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

May 24, 2013

Louis P. Cortopassi, Site Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

Subject: FORT CALHOUN STATION – NRC SPECIAL INSPECTION REPORT 05000285/2013-012

Dear Mr. Cortopassi:

On April 11, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed a reactive inspection in accordance with NRC Inspection Procedure 93812, "Special Inspection," at your Fort Calhoun Station. This special inspection was conducted to gather information associated with the improper design specifications associated with the raw water pump anchor bolts. Specifically, the inspection was in response to the discovery that unqualified anchor bolts were installed. This inspection report documents the inspection results, which were discussed on April 11, 2013, with you and other members of your staff.

The special inspection commenced on January 28, 2013, in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," and Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors," based on the initial risk and deterministic criteria evaluation made by the NRC on January 4, 2013. The special inspection assessed the cause of the event and your station's actions in response

Six NRC identified findings and one self-revealing finding of very low safety significance (Green) were identified in this inspection. Additionally, two findings with pending significance were identified by the NRC. These issues are pending significance because the NRC recently received additional information from the licensee for review. When the NRC has completed this review the safety significance of these issues will be communicated in a separate correspondence.

Although the issues discussed in this report were characterized as having very low safety significance it is of concern to the NRC the extent of problems identified that relate to various types of design control issues involving component supports. In general, the NRC inspection team found various degrees of design errors with all selected components reviewed. These errors were typically drawing errors, meaning the drawings did not reflect the installed configuration of the plant. Additionally, the inspectors reviewed a number of calculations and found outdated or incorrect design information being used. The NRC understands that your staff is taking both immediate and long term corrective actions to address these concerns. The NRC looks forward to reviewing the adequacy of these actions in the near future.

If you contest any of the non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Fort Calhoun Station.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC Resident Inspector at Fort Calhoun Station.

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 available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management 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/

Michael Hay, Chief
Project Branch F
Division of Reactor Projects

Docket No.: 50-285
License No.: DPR-40

Enclosure:
NRC Inspection Report 0500285/2013-012
w/Attachments
Attachment 1: Supplemental Information
Attachment 2: Special Inspection Charter
Attachment 3: Request for Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2013-012
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	P.O. Box 310 Fort Calhoun, NE 68023
Dates:	January 29, 2013 through April 11, 2013
Inspection Team:	C. Smith, Project Engineer (Team Lead) J. Bozga, Reactor Inspector
Approved By:	Michael Hay, Chief Project Branch F Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2013-012; 01/28/2013 – 03/15/2013; Fort Calhoun Station; Special Inspection Report; IP 93812; Design Control and Corrective Action Violations.

This report covered a 9-day period (January 28 – February 1, and February 19 – February 22, 2013) of onsite inspection, with additional in-office reviews through April 11, 2013. Nine findings 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" dated June 19, 2012. Cross-cutting aspects are determined using IMC 0310, "Components within the Cross-Cutting Areas" dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated January 28, 2013.

A. <u>NRC-Identified and Self Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Green</u>. The inspection team reviewed a self-revealing finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control", for the failure to translate the design basis into instructions, procedures, and drawings. The raw water pump anchor bolt design specifications and calculations incorrectly assumed headed stud cast-in-place anchor bolts instead of the as-built J-style anchor bolts.

  The finding was determined to be more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) because the finding did not ultimately affect the operability or functionality of the raw water pumps. (Section 3.6)

- <u>Green</u>. The inspection team identified a finding of very low safety significance involving the licensee's failure to meet the requirements of the American Concrete Institute (ACI) 349-01. Specifically, the licensee's past functionality calculation failed to ensure the raw water pump anchorage met ACI 349-01 requirements. This finding was entered into the licensee's corrective action program. No violation of NRC requirements was identified.

  The performance deficiency was determined to be more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) because the finding did not ultimately affect the operability or functionality of the anchorage. This finding had a cross-cutting aspect in the Decision-Making component of the Human Performance cross-cutting area because the licensee used non-conservative assumptions in a functionality evaluation of raw water pump anchorage. Specifically, the licensee failed to use strength reduction factors as required by ACI 349-01 in the evaluation of raw water pump anchorage [H.1(b)]. (Section 3.6)

- Green. The inspection team identified a finding of very low safety significance (Green) and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to correct a condition adverse to quality involving raw water system piping stresses that exceeded the allowable stresses. Specifically, since 1995 the licensee was using interim acceptance criteria that placed the piping and pipe supports in a non-conforming/degraded condition for an extended period of time because corrective actions were not implemented or planned. This finding was entered into the licensee's corrective action program.

  The performance deficiency was determined to be more than minor because it was associated with Mitigating System cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the use of interim acceptance criteria placed the RW-111A piping and pipe supports in a nonconforming and degraded condition. The finding screened as of very low safety significance (Green) because it did not ultimately affect the operability or functionality of RW-111A. (Section 3.6)

- Green. The inspection team identified a finding of very low safety significance (Green) and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" for the for the failure to ensure the adequacy of the design for the containment air coolers VA-16A and VA-16B. Specifically, the structural columns of the containment air coolers were subjected to greater than allowable stresses, and were not conservative or in compliance with Class I requirements as defined in Updated Safety Analysis Report (USAR) Section 5.11 and referenced codes.

  The performance deficiency was determined to be more than minor because it was associated with Mitigating System cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as of very low safety significance (Green) because the containment air cooler system was subsequently determined to be operable but degraded. (Section 3.6)

- Green. The inspection team identified a finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the adequacy of the design for Raw Water Pipe Support RWS-117. Specifically, the licensee failed to demonstrate compliance with vendor requirements for the U-bolt of pipe support RWS-117. This finding was entered into the licensee's corrective action program.

The performance deficiency was determined to be more than minor because it was associated with Mitigating System cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the applied stresses exceeded the allowable stress for the U-bolt of pipe support RWS-117. The finding screened as of very low safety significance (Green) because it did not ultimately affect the operability or functionality of pipe support RWS-117. (Section 3.6)

- Green. The inspection team identified several examples of very low safety significance (Green) non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the design basis for pipe supports SIH-17, SIH-94 and SIH-12 was correctly translated into specifications, drawings, procedures, and instructions. Specifically the design calculations were non-conservative with respect to requirements defined by the unistrut concrete insert vendor manual and the calculations did not match the as-built condition.

  The finding was determined to be more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) because the finding did not ultimately affect the operability or functionality of the pipe supports. (Section 3.6)

- Green. The inspection team identified several examples of a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the design basis for all the 480V and 4160V buss switchgear cabinets were correctly translated into specifications, drawings, procedures, and instructions. Specifically, each of the respective switchgear cabinet drawings depicted the equipment secured with concrete anchor bolts, however the cabinets were found secured with welds to an embedded steel plate.

  The finding was determined to be more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low safety significance (Green) because the finding did not ultimately affect the operability or functionality of the electrical switchgear. (Section 3.6)

- TBD. The inspection team identified an apparent violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for the failure to ensure the adequacy of the U-bolts for Containment Air Cooler pipe supports VAS-1 and VAS-2. Specifically the U-bolt design was non-conservative with respect to the design basis requirements. The licensee entered this issue into the corrective action program.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the safety injection tanks and safety injection valves. Specifically, the one-directional U-bolts are not designed to withstand two-directional loading and the failure of the aforementioned pipe supports could adversely impact the safety injection tanks and safety injection valves. The safety significance is pending additional analysis of the as-found configuration of the condensate drain piping line and associated pipe supports. (Section 3.6)

- TBD. The inspection team identified an apparent violation of 10 CFR 50, Appendix B, Criterion III, Design Control, for the failure to ensure the adequacy of the anchorage for the Containment Spray and Raw Water system pipe supports. Specifically the anchorage design was non-conservative with respect to the design basis requirements. The licensee entered this issue into the corrective action program.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the Containment Spray and Raw Water piping system. The safety significance is pending additional analysis of the as-found configuration of the condensate drain piping line and associated pipe supports. (Section 3.6)

B. Licensee-Identified Violations

None.

REPORT DETAILS

1.0 Basis for Special Inspection

On December 2, 2012, Fort Calhoun Station maintenance personnel were preparing to replace a corroded anchor bolt on raw water pump AC-10A when a system engineer discovered engineering drawings for all four raw water pump anchor bolts contained discrepancies. Further investigation revealed that the design basis analyses that certifies the anchor bolts on the raw water pumps assumed a cast-in-place “Type II” (headed stud style) anchor bolts, however the plant installed cast-in-place “Type I” (J-style) anchor bolts.

Additionally, the seismic bracing installed on raw water pump columns was found to be different than the design specifications. The engineering drawing for the seismic bracing specified that tabs be installed with a 30 degree angle bolting holes; however the in-field condition did not include the angled bolting holes in the tabs.

The anchor bolts and seismic braces on the raw water pumps installed were in accordance with the design requirements at the time of construction; however the seismic analyses that qualified the raw water pumps assumed a different, stronger bolt type and this assumption challenged the operability of all four raw water pumps. A preliminary calculation by the licensee concluded that the raw water pumps would fail in a seismic event, and would have been unable to fulfill their safety function.

In accordance with Management Directive (MD) 8.3, this event was evaluated to determine the appropriate level of NRC response to the event. This event met the MD 8.3 deterministic criterion for possible adverse, generic implications for a detailed follow up team inspection. The preliminary Conditional Core Damage Probability for the event was estimated and the NRC determined that the appropriate level of response was a Special Inspection.

The NRC conducted the Special Inspection to better understand the circumstances surrounding the installation of the incorrect anchor bolts and seismic bracing on the raw water pumps, which adversely affected the safety function of the safety related raw water system. The team used NRC Inspection Procedure 93812, “Special Inspection Procedure,” to conduct the inspection. The special inspection team performed field walkdowns, reviewed procedures, corrective action documents, and design documentation. The team interviewed various station personnel, reviewed the licensee’s root cause analysis report, past failure records, extent of condition evaluations, immediate and long term corrective actions, and applicable industry operating experience. A list of documents reviewed is provided in Attachment 1 of this report, and the Charter for the Special Inspection is included as Attachment 2.

2.0 Event Description

The raw water system at Fort Calhoun Station is a once-through cooling system utilizing the Missouri River as the ultimate heat sink. The raw water system has four pumps (designated A, B, C, and D) that provide the motive force to supply Missouri River water to the raw water system. Each pump is vertically installed and located in the intake structure. The associated motor for each pump is secured to the intake structure floor with concrete anchor bolts.

In November 2012, Fort Calhoun Station maintenance personnel started refurbishing the “A” motor for raw water pump AC-10A. On November 28, 2012 FCS maintenance personnel noted that one of the four anchor bolts for the “A” pump motor was corroded, and the bolt/nut combination could not hold the specified torque value. Shortly after, station personnel decided to replace the corroded anchor bolt.

On December 2, 2012, while preparing to replace the corroded anchor bolt on AC-10A, a system engineer referred to drawings to determine if any interferences existed (such as rebar obstructions) associated with drilling out the corroded anchor bolt.

The design drawing 11405-S-312, as shown in Figure 1, did not contain sufficient information about the anchor bolt details or potential interferences.

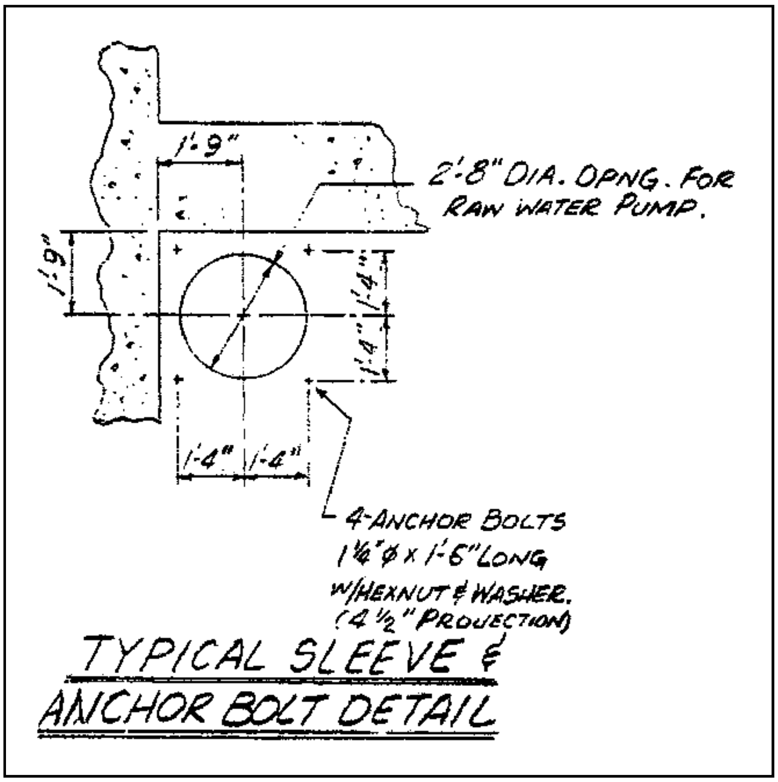
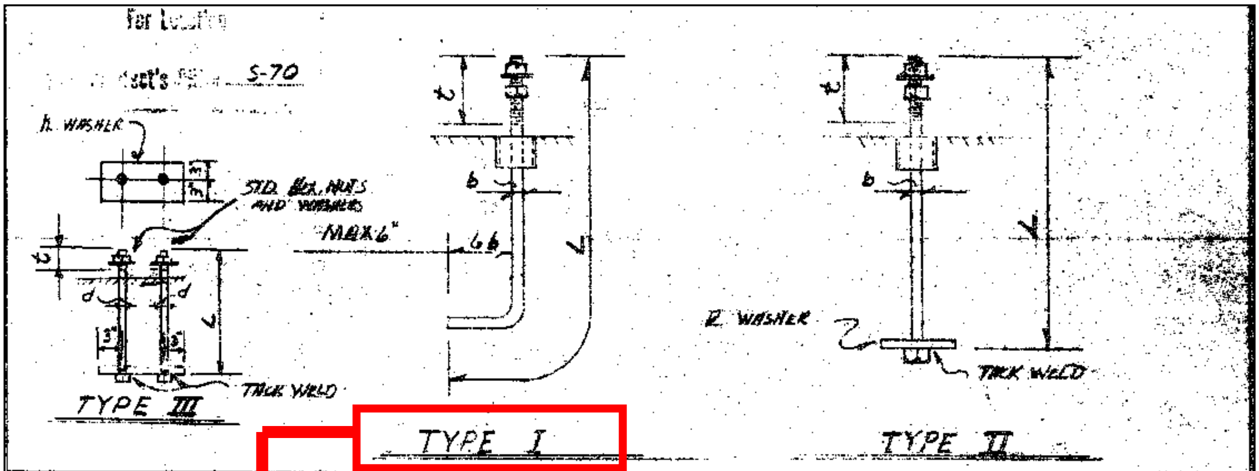


Figure 1 - DWG 11405-S-312 “design” drawing

The engineer then referred to “shop” drawing Number 154 (Figure 2) and the construction “pour” Drawing 7908-C (Figure 3).



ANCHOR BOLT SCHEDULE

MARK	OUR MARK No.	DIA. b	TYPE	No. REQ'D	LENGTH L	TREAD t	R. WASHER	STD. PIPE SLEEVE	LOCATION ON DRAWINGS
AB-30	AB-A154	3/4 ϕ	I	16	2'-6"	2"	–	–	S-54
AB-31	AB-B154	1" ϕ	I	32	2'-9½"	2½"	–	–	S-51 & E-35
AB-32	AB-C154	7/8" ϕ	I	7	1'-8"	2"	–	–	S-53
AB-33	AB-D154	7/8 ϕ	II	4	9"	2"	6x6x3/4	–	S-48
AB-34	AB-E154	5/8 ϕ	I	8	2'-5"	2½"	–	2ϕ x6" Lg	S-51 & E35
AB-35	AB-F154	7/8 ϕ	I	20	~~2'-5"~~	2½"	–	–	S-51 & E35
AB-36	AB-G154	1 ϕ	II	12	2'-6"	3"	6x6x3/4	3ϕ-10" Long	S-48
AB-37	AB-H154	1 ϕ	II	4	9"	2"	6x6x3/4	–	S-48
AB-38	AB-K154	1 ϕ	II	6	1'-0½"	2"	6x6x3/4	3ϕ x 8" Lg	S-48
	AB L154	1¼ ϕ	I	16	1'-6"	4½"	STD WASHER	–	S-312

Figure 2 - DWG 154 “shop” drawing

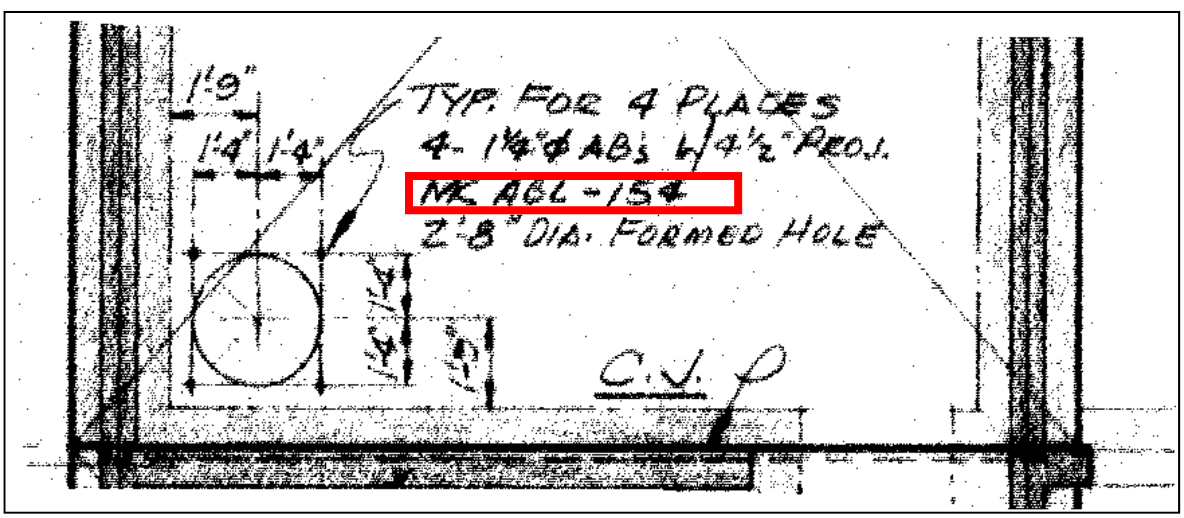


Figure 3 - DWG 7908-C "pour" drawing

The design specification for the raw water pump motors is designated MK ABL-154 as noted in Figure 3. The corresponding anchor bolt data for an MK ABL-154 anchor bolt is found in the "schedule" shown in Figure 2. Clearly, the design specification for the anchor bolts was a Type I J-style anchor bolt.

After reviewing the drawings, the system engineer did not expect that the shop drawing (Figure 2) would specify a Type I, or J-bolt. The J-bolt anchor was unexpected because station personnel assumed that the anchor bolts were Type II, or headed stud anchor bolts, based on design drawing (Figure 1). That incorrect assumption was used for the design basis of the anchor bolts. There was not an original design specification for the raw water pump anchors.

The NRC issued Generic Letter 87-02 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors (USI A-46)" because the older plants were not licensed or designed to a specific seismic standard. Therefore, the Generic Letter required those plants to review the seismic adequacy of certain equipment against seismic criteria not in use when plants were licensed. To review the seismic adequacy, utilities formed the Seismic Qualification Utility Group (SQUG) and developed Generic Implementation Procedure (GIP) as the seismic criteria.

To address the Generic Letter 87-12 requirements, Fort Calhoun Station implemented GIP as the design seismic code of record for the site. The qualification of the anchor bolts was evaluated with the GIP method. However, the station personnel who qualified the seismic design of the raw water pump motor anchor bolts assumed they were headed studs, Type II, and only consulted the design drawing (Figure 1) as a reference. As a result, the subsequent design calculations and seismic certifications for all four raw water pump anchors were incorrect.

After discovery of the error, the station used an ultrasonic probe to confirm the installed anchor bolts were Type I J-bolts on all four raw water pumps (AC-10A, AC-10B, AC-10C, and AC10D). As a result of the testing, engineering personnel performed the GIP analysis for the 7.5 inch J-bolts and concluded they would not meet the design criteria. Subsequently, the licensee declared all four raw water pumps inoperable.

Additionally, the seismic bracing installed on raw water pumps was found to be different than the design specifications. The engineering drawing for the seismic bracing specified that the bracing bolt holes be angled at 30 degrees; however the in-field condition had straight bolt holes. Subsequent investigations determined this did not affect the ability of the seismic bracing to perform its function; it was merely a drawing discrepancy and the angled bolt holes were used for ease of installation. This issue is documented in CR 2012-19097.

The licensee submitted Licensee Event Report (LER) 05000285/2012020-00 on January 31, 2013 for this issue for an event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat. However, revised analyses by the licensee determined that the anchors would remain functional in a seismic event. The LER was updated by the licensee to report a condition which was prohibited by the plant's Technical Specifications.

3.0 Inspection Results

3.1 Timeline (Charter Item 1)

a. Inspection Scope

The team developed and evaluated a timeline of significant events for the installation and design of the anchorage for the raw water pumps. The team developed the timeline, in part, through a review of post-event statements from the on-shift operators, and interviews with system engineers, mechanical maintenance personnel, and members from the root cause evaluation team.

b. Findings and Observations

Timeline of Events Identified by the Team

Some of the entries in the timeline are approximate due to the lack of evidence preservation and lack of precise pre-construction drawing dates by the licensee. The team reviewed a period leading up to the event as well as the day of the event. A brief timeline of post-event actions is provided. The following timeline was developed:

PRIOR TO THE EVENT

1969	Fort Calhoun Station design drawing 11405-S-312 specifies 18 inch anchor bolts with hexnuts and washers for the raw water pumps are “Typical sleeve & anchor bolt detail.” Engineering personnel interpreted this to mean a cast-in-place Type II (headed stud) anchor bolt.
July 1, 1969	“Shop” drawing number 154 specifies “ABL-154” markings for Type I (J-style) anchor bolts.
July 10, 1969	“Pour” drawing 7908-C specifies ABL-154 anchor bolts to secure the raw water pumps.
February 19, 1987	NRC issues Generic Letter 87-02 “Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors (USI A-46).” This Generic Letter concluded that the seismic adequacy of certain equipment in nuclear power plants must be reviewed against seismic criteria not in use when plants were licensed. To review the seismic adequacy, utilities formed the Seismic Qualification Utility Group (SQUG).

October 28, 1993	OPPD SQUG engineers complete analysis for the raw water pump anchor bolts to respond to Generic Letter 87-02. The raw water pump anchor bolt SQUG calculation used design drawing 11405-S-312 as the only input, and incorrectly assumed the anchor bolts were Type II (headed stud).
September 17, 2009	The NRC reviewed the SQUG calculation for the raw water pump anchor bolts during an inspection, and determined it has several non-conservative errors. NRC issued a non-cited violation of 10 CFR 50, Appendix B, Criterion III for an inadequate seismic qualification of the raw water pump anchor bolts, to ensure that the pumps' anchor bolts embedded in the floor would meet Seismic Class I standards.
August – October, 2009	OPPD develops corrective actions to restore compliance for NRC identified violation. Licensee corrective actions verified the embedment of the raw water anchor bolts and correct the un-conservative errors in the calculation.
June 1, 2010	Intake structure barge impact analysis FC07083 completed. Analysis only uses design drawings for input, again assumed raw water anchor bolts were Type II (headed stud).
March 12, 2012	Raw water pump hold down torque analysis FC08069 completed. Analysis only uses design drawings for input, again assumed raw water anchor bolts were Type II (headed stud).
November 28, 2012	Raw water pump motor AC-10A is removed for service, and maintenance personnel identify that one of the anchor bolts cannot obtain the required torque value. The threads on one of the four anchor bolts is corroded, and it fails the visual inspection for minimum thickness. Station personnel evaluate plans to replace degraded raw water pump anchor bolt

DISCOVERY OF EVENT

November 28 – 30, 2012	Station personnel evaluate options for degraded anchor bolt on AC-10A raw water pump, and decide to replace the bolt.

December 2, 2012	System engineer prepares modification package to replace degraded anchor bolt, refers to drawings to determine if any interferences exist (such as rebar) with drilling the old anchor bolt out. The design drawing did not contain sufficient information about the anchor bolt details, so the engineer referred to the "Pour" drawing 7908-C and "Shop" drawing number 154.
December 2, 2012	System engineer prepares modification and included Type I, J-bolts in the package.
December 2, 2012	Design engineering personnel review the modification package, and did not expect the anchor bolts to be Type I J-bolts because all design information (calculations, analyses) assumed the bolt was a Type II headed stud anchor.

OPPD ACTIONS

January 9, 2013	Licensee declares all four raw water pumps inoperable
January 31, 2013	Licensee submits LER 2012-020
January – May 17, 2013	Licensee restores raw water pumps AC-10A, AC-10B, AC-10C, and AC-10D by replacing the Type I J-bolts with safety-related headed stud MaxiBolt anchor bolts.

3.2 Licensee Immediate Response to Event (Charter Item 2)

a. Inspection Scope

The team reviewed the licensee's immediate response to the event, including the corrective actions.

b. Findings and Observation

The team found that the licensee took adequate and timely corrective actions when the design issues related to the raw water pump anchorage were discovered. The NRC team concluded that the event notifications and declaration of all four raw water pumps as inoperable were appropriate. At the time of discovery, there was no immediate safety concern because the reactor was shutdown and all the fuel was located in the spent fuel pool.

3.3 Root Cause Analysis Report Review (Charter Item 3)

a. Inspection Scope

The inspection team evaluated the licensee's root cause analysis "Fort Calhoun Station Corrective Action Program Root Cause Analysis Report, Raw Water Pump Anchors, Condition Report 2012-19013," Revision 1.

Specifically, the team:

- Reviewed corrective actions,
- Reviewed licensee's extent of condition reviews associated with structural anchors and seismic restraints in other safety related system, structures, and components,
- Interviewed licensee personnel involved with the identification, removal, and replacement of the structural anchors and seismic restraints in the raw water systems,
- Interviewed personnel assigned to the licensee's root cause investigation team,
- Assessed whether the corrective actions were appropriate to correct the root and contributing causes,
- Inspected the remaining quarantined equipment and parts, and
- Performed inspections of selected risk-significant SSCs to verify that structural anchors installed in the plant matched the design and engineering drawings and calculations.

b. Observations (Extent of Condition)

In general, the NRC inspection team found various degrees of design errors with all selected components reviewed. These errors were typically minor drawing errors, meaning the drawings did not reflect the installed configuration of the plant. Additionally, the inspectors reviewed a number of calculations and found outdated or incorrect design information within those, as discussed below and in Section 3.6 (Findings). The following are observations related to the extent of condition for design control issues discovered by the NRC inspection team during the inspection period. These issues did not meet the criteria for regulatory findings.

.1 The inspection team reviewed design basis calculation for Raw Water (RW) Support RWS-46. These calculations are FC00043, "Redesign of Existing Seismic Support RWS-46," Revision 0, and FC 00206, "Design Calc Permanent Seismic Supports for Overstressed Restraints," Revision 0. The anchors that are part of the baseplate connection to the concrete structure were specified as Phillips Red Head type anchors. However, in response to NRC questions on the design basis adequacy of the anchors the licensee identified that the design drawings identified the anchors as Hilti Kwik Bolt. The licensee performed an engineering evaluation of RWS-46 and determined the anchorage met design basis requirements based on the current as-built configuration. The licensee initiated condition report CR 2013-03720 to address the design basis discrepancy and documentation error.

.2 The inspection team reviewed calculation FC1648, “Design of Pipe Support RWH-9,” Revision 2. This calculation detailed the 1 inch diameter lug that is part of raw water hanger (RWH-9) showed an overstress condition (applied stress is greater than the allowable stress). The inspectors asked whether or not this overstress condition was resolved. In response to the NRC question, the licensee provided calculation FC6390, “Resolution of Design Basis Open Item 113 for Subsystem RW-231A Pipe Support Analysis to Supplement SWEC Calculation 04170 NP-27,” Revision 0. This calculation determined the lug is not overstressed and met design basis requirements. Calculation FC6390 was not provided to inspectors in response to Request for Information (Attachment 3) for this Special Inspection as part of design basis for pipe support RWH-9. The only calculation provided was calculation FC1648 which appears to be superseded. Furthermore, it appears a large number of design issues were previously identified as part of the Design Basis Open Items project from the early 1990’s.

.3 The inspection team reviewed calculations FC1532, “Auxiliary Building, Design Calculations for Pipe Support RWS-46A,” Revision 0, FC1534, “Auxiliary Building, Design Calculations for Pipe Support RWS-48A,” Revision 0, FC2401, “Auxiliary Bldg., Design Calculations for Pipe Support RWH-1,” Revision 0, and FC2402, “Auxiliary Bldg., Design Calculations for Pipe Support RWH-2,” Revision 0. These calculations did not appear to contain sufficient documentation to establish design basis compliance for the raw water pipe supports. The licensee reviewed these calculations and agreed with the inspectors. The licensee initiated condition report CR 2013-03698, CR 2013-03718, and CR 2013-03727, to address the documentation issues. The licensee performed evaluations to validate that each pipe support will be able to perform their safety functions and was able to demonstrate design basis compliance for each pipe support.

.4 The inspection team requested the anchorage qualification for the containment air coolers VA-15 and VA-16 A/B. The licensee determined that no design calculation or record existed. The licensee performed an evaluation to validate that the anchorage will be able to perform their safety functions and was able to demonstrate design basis compliance for each anchorage. The licensee initiated condition report CR 2013-04158 to address the design basis documentation issue.

.5 The inspection team reviewed the U-bolt qualification for pipe support RWH-85 and RWS-89 contained in calculation FC1827, “Design of RWH-85,” Revision 0 and FC 1852, “Design of Pipe Support RWS-89,” Revision 0. The U-bolt evaluation did the address the U-bolt was subject to two-directional loading. The licensee provided an evaluation to demonstrate compliance in response to the NRC question. The licensee initiated corrective action to incorporate this new evaluation into the design basis calculation as documented in condition report CR 2013-03716.

.6 The inspection team reviewed calculation FC00924, “Calc.-Qualification Data Corresponding to Stress Analysis SI-192A,” Revision 2, to determine design basis adequacy of containment spray pipe supports SIH-24, SIS-84, and SIS-190. The support capacity appeared to be less than the applied design loads. The licensee reviewed and determined that the calculation of record did not reflect current design load information. The licensee found the correct design load information and the licensee demonstrated that the support capacity was greater than the applied design loads. The licensee will revise calculation FC00924 to incorporate the correct design information as described condition report CR 2013-02865.

.7 The inspection team reviewed calculation FC1634, "Calc-Input Data Corresponding to Stress Analysis AC-223A," Revision 2. The inspectors identified that Section 3.2.1 had an open item requiring confirmation for pipe supports ACS-156A, ACS-156, RWS-109, RWS-109A, and RWH-9. The licensee did further review and determined these open items had been closed in other design calculations, however, calculation FC1634 had not been updated to reflect this information. The licensee initiated condition report CR 2013-02929 to update the calculation with the current information.

.8 The inspection team reviewed calculation FC1534, "Auxiliary Building, Design Calculations for Pipe Support RWS-48A," Revision 0. The inspectors identified that in this calculation the pipe clamp associated with pipe support RHW-48A had a design load greater than the allowable capacity of the clamp. The inspector asked the licensee to provide the reference for the capacity of the pipe clamp since the clamp was not delineated in the calculation. The licensee identified in calculation FC00442, "Bergen-Patterson Pipe Clamp Shear Lug Mod Stress Calc.," Revision 1, that the clamp capacity was in fact greater than the design loads. The licensee initiated condition report CR 2013-03727 to update calculation FC1534 and resolve the error.

.9 The inspection team reviewed the GIP seismic calculations for the 480V and 4160V safety-related switchgear, specifically the screening evaluation worksheets (SEWS). The SEWS for busses 1B4A, 1B4C, 1B3A, 1A1, and 1A3 all concluded that the switchgear design was not seismically adequate because of an interference from overhead fluorescent lighting that was secured with an S-hook. However, the SEWS was never updated to reflect the as-built condition where the overhead lighting was secured and tied off to prevent any interaction with the electrical busses. This issue is considered minor, but is another example of design control documentation issues at the plant. The licensee initiated CR 2013-02220 to address this concern.

.10 The inspection team walked down the emergency feedwater storage tank located in Room 81 and compared the stirrup mounting supports to the design drawing. The inspectors determined that the drawing had a discrepancy associated with the bolting pattern for the stirrups. To address this concern, the licensee initiated CR 2013-02333.

3.4 Review of Raw Water Pump Repairs (Charter Item 4)

a. Inspection Scope

The inspection team observed all aspects of the repair for the raw water pump anchor bolts. The team reviewed work packages, engineering changes, and observed maintenance remove the J-style anchor bolts and replace them with DrillCo MaxiBolt anchor bolts (an undercut style anchor bolt).

b. Findings and Observations

The inspection team found that the repair (replacement) of the anchor bolts for the raw water pumps was adequate. The original J-style anchor bolts for the raw water pumps were cast-in-place in the concrete to an embedment depth of 7.5 inches. The replacement anchor bolts have an embedment depth of 16 inches and are considerably stronger than the old J-style bolts. However, the team identified issues with the calculation of record that certifies the use of the replacement anchor bolts.

Specifically, the team reviewed calculation FC07152, which qualifies the use of DrillCo MaxiBolt anchor bolts to a factor of safety of 2.5. The licensing basis requires a factor of safety of 4.0 for this type of undercut concrete anchor bolt. As a result, the engineering change package EC58524 was inadequate with respect to the bolts capacity when the higher safety factor is considered (4.0 versus 2.5). However, the licensee preformed a subsequent evaluation of the replacement DrillCo MaxiBolt anchor bolts with a safety factor of 4.0 and found they were acceptable. In response to this concern, the licensee generated CR 2013-5629. Additional regulatory findings associated with the raw water pump repairs are detailed in Section 3.6 of this report.

3.5 Independent Risk Assessment (Charter Item 5)

a. Inspection Scope

The team gathered information needed to assess the risk impact of the non-conforming pump anchor bolts. The team identified the total population of impacted equipment included all four safety-related raw water pumps. The team identified the length of time the equipment was susceptible to failure as a result of the insufficient anchor bolts.

b. Findings and Observations

The inspection team concluded that although the anchor bolts for the raw water pumps did not meet the design basis requirements, they had sufficient strength such that the safety function of the raw water pumps would not have been challenged during a seismic or other design basis event, and therefore the issue was of very low safety significance (Green). The violations of NRC requirements are listed in Section 3.6 of this report.

3.6 Findings

.1 Failure to translate raw water pump anchor bolt specifications into calculations and drawings

Introduction. The inspection team reviewed a self-revealing, Green, NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to translate the design basis into instructions, procedures, and drawings. The raw water pump anchor bolt design specifications and calculations incorrectly assumed they were a headed stud cast-in-place anchor bolt instead of the as-built J-style anchor bolts.

Description. During maintenance activities to replace a corroded anchor bolt on the raw water pump AC-10A, it was discovered that the anchor bolts were J-style anchor bolts with 7.5 inches of concrete embedment. The design specification and calculations for the raw water pump anchor bolts incorrectly assumed they were cast-in-place headed stud anchor bolts. Cast-in-place headed stud anchor bolts resist pullout more than J-style anchor bolts, therefore the seismic calculation for the raw water pumps was over-estimating the actual strength of the installed J-style anchor bolts.

Analysis. The failure to correctly translate the raw water pump anchor bolt design specifications into drawings and calculations is contrary to the requirements of Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" and is a performance deficiency. The finding was determined to be more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences.

Using Inspection Manual Chapter 0609, Attachment 4 "Initial Characterization of Findings," and Appendix A "The Significance Determination Process (SDP) for findings at-power," both dated 6/19/12, the inspectors determined performance deficiency affected the mitigating systems cornerstone and screened to Green because the finding affected the design and qualification of a mitigating SSC but remained operable. The inspectors used the at-power SDP because the condition existed since construction and while the plant was predominantly at power.

Because the finding did not ultimately affect the operability or functionality of the raw water pumps, the inspectors answered 'Yes' to Screening Question 1 of Exhibit 2 and determined the finding was of very low safety significance (Green).

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, drawings, procedures, and instructions and that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to this requirement, since construction the licensee's design control measures failed to ensure the design basis of the anchor bolts for the raw water pumps were correctly translated into specifications, drawings, and calculations.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 2013-02183. (NCV 05000285/2013012-01, "Failure to translate raw water pump anchor bolt specifications into calculations and drawings.")

.2 Inadequate functionality evaluation of the raw water pump anchor bolts

Introduction. The team identified a finding of very low safety significance (Green) for the licensee's failure to perform an adequate technical review of the past functionality of raw water pumps. Specifically, the licensee failed to meet the self-imposed standard required by the American Concrete Institute 349-01 code.

Description. The inspection team reviewed calculation FC 08216, "Raw Water Pump AC-10A/B/C/D Ultimate Failure Analysis," Revision 0. The objective of calculation FC 08216 was to determine the capacity (strength) of the Type I (J-style) anchor bolts for the raw water pumps.

The licensee used Appendix B of American Concrete Institute (ACI) 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures" to assess the past functionality of the concrete anchor bolts. NRC Regulatory Guide 1.199, "Anchoring Components and Structural Supports in Concrete," issued November 2003 generally endorsed Appendix B to ACI 349-01, except in the area of load combinations.

ACI 349-01 Appendix B, Section B.4.4 requires the use of strength reduction factors to establish the capacity of the anchorage. The strength reduction factors are based on either the failures mode of the steel anchorage or the concrete and are dependent on whether the applied loading is tensile or shear.

Contrary to the requirements of ACI 349-01, the inspectors identified that calculation FC 08216 did not apply the required strength reduction factors which resulted in a 25% overestimation of the anchorage capacity.

Analysis. The inspectors determined that the failure to consider strength reduction factors in the calculation was contrary to the requirements of ACI 349-01 and was a performance deficiency.

The finding was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors reviewed IMC 0609 Attachment 4, "Initial Characterization of Findings", Table 3 – SDP Appendix Router. In accordance with IMC 0609, "Significance Determination Process," Attachment 4, "Initial Characterization of Findings," Table 2, the inspectors determined the finding affected the Mitigating Systems cornerstone. As a result, the inspectors determined the finding could be evaluated using Appendix A, "The SDP for findings at-power,"

Exhibit 2, for the Mitigating Systems Cornerstone. The inspectors reviewed the revised evaluation and concluded that the licensee had corrected the deficiencies and provided reasonable assurance as to the past functionality of the raw water pump anchorage. Because the finding did not ultimately affect the operability or functionality of any equipment, the inspectors answered 'Yes' to Screening Question 1 and determined the finding was of very low safety significance (Green).

This finding had a cross-cutting aspect in the decision making component of the Human Performance cross-cutting area because the licensee used non-conservative assumptions in a functionality evaluation of raw water pump anchorage. Specifically, the licensee failed to use strength reduction factors in the evaluation of raw water pump anchorage [H.1(b)].

<u>Enforcement</u>: This finding did not involve enforcement action because no regulatory requirement was violated. This issue was entered into licensee's corrective action program as CR 2013-03600 (FIN 05000285/2013012-02, "Inadequate functionality evaluation of raw water pump anchor bolts").

.3 <u>Failure to correct thermal stress acceptance limits in raw water piping and piping support calculations</u>

<u>Introduction</u>. The team identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to correct a condition adverse to quality. Specifically, the licensee failed to correct design basis acceptance limits for the raw water RW-111A support and piping.

<u>Description</u>. The inspection team reviewed calculation FC02400, "Input Data Corresponding to Stress Analysis RW-111A and Qualification Summary", Revision 4, dated April 4, 1995. The purpose of FC02400 was to analyze pipe stresses of the raw water system. Raw water subsystem RW-111A runs from the component cooling water heat exchangers (AC-1A, 1B, 1C and 1D) in the auxiliary building to the discharge tunnel connection in the turbine building.

FC02400 concluded that sections of RW-111A would exceed the allowable stress specified in the licensing basis. Specifically, the calculation determined that the thermal stresses induced by a loss-of-coolant-accident would exceed the allowable stress for the raw water piping outlet to the component cooling water heat exchanger.

However, the licensee failed to correct this condition adverse to quality, and instead changed the allowable stress acceptance criteria for the piping system. Specifically, to address the applied stress on the piping and pipe supports in RW-111A that resulted from the thermal condition, the licensee used operability acceptance limits as specified by procedure MEI-17, "Application of Interim Operability Criteria," Revision 0. Procedure MEI-17 states, in part, "The interim operability criteria provides operability determination requirements for Critical Quality Element and Limited Critical Quality Element piping together with its associated supports if analysis determines that stresses or other criteria exceed allowables presented in the USAR and/or design code. These criteria permit operation for an interim period only. The interim period will be closed by modifications which return the system to USAR design basis conditions. Normally, modifications will be installed at the next refueling outage."

These interim operability acceptance criteria for RW-111A allowed the system to yield (permanently deform). The USAR Appendix F Table F-1 requires supports to be within yield (i.e., no deformation or linear behavior only) for their applied stresses. The piping is required to meet American Society of Mechanical Engineers (ASME) B31.7 code and USAR Appendix F requirements. The inspectors identified that the licensee failed to correct the overstress condition in accordance with ASME B31.7 code and USAR Appendix F but instead used operability acceptance limits without actions to restore conformance.

Analysis. The inspectors determined that failure to restore the Raw Water (RW) RW-111A piping and pipe supports to within license basis requirements was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the raw water piping system. Specifically, the use of interim acceptance criteria placed the RW-111A piping and pipe supports in a non-conforming/degraded condition for an extended period of time because corrective actions were not implemented or planned.

The inspectors reviewed IMC 0609 Attachment 4, "Initial Characterization of Findings", Table 3 – SDP Appendix Router. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "The Significance Determination Process (SDP) for Findings at-power," Appendix A, Exhibit 2, "Mitigating Systems Screening Questions." The inspectors answered "yes" to the question of "If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or functionality." The piping and pipe supports were evaluated and found to be operable.

Therefore, this finding screened as having very low safety significance (Green). The inspectors determined there was no cross-cutting aspect associated with this finding because the calculation was from the 1990s, therefore was not reflective of current performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected.

Contrary to the above, since April 4, 1995, the licensee failed to promptly correct a condition adverse to quality regarding the RW-111A piping and pipe supports. Specifically, the licensee failed to recognize that this condition did not meet the ASME B31.7 code and USAR Appendix F requirements. Because the licensee failed to recognize the requirements, the condition was evaluated incorrectly and determined to be acceptable such that corrective actions were not implemented. This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 2013-02276 (NCV 05000285/2013012-03, "Failure to correct thermal stress acceptance limits in raw water piping and pipe support calculations").

.4 Failure to adequately design containment air coolers structural bracing

Introduction. The inspection team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the adequacy of the design for the containment air coolers VA-16A and VA-16B. Specifically, the cooler design was not conservative and not in compliance with Class I requirements as defined in USAR Section 5.11 and referenced codes.

Description. Calculation FC03901, titled "Dynamic Seismic Analysis (VA-15A&B and VA-16A&B)", Revision 0 is the design basis seismic analysis for the containment air coolers. The inspectors reviewed the calculation and identified that the applied stress for the containment air coolers VA-16A & B structural column was 43,000 pounds per square inch – which is greater than the allowable yield stress of 36,000 pounds per square inch. The calculation determined that diagonal braces were necessary in order for the structural column to demonstrate compliance, however there were no records to indicate this bracing was installed. The NRC inspectors walked down the air coolers VA-16A & B to determine whether or not that diagonal bracing was installed and confirmed that it was not.

The containment air cooler system was designed to Class 1 as described in USAR Appendix F, Section 1.3, titled "Classification of Systems and Equipment." USAR Section 5.11.3, titled "Design Criteria - Class I Structures," states, in part, "The design of Class I structures, other than the containment, was governed by the then applicable building design codes and standards…Structural steel was designed in accordance with the requirements of the Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1963 edition, of the AISC. Elastic theory was the basis of design for all structural steel except for the hold-back bolts at the steam generators."

USAR, Appendix F, Section 2.1.2 states, in part, "The stress and deformation criteria for other Class I piping systems, vessels and supports for the various design load combinations are presented in Table F-1. The design load combination that includes the Maximum Hypothetical Earthquake has acceptance criteria for the support to be within yield stress (no permanent deformation/linear elastic). This acceptance criterion is consistent with elastic theory described in USAR Section 5.11.3."

Analysis. The inspectors determined that the failure to ensure the containment air cooler applied stress was less than yield stress was a performance deficiency. Specifically, compliance with Class 1 requirements was to ensure the containment air coolers can withstand all design basis events and maintain structural integrity with no permanent deformation. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the containment air cooler system.

The inspectors reviewed IMC 0609, Attachment 4, "Initial Characterization of Findings," Table 3 – SDP Appendix Router. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "The Significance Determination Process (SDP) for Findings at-Power," Appendix A, Exhibit 2, "Mitigating Systems Screening Questions." The inspectors answered "yes" to the question of "If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or functionality."

The containment air cooler structural column support was evaluated by the licensee and found to be operable, therefore, this finding screened as having very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to calculations from the 1970s and thus was not necessarily indicative of current licensee performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, the NRC identified that the licensee's design control measures failed to ensure adequacy of the containment air cooler design. Specifically, calculation FC03901 did not conform to the USAR requirements in that the stresses structural column in VA-16A & B exceeded the yield stress.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 2013-2260 (NCV 05000285/2013012-04, "Failure to adequately design containment air coolers structural bracing").

.5 Failure to adequately implement design requirements for U-bolt support

Introduction. A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspection team for the failure to ensure the adequacy of the U-bolt for Raw Water pipe support RWS-117. Specifically the U-bolt design did not demonstrate compliance to the vendor requirements.

Description. The U-bolt is a bent rod used to connect the piping to the pipe support steel. The U-bolt is evaluated based on the load capacities found in the vendor manual. The vendor manual specifies allowable stresses for the U-bolt that resist two-directional loads. The applied normal loading causes direct tension on the U-bolt while the applied side load causes shear and bending on the U-bolt. The two-directional U-bolt must satisfy the following interaction equation: the ratio of the applied vertical load over the allowable vertical load plus the ratio of the applied side load over the allowable side load must be less than or equal to 1.0. The inspectors reviewed calculation FC1859, “Raw Water Pipe Support Calculation for RWS-117,” Revision 1, which was the calculation to demonstrate that each structural element of pipe support RWS-117 met licensing basis requirements. The inspectors determined that the U-bolt interaction equation exceeded 1.0.

Analysis. The inspectors determined that failure to demonstrate compliance with vendor requirements for the U-bolt for RW Pipe Support RWS-117 was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the raw water piping system and pipe supports. Specifically, the calculation failed to consider the applied shear and tension loads in the interaction equation for the U-bolt.

The inspectors reviewed IMC 0609 Attachment 4, “Initial Characterization of Findings”, Table 3 – SDP Appendix Router. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “The Significance Determination Process (SDP) for Findings at-Power,” Appendix A, Exhibit 2, “Mitigating Systems Screening Questions.” The inspectors answered “yes” to the question of “If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or functionality.”

The U-bolt was evaluated by the licensee in accordance with their operability criteria as described in PED-MEI-17, “Application of Interim Operability Criteria,” Revision 2. The evaluation determined the U-bolts for RWS-117 were operable, therefore, this finding screened as having very low safety significance (Green).

The inspectors determined there was no cross-cutting aspect associated with this finding because the calculations were from the 1980s, therefore were not reflective of current performance

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program.

Contrary to the above the inspectors determined that for calculation FC01859, Revision 1, the licensee’s design control measures failed to ensure adequacy of the design. Specifically, this calculation did not conform to the U-bolt vendor requirements because the applied stresses exceeded the allowable stresses.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The violation was entered into the licensee’s corrective action program as CR 2013-03716 (NCV 05000285/2013012-05, “Failure to adequately implement design requirements for U-bolt support”).

.6 Failure to translate design requirements for embedded unistrut supports into calculations

Introduction. The inspection team identified several examples of very low safety significance (Green) non-cited violations of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to translate the design basis into instructions, procedures, and drawings for containment spray pipe supports SIH-17, SIH-94, and SIH-12. Specifically the initial design was non-conservative with respect to requirements defined by the unistrut concrete insert vendor manual and the calculations did not match the as-built condition.

Description. The unistrut concrete inserts are described in Section 4.2.5E of design basis document PLDBD-ME-10, "Pipe Stress and Supports", and are typically a type P3370 unistrut with a load carrying capacity of 1500 pounds.

The containment spray system was designed to Class 1 as described in USAR Appendix F, Section 1.3, titled "Classification of Systems and Equipment." Each pipe support is supported vertically by concrete inserts embedded into concrete structure ceiling. Multiple calculations for the pipe supports failed to incorporate a design change that was implemented to address overstressed unistruts. Specifically, the pipe supports and unistrut concrete inserts were modified by adding additional base plates, supported by anchor bolts, to meet design basis requirements. However, this modification resulted in a reduction in anchorage capacity due to the embedded unistrut being in close proximity to the anchorage. Additionally, the calculations were never updated to reflect the design change. The licensee has planned corrective actions to update calculations to reflect the current field configuration and incorporate the design change so they would meet design basis requirements. The affected calculations are listed below:

Calculation FC01777, "Design of Pipe Support SIH-94," Revision 0, states that the applied load from pipe support SIH-94 on unistrut is 1538 pounds which is greater than allowable load of 1500 pounds (based on a factor of safety of 3). The licensee also did not consider the self weight and self-weight seismic effects which will further degrade the margin.

Calculation FC10934, "Design of Support SIH-12," Revision 5, states that the applied load from pipe support SIH-12 on embedded unistrut is 1971 pounds which is greater than allowable load of 1500 pounds (based on a factor of safety of 3). The licensee also did not consider the self weight and self-weight seismic effects which will further degrade the margin.

Calculation FC02547, "Auxiliary Bldg., Design Calculations for Pipe Support SIS 70, SIS-100," Revision 0 states that the applied load from pipe support SIS-70 and SIS-100 on embedded unistrut is 792 pounds which is greater than allowable load of 750 pounds (based on a factor of safety of 3). The licensee also did not consider the self weight and self-weight seismic effects which will further degrade the margin.

Calculation FC03770, "Calc-Qualification Data Corresponding to Stress Analysis SI-187A", Revision 2 states that the applied load from pipe support SIH-17 on embedded unistrut is 1651 pounds which is greater than allowable load of 1500 pounds (based on a factor of safety of 3). The licensee also did not consider the self weight and self-weight seismic effects which will further degrade the margin.

Analysis. The inspectors determined that the failure to translate design requirements into instructions, procedures, and drawings for the embedded concrete inserts was a performance deficiency. Specifically, Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to ensure the design basis correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to the above, since construction the licensee failed to ensure the containment spray pipe support design basis was translated into plant calculations. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of the containment spray pipe supports.

Using Inspection Manual Chapter 0609, Attachment 4 "Initial Characterization of Findings", and Appendix A "The Significance Determination Process (SDP) for findings at-power," the inspectors determined performance deficiency affected the mitigating systems cornerstone and screened to Green because the finding affected the design and qualification of a mitigating SSC but remained operable.

The inspector did not identify a cross-cutting aspect associated with this finding because the concern was related to calculations from the 1970s and thus was not necessarily indicative of current licensee performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, the inspection team determined that since construction, the licensee failed to translate the design requirements of embedded unistruts into instructions, procedures, and drawings. Specifically, the pipe support calculations did not conform to the concrete insert design requirements and did incorporate an implemented design change to meet design basis requirements.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 2013-4940, 2013-4960, 2013-5222, and 2013-5243. (NCV 05000285/2013003-06, "Failure to translate design requirements for embedded unistrut supports into calculations.")

.7 Failure to translate electrical switchgear cabinet anchor bolt design specifications into drawings

Introduction. The inspection team identified several examples of a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to ensure the design basis for all the 480V and 4160V buss switchgear cabinets was correctly translated into specifications, drawings, procedures, and instructions. Specifically, each of the respective switchgear cabinet drawings show the equipment secured with concrete anchor bolts, however the inspectors identified the cabinets are secured with welds to an embedded steel plate.

Description. The switchgear cabinets contain safety-related electrical breakers and attendant equipment required for electrical distribution throughout the plant. Each electrical buss has its own cabinet, for a total of nine 480V switchgear cabinets (1B3A, 1B3A-4A, 1B3B, 1B3C, 1B3C-4C, 1B4A, 1B4B, 1B3B-4B, and 1B4C) and four 4160V switchgear cabinets (1A1, 1A2, 1A3, and 1A4). All thirteen switchgear cabinets depict 0.5 inch anchor bolts as the installed configuration of the plant. However, the switchgear cabinets are welded to a steel plate embedded in the concrete under the cabinets in the locations where the drawings show anchor bolts.

Analysis. The failure to translate the design specifications into drawings for all the switchgears cabinets is a performance deficiency. Specifically, Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that measures shall be established to ensure the design basis correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to the above, since construction the licensee failed to ensure the switchgear cabinet design basis was translated into plant drawings.

The performance deficiency is more than minor because if left uncorrected, it has the potential to lead to a more significant safety concern. Specifically, the licensee uses drawing information as the basis for activities affecting quality such as design changes, modifications, maintenance, and inspection activities. If the aforementioned activities are performed with incorrect information, as was the case with the raw water pump anchors described in section 3.6.1 of this report, it could lead to a more significant safety concern.

Using Inspection Manual Chapter 0609, Attachment 4 “Initial Characterization of Findings”, and Appendix A “The Significance Determination Process (SDP) for findings at-power”, both dated 6/19/12, the inspectors determined performance deficiency affected the mitigating systems cornerstone and screened to Green because the finding affected the design and qualification of a mitigating SSC but remained operable.

The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to calculations from the 1970s and thus was not necessarily indicative of current licensee performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, drawings, procedures, and instructions and that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to this requirement, since construction the licensee's design control measures failed to ensure the design basis of the anchorage for the electrical switchgear cabinets was correctly translated into drawings.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as CR 2013-02183 (NCV 05000285/2013012-07, "Failure to translate electrical switchgear cabinet anchor bolt design specifications into drawings").

.8 Failure to adequately design anchorage for containment spray and raw water system pipe supports

Introduction. The inspection team identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the adequacy of the anchorage for several raw water system and containment spray system pipe supports. Specifically the anchorage design was non-conservative with respect to the design basis requirements.

Description. The inspectors reviewed the following calculations:

- FC 00607, "Pipe Support SIH-14 Maximum Capacity & Qualification," Revision 2
- FC 01785, "Design of Pipe Support SIH-121," dated November 4, 1980
- FC 01786, "Design of Pipe Support SIH-241," dated October 13, 1980
- FC 01791, "Design of Pipe Support SIS-32A," Revision 2
- FC 01864, "Auxiliary Bldg., Design Calculations for Pipe Support ACH-53, ACH-52, ACS-6, ACS-13, SIH-113," dated February 10, 1981
- FC 01691, "Calculation for the Design of Pipe Support RWH-113," Revision 5
- FC 01902, "Auxiliary Bldg., Design Calculations for Pipe Support SIH-88," dated September 25, 1980
- FC 02409, "Turbine Bldg., Design Calculations For Pipe Support RWH-67," dated December 17, 1980
- FC 02412, "Turbine Bldg., Design Calculations For Pipe Support RWH-70," dated June 3, 1981
- FC 04228, "Design and Qualification of Pipe Supports for MR-FC-81-51," Revision 1
- FC 02433, "Intake Structure, Design Calculations for Pipe Support RWH-146," dated August 6, 1981
- FC 02436, "Design of Pipe Support RWH-16, RWS-43," dated March 4, 1981
- FC 02425, "Intake Structure, Design Calculations for Pipe Support RWH-129," dated July 1, 1981

The inspectors identified the above design calculation for Pipe Supports, ACS-412, ACS-434, ACS-421, ACS-448, ACS-228, ACS-449, ACS-429, RWS 135, RWS-137, RWS-138, SIH-14, SIH-121, SIH-241, SIS-32A, SIH-113, RWH-113, SIH-88, RWH-

67, RWH-70, RWH-146, RWH-16, RWH-43 and RWH-129 did not address one or several of the design requirements listed below for each anchorage analysis.

1) The anchorage was not evaluated for combined applied shear and tension interaction.
2) The anchorage interaction ratio was greater than 1.0 (applied stress greater than allowable stress)
3) The anchorage was not evaluated for the additional stresses due to self-weight and self-weight vertical and horizontal seismic excitation of pipe support and of pipe support structural elements.
4) The anchorage was not evaluated for the base plate flexibility in accordance with Bulletin 79-02 requirements.
5) The reduction in anchorage capacity was not evaluated for anchors that are in close proximity to an embedded unistrut or adjacent anchor bolts from another support attachment.

In response to this concern, the licensee initiated Condition Report (CR) 2013-05304, “Pipe Support Calculation Verification,” dated March 9, 2013 to address this concern.

The finding does not present an immediate safety concern because the plant is shutdown and the fuel in the reactor core has been offloaded.

Analysis. The inspectors determined that the failure to ensure adequacy of the anchorage of the aforementioned Containment Spray Pipe Supports and Raw Water Pipe Supports was not in accordance with design basis requirements and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the containment spray system and raw water system.

The safety significance is to be determined pending additional analysis of the as-found configuration of the anchorage and associated pipe supports by the licensee.

The inspectors determined there was no cross-cutting aspect associated with this finding because the calculation were from the 1980’s and therefore were not reflective of current performance

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above the inspectors identified that calculations FC00607, Revision 2, FC01785, dated November 4, 1980, FC01786, dated October 13, 1980, FC01791, Revision 2, FC01864, dated February 10, 1981, FC01691, Revision 5, FC01902, dated September 25, 1980, FC02409, dated December 17, 1980, FC02412, dated June 3, 1981, calculation FC04228, Revision

1, FC02433, dated August 6, 1981, FC02436, dated March 4, 1981 and FC02425, dated July 1, 1981, failed to ensure adequacy of the design. Specifically, these anchorage calculations did not conform to applicable design requirements.

The licensee's entered these issues into the corrective action program as CR 2013-05304. These issues are being characterized as an apparent violation in accordance with the NRC's Enforcement Policy, and its final significance will be dispositioned in a separate future correspondence (AV 05000285/2013012-08, "Failure to adequately design anchorage for containment spray and raw water system pipe supports").

.9 Failure to adequately implement design requirements for containment air cooler pipe supports

Introduction. The inspection team identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the adequacy of the U-bolts for Containment Air Cooler pipe supports VAS-1 and VAS-2. Specifically the U-bolt design was non-conservative with respect to the design basis requirements.

Description. VAS-1 and VAS-2 are pipe supports on the 4 inch condensation drain line for the containment air cooling units. The 4 inch condensation drain line begins as a drain for condensation from the cooling coils (VA-8A/B) inside of the air cooling unit housing (VA-16A/B) and extends down through the floor into the Elevation 1013 ft. level and eventually to the containment sump. These supports are safety-related and seismic class 1 because of the potential Seismic II/I interaction at Elevation 1013 ft. The condensation drain lines are positioned in close proximity to the safety-related Safety Injection (SI) tanks and over the top of SI valves. Calculation FC5918, "Containment Air Cooler Pipe Support VAS-1 and VAS-2," Revision 0, evaluated the VAS-1 and VAS-2 pipe supports. The calculation specified the U-bolts as 4 inch diameter Bergen Paterson Number 283 type U-bolts. Bergen-Paterson Pipe Support Catalog No. 66R provided a capacity for this U-bolt in only one direction (in tensile direction). However, the calculation required the U-bolt to withstand two-directional applied loading.

In response to this concern, the licensee initiated condition report CR 2013-03722. The finding does not present an immediate safety concern because the plant is shutdown and the fuel in the reactor core has been offloaded.

Analysis. The inspectors determined that the failure to ensure adequacy of the U-bolts for Containment Air Cooler pipe supports VAS-1 and VAS-2 in accordance with design basis requirements was a performance deficiency.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability, and capability of the safety injection tank and valves. Specifically, the one-directional

U-bolts for VAS-1 and VAS-2 are not designed to withstand two-directional loading and the condensate drain piping line has the potential to adversely impact the Safety injection tank and valves during a design basis event.

The safety significance is to be determined pending additional analysis of the as-found configuration of the condensate drain line and associated pipe supports by the licensee. The inspectors determined there was no cross-cutting aspect associated with this finding because the calculation was from the 1980s, therefore was not reflective of current performance

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program.

Contrary to the above, the inspectors identified that calculation FC05918, Revision 1, failed to ensure adequacy of the design. Specifically, this calculation did not conform to the U-bolts requirements by applying two-directional loading to a U-bolt restraint that is qualified for only one-directional loading. The licensee’s entered this issue into the corrective action program as CR 2013-03722. This issues is being characterized as an apparent violation in accordance with the NRC's Enforcement Policy, and its final significance will be dispositioned in a separate future correspondence. (AV 05000285/2013012-09, “Failure to adequately implement design requirements for containment air cooler pipe supports”).

.10 Unverified design for seismic damping values for raw water piping inside the turbine building

Introduction. The inspection team identified an unresolved item (URI) concerning the damping values used for the seismic analysis of the safety-related raw water outlet piping to the CCW heat exchangers.

Description. The inspectors reviewed calculation FC02400, “Input Data Corresponding to Stress Analysis RW-111A and Qualification Summary”, Revision 4. The raw water piping subsystem RW-111A runs from the CCW heat exchangers (AC-1A, 1B, 1C and 1D) in the auxiliary building to the discharge tunnel connection in the turbine building. On Page 2.8 of FC02400 it stated, in part, “Therefore pending NRC approval of Turbine Building spectra, PVRC spectra developed in References 5.4.3, Figures A-33 and A-34, for SSE Turbine Building conditions were compared to design basis OBE spectra.” The USAR Appendix F damping for piping is not described as PVRC damping and is 0.5% for the OBE and SSE condition.

In response to the concern, the licensee initiated corrective action program document CR 2013-07351.

Near the end of the inspection period, the licensee provided the inspectors additional information relevant to the licensing basis damping which will require additional review. Therefore, this issue is considered an unresolved item pending additional inspector review to determine whether design requirements are met (URI 05000285/2013012-10, “Unverified design for seismic damping values for raw water piping inside the turbine building”).

4OA3 Event Follow-up (71153)

<u>(Closed) LER 05000285/2012020-00, “Raw Water Pump Anchors.”</u>

On December 2, 2012, it was discovered that the anchor bolts for all the raw water pumps were not in accordance the design specifications. Specifically, the anchor bolts had inadequate strength and embedment depth. On January 31, 2013, the licensee submitted this LER which included information about the root cause of the event and planned corrective actions. The details and findings associated with this event are described in this inspection report. This LER is closed.

4OA6 Meetings

<u>Exit Meeting Summary</u>

The inspection team briefed members of Fort Calhoun Station staff on February 1, 2013, following completion of the first onsite portion of the inspection. An exit meeting was performed on April 11, 2013, with Mr. L. Cortopassi, Vice President and Chief Nuclear Officer, and other members of Fort Calhoun Station staff.

The inspectors verified whether the licensee considered any materials provided to or reviewed by the inspectors to be proprietary. None were identified.

ATTACHMENT 1: SUPPLEMENTAL INFORMATION
ATTACHMENT 2: SPECIAL INSPECTION CHARTER
ATTACHMENT 3: REQUEST FOR INFORMATION

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

L. Cortopassi	Vice President and Chief Nuclear Officer
M. Prospero	Plant Manager
B. Rash	Manager – Recovery
J. Carlson	Principal Engineer – Mechanical
J. Denton	Nuclear Engineer
M. Bare	Senior System Engineer
P. Koneck-Wilcox	Nuclear Engineer
J. Wiegand	Manager – Operations Support
J. Dolton	System Engineer – Raw water
T. Simpkin	Supervisor – Regulatory Compliance
E. Matzke	Compliance

NRC Personnel

M. Hay	Branch Chief
J. Kirkland	Senior Resident Inspector
J. Wingebach	Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000285/2013012-8	AV	Failure to adequately design anchorage for containment spray and raw water system pipe supports (Section 3.6.8)
05000285/2013012-9	AV	Failure to adequately implement design requirements for containment air cooler pipe supports (Section 3.6.9)
05000285/2013012-10	URI	Unverified design for seismic damping values for raw water piping inside the turbine building (Section 3.6.10)

Opened and Closed

05000285/2013012-1	NCV	Failure to translate raw water pump anchor bolt specifications into calculations and drawings (Section 3.6.1)
05000285/2013012-2	FIN	Inadequate functionality evaluation of the raw water pump anchor bolts (Section 3.6.2)
05000285/2013012-3	NCV	Failure to correct thermal stress acceptance limits in raw water piping and pipe support calculations (Section 3.6.3)
05000285/2013012-4	NCV	Failure to adequately design containment air coolers

		structural bracing (Section 3.6.4)
05000285/2013012-5	NCV	Failure to adequately implement design requirements for U-bolt support (Section 3.6.5)
05000285/2013012-6	NCV	Failure to translate design requirements for embedded unistrut supports into calculations (Section 3.6.6)
05000285/2013012-7	NCV	Failure to translate electrical switchgear cabinet anchor bolt design specifications into drawings (Section 3.6.7)

Closed

05000285/2012020-00	LER	Raw Water Pump Anchors (Section 4OA3)

DOCUMENTS REVIEWED

CALCULATIONS

Number	Title	Revision/ Date
FC08216	Raw Water Pump AC-10A/B/C/D Ultimate Failure Analysis	0
FC02400	Input Data Corresponding to Stress Analysis RW-111A and Qualification Summary	4
FC03770	Calc-Qualification Data Corresponding to Stress Analysis SI-187A	2
FC00206	Design Calc Permanent Seismic Supports for Overstressed Restraints	0
FC05918	Calculations for Supports VAS-1 and VAS-2 on Containment Coils VA-8A and VA-8B Drain Lines for Cooling Units VA-16A and VA-16B	0
FC03901	Dynamic Seismic Analysis VA-15A&B and VA-16A&B	0
FC10934	Design of Support SIH-12	5
FC01777	Design of Pipe Support SIH-94	0
FC01859	Raw Water Pipe Support Calculation for RWS-117	1
FC07152	DrillCo MaxiBolt Qualification	0
FC08069	Raw Water Pump Hold Down Bolt Torque Calculation	0
FC08072	Seismic Analysis Of Raw Water	1
FC00206	Design Calc Permanent Seismic Supports for Overstressed Restraints	0
FC02547	Auxiliary Bldg., Design Calculations for Pipe Support SIS 70, SIS-100	0

FC01691	Calculation for the Design of Pipe Support RWH-113	5
FC01957	Containment Design Calculations for Pipe Support VAS-45	0
FC01946	Calculations for the Design of Pipe Support ACS-172, VAS-1A	0
FC02465	Design of Pipe Support RWS-35A, RWH-30 and RWS-36	1
FC01534	Auxiliary Bldg., Design Calculations For Pipe Support RWS-48A	December 22, 1980
FC01961	Containment Design Calculations for Pipe Support VAS-44	0
FC01962	Containment Design Calculations for Pipe Support VAS-100	0
FC02063	Design of Pipe Support VAS-2A	0
FC02067	Design of Pipe Support VAS-33	1
FC02070	Design of Pipe Support VAS-36	1
FC02074	Design of Pipe Support VAS-40	1
FC02075	Design of Pipe Support VAS-41	0
FC02076	Design of Pipe Support VAS-101	2
FC00043	Redesign of Existing Seismic Support RWS-46	0
FC00607	Pipe Support SIH-14 Maximum Capacity & Qualification	2
FC01785	Design of Pipe Support SIH-121	November 4, 1980
FC01786	Design of Pipe Support SIH-241	October 13, 1980
FC01791	Design of Pipe Support SIS-32A	2
FC01864	Auxiliary Bldg., Design Calculations for Pipe Support ACH-53, ACH-52, ACS-6, ACS-13, SIH-113	February 10, 1981
FC01902	Auxiliary Bldg., Design Calculations for Pipe Support SIH-88	September 25, 1980
FC02409	Turbine Bldg., Design Calculations For Pipe Support RWH-67	December 17, 1980
FC02412	Turbine Bldg., Design Calculations For Pipe Support RWH-70	June 3, 1981

FC01827	Design of RWH-85	0
FC02436	Design of Pipe Support RWH-16, RWS-43	March 4, 1981
FC02433	Intake Structure, Design Calculations for Pipe Support RWH-146	August 6, 1981
FC02425	Intake Structure, Design Calculations for Pipe Support RWH-129	July 1, 1981
FC04228	Design and Qualification of Pipe Supports for MR-FC-81-51	1
FC00442	Bergen-Patterson Pipe Clamp Shear Lug Mod Stress Calc	1
FC00206	Design Calc Permanent Seismic Supports for Overstressed Restraints	0
FC01534	Auxiliary Building, Design Calculations for Pipe Support RWS-48A	0
FC01634	Calc-Input Data Corresponding to Stress Analysis AC-223A	2
FC01852	Design of Pipe Support RWS-89	0
FC00924	Calc.-Qualification Data Corresponding to Stress Analysis SI-192A	2
FC06390	Resolution of Design Basis Open Item 113 for Subsystem RW-231A Pipe Support Analysis to Supplement SWEC Calculation 04170 NP-27	0
FC01648	Design of Pipe Support RWH-9	2
FC02402	Auxiliary Bldg., Design Calculations for Pipe Support RWH-2	0
FC02401	Auxiliary Bldg., Design Calculations for Pipe Support RWH-1	0
FC01532	Auxiliary Building, Design Calculations for Pipe Support RWS-46A	0

CONDITION REPORTS (CR)

2013-02330*	2013-03383*	2013-03698*	2013-03727*	2013-05222*
2013-06216*	2013-04960*	2013-08037*	2013-07351*	2013-05103*
2012-18652	2012-18959	2012-19013	2013-02329*	2013-03384*

2013-02106*	2013-02183*	2013-02212*	2013-02220*	2013-02260*
2013-02276*	2013-02332*	2013-02333*	2013-02335*	2013-02596*
2013-02865*	2013-02929*	2013-03015*	2013-03589*	2013-03597*
2013-03598*	2013-03600*	2013-03601*	2013-03696*	2013-03716*
2013-03718*	2013-03720*	2013-03722*	2013-03725*	2013-03726*
2013-03728*	2013-03876*	2013-04158*	2013-04940*	2013-05243*
2013-05304*	2013-04326*	2012-20169	2012-19097	2009-3977
2013-1431	2013-2122			

*Issued as a result of NRC inspection activities.

DESIGN BASIS DOCUMENTS (DBD)

Number	Title	Revision
SDBD-AC-RW-101	Raw Water	39
SDBD-STRUC-503	Intake Structure	12
PLDBD-CS-51	Seismic Criteria	21
PLDBD-ME-10	Pipe Stress and Supports	15

DRAWINGS

Number	Title	Revision
11405-S-312	Intake structure and tunnels plan at Elev. 985’-0” & 993’-6” and details	13
7908-C	Intake structure	1
DWG-154	Venetian Ornamental Iron Works, Inc. Shop Drawing	3
2C-4825	Raw water pump AC-10A, B, C, D Technical spec H.16	5
A-6454	Raw water pump mounting plate AC-10A, B, C, & D	0
XC-545-S-107	Seismic restraints for raw water pumps	5
11405-E-75	Switchgear – Electrical penetration area section and details	71
15601	AKD-5 Powermaster Indoor Unit Substation No. 03	13

MISCELLANEOUS DOCUMENTS

Number	Title	Revision/Date
LIC-13-0009	Licensee Event Report 2012-020, Revision 0, for the Fort Calhoun Station	0
NED-12-0212	Memorandum: EC-58524 Rev 2- Replacement of All Foundation Bolts on AC-1O A/B/C/D	November 30, 2012
OPPD SEWS	Fort Calhoun Station Screening Evaluation Work Sheet (SEWS) for all switchgear	0
Attachment 9.6	Seismic Walkdown Checklist: Fukushima Near-Term Task Force Recommendation 2.3 Seismic Walkdowns	0

MODIFICATIONS (EC)

Number	Title	Revision
58524	Replacement of All Foundation Bolts on AC-10A/B/C/D	2

PROCEDURES

Number	Title	Revision
PED-CWP-5	Concrete Expansion Anchor Work Procedure	16a
PED-CSS-5	Standard Specification for Concrete Expansion Anchor Work	7
PED-CEI-5	Concrete Expansion Anchor Design	7
MEI-17	Application of Interim Operability Criteria	0
MM-RR-RW-0001	Removal and Installation of Raw Water Pumps	34
GIP	Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment	3A
MD-AD-0007	Administrative Procedure Bolting	8
NOD-QP-31	Operability Determination Process	53

VENDOR DOCUMENTS

Number	Title	Revision
GEK-7302	Drillco MaxiBolt Vendors Manual	

WORK ORDERS (WO)

466087-01	00461352	181503-19	357838	417316

ATTACHMENT 2

SPECIAL INSPECTION CHARTER

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

January 9, 2013

MEMORANDUM TO: Chris Smith, Project Engineer
Branch F
Division of Reactor Projects

THRU: Mike C. Hay, Chief */RA/*
Branch F
Division of Reactor Projects

FROM: Kriss M. Kennedy, Director
Division of Reactor Projects

SUBJECT: SPECIAL INSPECTION CHARTER TO EVALUATE THE INCORRECT INSTALLATION OF ANCHOR BOLTS AND SEISMIC RESTRAINTS ON THE RAW WATER PUMPS AT FORT CALHOUN STATION

In response to the condition of the raw water pumps being improperly anchored and restrained at the Fort Calhoun Station, a Special Inspection Team (SIT) is being chartered. You are hereby designated as the SIT leader.

A. Basis

On December 2, 2012, Fort Calhoun Station maintenance personnel were attempting to replace a damaged concrete anchor bolt for raw water pump motor AC-10A when they discovered the bolt was the incorrect size and type. Specifically, the raw water pump motor was secured to the floor with a nine inch embedment "J" bolt, instead of the required sixteen inch embedment straight bolt as specified in the design drawings and calculations. Each of the four raw water pump motors is secured to the floor with four anchor bolts, for a total of sixteen. The licensee performed an extent of condition review for the raw water pump motor concrete anchors and found that all sixteen anchor bolts were nine inch "J" bolts. Additionally, the installed seismic restraints on all four of the raw water pumps did not match design specifications.

The incorrect anchor bolts and seismic restraints have been installed in the plant since original construction, and these conditions degrade the ability of the raw water pumps to withstand a seismic event. The licensee has not analyzed the seismic capabilities of the as-found configuration. The preliminary data indicates that the as-found anchoring configuration would have been able to withstand 9 kips of tensile stress (the most pertinent seismic characteristic for this component). The required tensile strength for the design basis seismic event is 42 kips.

In accordance with Management Directive 8.3, "NRC Incident Investigation Program," deterministic and conditional risk criteria were used to evaluate the level of NRC response for this operational event. This event met the deterministic criterion for possible adverse generic implications. The initial risk assessment, while subject to uncertainty, indicates that the conditional core damage probability for the event is 1.9 x 10^{-5}, which is in the overlap range for a special inspection or augmented inspection. Region IV, in consultation with the Offices of Nuclear Reactor Regulation and Nuclear Security and Incident Response, concluded that the appropriate NRC response is to conduct a Special Inspection. Conduct of this inspection will be coordinated with Inspection Manual Chapter (IMC) 350 oversight and inspection activities.

This Special Inspection is being dispatched to obtain a better understanding of the event and to assess the licensee's actions to address this issue.

B. Scope

The SIT is to perform data gathering and fact finding in order to address the following:

1. Develop a complete sequence of events related to the event.

2. Assess the licensee's response to the event, including: implementation of immediate corrective actions, determination of the extent of condition, and initiation of event reports.

3. Assess the licensee's determination of the root and/or apparent causes of the event.

4. Assess the adequacy of the actions taken to restore the anchor bolts and seismic restraints to the designed specifications.

5. Collect data to support an independent risk assessment of the event. Specifically:

    - Collect information to determine the as-found seismic fragility of the raw water pumps and to estimate the likelihood that failures of one pump would lead to flood-related failures of the other pumps

6. Review any lessons learned from this special inspection and, when appropriate, prepare a feedback form on recommendations for improving the Reactor Oversight Process.

C. Guidance

The team will report to the site and conduct an entrance meeting. While on site, you will provide daily status briefings to Region IV management. You should notify Region IV management of any potential generic issues related to this event for discussion with the program office. Safety concerns that are not directly related to this event should be reported to the Region IV office for appropriate action. In addition, the team will brief the IMC 350 Panel on the results of the inspection and any associated findings. The inspection results will be documented in an inspection report as determined most appropriate by the IMC 350 Panel following the inspection.

The guidance in NRC Inspection Procedure 93812, "Special Inspection," and NRC Management Directive 8.3, apply to your inspection. This Charter may be modified should the team develop significant new information that warrants review. If you have any questions regarding this charter, contact Mike Hay at (817) 200-1147.

ATTACHMENT 3

REQUEST FOR INFORMATION

Information Request
January 15, 2013
Special Inspection – Raw water pump anchorage and restraints
January 28 – February 8, 2013
Fort Calhoun Station
Inspection Report 50-285/2013-012

This inspection will cover the period from January 28 to February 8, 2013. All requested information should be limited to this period or to the date of this request unless otherwise specified. To the extent possible, the requested information should be provided electronically in Adobe PDF (preferred) or Microsoft Office format. Lists of documents should be provided in Microsoft Excel or a similar sortable format.

Please provide the following no later than January 22, 2013:

1. Document Lists
   Note: For these summary lists, please include the document/reference number, the document title, initiation date, current status, and long-text description of the issue.

   a. Summary list of all historical corrective action documents related to anchorage or seismic restraints/supports.

   b. Summary list of all operating experience documents related to design or configuration control for anchorage or seismic restraints/supports.

2. Full Documents with Attachments

   a. Root or Apparent Cause Evaluations related to anchorage or seismic restraints/supports.

   b. Calculations used to determine seismic loading for the raw water pumps, containment air coolers, containment spray system, and the safety-related 480V switchgear cabinets

   c. Calculations to determine the pipe stresses within the raw water system, containment coolers, and the containment spray system

   d. Design calculations used to determine the sizing requirements for anchorage or seismic supports/restraints for the raw water system, containment air coolers, containment spray system, and the safety-related 480V switchgear cabinets (if different than items 2 a., 2 b., and 2 c.)

3. Drawings and descriptions

   a. Design, construction, and in-plant (field) drawings for the raw water system, containment air coolers, containment spray system, and the safety-related 480V switchgear cabinets. These drawings should include details sufficient to determine the anchorage or seismic restraints/supports installed for each system.

b. Current system health reports or similar information for the raw water system, containment air coolers, containment spray system, and the safety-related 480V switchgear

4. Procedures

a. Concrete anchorage program procedures

b. Seismic restraint or supports program procedures

c. Any walkdown procedures for verifying as-built versus design drawings

Note: “Corrective action documents” refers to condition reports, notifications, action requests, cause evaluations, and/or other similar documents, as applicable to Fort Calhoun Station.

All requested documents should be provided electronically. Regardless of whether they are uploaded to an internet-based file library (e.g., Certrec’s IMS), please provide copies on CD or DVD. Two copies of the CD or DVD should be sent to the team lead, to arrive no later than February 8, 2013:

Christopher Smith
U.S. NRC Region IV
1600 East Lamar Blvd.
Arlington, TX 76011-4511