended resolutions to the identified gaps. The inspectors will review the licensee resolution of these differences during a later inspection.

40A6 <u>Meetings, Including Exit</u>

The inspectors presented the inspection results to Mr. Dave Bannister, Vice President and Chief Nuclear Officer and other members of the licensee staff during an exit meeting conducted on June 7, 2012. The licensee acknowledged the NRC inspection observations. The inspectors returned all proprietary information reviewed during this inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Bakhit, Nuclear Engineer
D. Bannister, Senior Vice President and Chief Nuclear Officer
K. Erdman, Engineering Programs Supervisor
R. Giachetti, Principal Engineer – Westinghouse
K. Holthaus, Principal Engineer
B. Huebner, Mechanical Lead
B. Lisowyj, Manager, Material Projects
K. Maassen, Program Engineer
E. Matzke, Compliance Engineer
B. Mierzejewski, Senior System Engineer
M. Prospero, Plant Manger
J. Sweley, System Engineer
P. Turner, Program Engineer

NRC Personnel

J. Kirkland, Senior Resident Inspector
J. Wingebach, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened	Item	Description
05000285/2012009-001	URI	Evaluation of Future Discharge Tunnel Inspection Plans (Section 4OA5.01.b.8)
05000285/2012009-002	URI	Impact of Wetting on Safety-Related Cables (Section 4OA5.02.b.2)
05000285/2012009-003	URI	Assess Corrective Actions and Determine Structural Effect of Water Intrusion and Boric Acid on Interior Containment Walls (Section 4OA5.02.b.2)

Commitments

The inspectors closed Commitments 1, 2, 3, 4, 5, 17, and 40 during this inspection.

The inspectors reviewed Commitments 28 and 32 during this inspection.

DOCUMENTS REVIEWED

General

Condition Records

2009-01244	2009-05727	2011-08072	2011-08076
2010-01538	2010-05729	2011-08074	2012-00700

Audits and Assessments

NUMBER	TITLE	DATE
2009-01244	License Renewal Self Assessment	11/10/2010
2010-01538	Engineering Programs Self-Assessment of License Renewal	08/29/2011

License Renewal

NUMBER	TITLE	REVISION/DATE
NUREG-1782	Safety Evaluation Report Related to the License Renewal of Fort Calhoun Station	10/31/2003
USAR 15.1	License Renewal Supplement	5
USAR 15.1	License Renewal Supplement – Introduction	7
USAR 15.2	License Renewal Supplement – Programs and Activities for Managing the Effects of Aging	0
USAR 15.3	License Renewal Supplement – Evaluation of Time-Limited Aging Analyses	0
USAR 15.4	License Renewal Supplement – License Renewal Commitment Listing	0
PBD-29	One-Time Inspection	4
PBD-34	License Renewal	3

Miscellaneous

NUMBER	TITLE	REVISION/DATE
TR-107514	Age-Related Degradation Inspection Method and Demonstration	April 1998
NEI 99-04	Guidelines for Managing NRC Commitment Changes	0

Miscellaneous

TITLE
Commitment list with status of implementation for each commitment
Inspection Report 05000285/2002007
Inspection Report 05000285/2003007
List of activities being performed the week of June 4 credited with aging management
List of outstanding actions related to each commitment

Verified Visual Inspector Qualifications, including eye examinations for the contract

T. Armstrong	J. Clark	L. Hare	A. Mitchell	K. Wittstruck
G. Brannan	P. Donaldson	D. Honse	J. Stapp	J. Zelfel
R. Bunz	D. Freytag	C. McMullen	T. Stahly	

Commitments

Condition Reports

2012-05389*

Letters

NUMBER	TITLE	DATE
LIC-01-0075	Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles	08/31/01
LIC-05-0057	Fort Calhoun Station Unit No. 1, Revised Relaxation Request for First Revised Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors	05/14/05
NRC-05-0065	Fort Calhoun Station, Unit No. 1 – Relaxation Request From U.S. Nuclear Regulatory Commission (NRC) Order EA-03-009 for the Control Element Drive Mechanism Nozzles	05/24/05

License Renewal

NUMBER	TITLE	REVISION/DATE
EA-FC- 00-069	Fuel Handling Equipment and Heavy Load Cranes Scoping, Screening, and Aging Management Review for License Renewal	1
EA-FC-00-135	Reactor Vessel Internals Inspection Program Evaluation for License Renewal	12/16/2005
EA-FC- 00-143	Overhead Load Handling Systems Inspection Program Evaluation for License Renewal	0
PBD-18	Alloy 600	6
PBD-40	Overhead Loads and Handling Equipment	1
PBD-44	Reactor Vessel Internals Inspection	0
PLDBD-CS-52	Heavy Loads	21

Miscellaneous

NUMBER	TITLE	DATE
CFTC-09-51	Transmittal of Neutron Noise Data Analysis Report	07/09/2009
CFTC-11-91	Transmittal of WCAP-17347-NP; PWR Vessel Internals Program Plan for Aging Management of Reactor Internals	03/31/2011

Miscellaneous

TITLE
Inspection Report 05000285/2008003
Inspection Report 05000285/2009005

Preventive Maintenance Items

2655	2657	2658	2659	4315

Procedures

NUMBER	TITLE	REVISION/DATE
IC-PM-IX-1350	Internals Vibration Monitoring Core Support Barrel/Thermal Shield Motion	6
MM-PM-MX-0551	Inspection of Reactor Vessel Closure Head Lift Rig	6
MM-RI-HE-0550	Polar Crane Inspection	32
MM-RI-HE-0551	Annual Inspection of Auxiliary Building Crane	19
MM-RI-FH-0700	Refueling Machine Preoperational Inspection and Maintenance	31
MM-RI-FH-0703	Visual Inspection Spent Fuel Machine Preoperational Inspection and Maintenance	22

Procedures

NUMBER	TITLE	REVISION/DATE
MM-RR-CEDM-1202	Control Element Drive Mechanism Disassembly, Inspection, and Reassembly	12/27/06
OI-RC-3A	Reactor Coolant System Cold Hydrostatic Test	04/13/10
OP-ST-RC-3007	Periodic Reactor Coolant System Integrity	07/18/11
PE-RR-RC-1000	Removal of Reactor Vessel Head Insulation	8
PE-RR-RC-1001	Installation of Reactor Vessel Head Insulation	8
PED-QP-33	Inservice Inspection and Inservice Test Program	8
PED-QP-27	Repair/Replacement Process	16

Work Order

00391448-01

Aging Management Programs

Condition Reports

1998-01719	2009-03728	2010-03776	2011-04163	2012-04997*
2002-00707	2009-03786	2011-04156	2011-04729	2012-05044*
2003-03665	2009-03790	2011-04157	2011-05759	2012-05051*
2003-04261	2009-03957	2011-04158	2011-05761	2012-05067*
2005-03247	2009-03967	2011-04159	2011-05792	2012-05070*
2008-02829	2009-04216	2011-04161	2012-00116	2012-05389*
2009-03077	2009-04619	2011-04162	2012-03113*	

*indicates CR was written as a result of this inspection

Drawings

NUMBER	TITLE	REVISION
11405-S-275	Piling Plan Intake Structure	6

Drawings

NUMBER	TITLE	REVISION
11405-S-311	Intake Structure & Tunnels Foundation Plan & Details	6
11405-S-317	Intake Structure & Tunnel Sections & Details	6
11405-S-320	Intake Structure & Tunnels Miscellaneous Details II	3

Letters

NUMBER	TITLE	DATE
LIC-98-0124	Response to Request for Additional Information Related to Generic Letter 92-01, Revision 1, Supplement 1	09/28/1998
LIC-01-0018	Supplemental Information to Support an Application for Amendment of Operating License	02/14/2001
LIC-01-0107	Reactor Vessel Surveillance Capsule Removal Schedule Change Request	11/08/2001
LIC-01-0114	Fort Calhoun Station Unit No. 1 License Amendment Request ‘Pressure and Temperature (P-T) Limit Curve for 40 Effective Full Power Years (EFPY)	12/16/2001
	Letter from NRC (A.B. Wang) to OPPD (S.K. Gambhir), "Fort Calhoun Station, Unit No. 1 -Issuance of Amendment -Deletion of Section 3.D "License Term"	06/07/2001

License Renewal

NUMBER	TITLE	REVISION
EA-FC-01-024	Upper Shelf Energy (USE) Evaluation for 2033 Operation	0

Miscellaneous

TITLE

Discharge Tunnel White Paper, dated April 10, 2012

Engineering Change EC 55478, Change USAR 15.4 Implementation Schedule, dated March 16, 2012

Information Notice 2002-012, Submerged Safety-Related Electrical Cables, dated March 21, 2002

Letter LIC-12-0041, NRC Inspection Report 05000285/2011006, Reply to a Notice of Violation (NOV); EA-12-035 (Revision 0), dated April 13, 2012

Procedure SE-PM-AE-1002, Intake Building and Miscellaneous Structures Inspection, Revision 9

Program Basis Document PBD 42, Structures Monitoring, Revision 1

System Training Manual - Circulating Water System, Revision 37

Work Orders

00199350-01
00310128-01

Operating Experience and Corrective Action Programs

Miscellaneous

NUMBER	TITLE	REVISION/DATE
Regulatory Information Summary 2011-05	Information on Revision 2 to the Generic Aging Lessons Learned Report for License Renewal of Nuclear Power Plants	07/01/2011
LTR-PEUS-11-32, Volume 1	Fort Calhoun Station Generic Aging Lessons Learned (GALL) Gap Analysis	0