Omaha Public Power District
*444 South 16th Street Mall*
*Omaha, NE 68102-2247*

LIC-13-0003
January 18, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Reference: Docket No. 50-285

**Subject: Licensee Event Report 2012-014, Revision 1, for the Fort Calhoun Station**

Please find attached Licensee Event Report 2012-014, Revision 1, dated January 18, 2013. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(i)(B).

No commitments are being made in this letter.

If you should have any questions, please contact me.

Sincerely,

Louis P. Cortopassi
Site Vice President and CNO

LPC/rjr

Attachment
c: E. E. Collins, Jr., NRC Regional Administrator, Region IV
L. E. Wilkins, NRC Project Manager
J. C. Kirkland, NRC Senior Resident Inspector

| NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013 |
|---|---|
| **LICENSEE EVENT REPORT (LER)**<br>(See reverse for required number of digits/characters for each block) | Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection. |

| 1. FACILITY NAME | 2. DOCKET NUMBER | 3. PAGE |
|---|---|---|
| Fort Calhoun Station | 05000285 | 1 OF 4 |

**4. TITLE**
Containment Beam 22 Loading Conditions Outside of the Allowable Limits

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO. | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER 05000 |
| 07 | 11 | 2012 | 2012 | - 014 - | 1 | 01 | 18 | 2013 | FACILITY NAME | DOCKET NUMBER 05000 |

| 9. OPERATING MODE | 11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)* | | | |
|---|---|---|---|---|
| 5 | ☐ 20.2201(b) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(i)(C) | ☐ 50.73(a)(2)(vii) |
| | ☐ 20.2201(d) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(ii)(A) | ☐ 50.73(a)(2)(viii)(A) |
| | ☐ 20.2203(a)(1) | ☐ 20.2203(a)(4) | ☐ 50.73(a)(2)(ii)(B) | ☐ 50.73(a)(2)(viii)(B) |
| | ☐ 20.2203(a)(2)(i) | ☐ 50.36(c)(1)(i)(A) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(ix)(A) |
| 10. POWER LEVEL | ☐ 20.2203(a)(2)(ii) | ☐ 50.36(c)(1)(ii)(A) | ☐ 50.73(a)(2)(iv)(A) | ☐ 50.73(a)(2)(x) |
| 0 | ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(v)(A) | ☐ 73.71(a)(4) |
| | ☐ 20.2203(a)(2)(iv) | ☐ 50.46(a)(3)(ii) | ☐ 50.73(a)(2)(v)(B) | ☐ 73.71(a)(5) |
| | ☐ 20.2203(a)(2)(v) | ☐ 50.73(a)(2)(i)(A) | ☐ 50.73(a)(2)(v)(C) | ☐ OTHER |
| | ☐ 20.2203(a)(2)(vi) | ☒ 50.73(a)(2)(i)(B) | ☐ 50.73(a)(2)(v)(D) | Specify in Abstract below or in NRC Form 366A |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER *(Include Area Code)* |
|---|---|
| Erick Matzke | 402-533-6855 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| ☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* | ☒ NO | | | | |

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*


On January 24, 2012, during the modification preparation for pipe supports for component cooling water piping in containment, multiple discrepancies were identified between the design calculations and the design drawings for concrete beams in the steam generator bays, 1060 foot platform elevation, and the floor slab at the 1045 foot elevation of containment. On July 11, 2012, while performing the Extent of Condition, it was determined that the loading conditions for Beam B-22, a structural member of the containment internal structure at the 1013 foot elevation, were outside the allowable limits for both Working Stress and No Loss of Function load combinations as noted in the USAR Section 5.11. Additional analysis has been completed which shows that with a live load of 140 psf or less, Beam 22 is able to meet its design function.

The Root Cause Analysis completed December 21, 2012, and determined the condition described in this report was due to inadequate ownership review by Omaha Public Power District of plant construction architect/engineer produced calculations.

**NARRATIVE**

BACKGROUND

Fort Calhoun Station (FCS) is a two-loop reactor coolant system of Combustion Engineering (CE) design.

Limiting Condition for Operation (LCO)

2.6 Containment System

This LCO does not contain a specification stating the requirements related to the containment internal support structures. However, systems such as the containment and safety injection rely on the containment support structures integrity to be maintained under accident conditions for each individual system to be OPEARABLE.

Operable – Operability is defined in the Technical Specifications (TS) as:

> A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power sources, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety functions(s) are also capable of performing their related support function(s).

The FCS Updated Safety Analysis Report (USAR), Section 5.11.3 Design Criteria - Class I Structures, Sub-section, a. Loadings states in part:

> Class I structures were designed on the basis of working stress for the following load combinations:
> S = D + L
> S = D + L + W or E
> S = D + F
> where:
> S = Required section capacity
> D = Dead load
> L = Live load, including hydrostatic load
> W = Wind load
> E = Design earthquake
> F = Hydrostatic load to elevation 1007 feet
>
> The ACI Code 318-63 and the AISC Code for Structural Steel, 1963 edition, design methods and allowable stresses were used for reinforced concrete and steel structures, respectively.
>
> The concrete structure within the containment was considered as a Class I structure and was subject to the loads and analysis noted above with the exception of wind and tornado loads. In addition, a transient analysis was made to determine the maximum differential pressure across the interior shielding and structural walls and floors. Openings in the interior concrete walls and floors are provided and grating floors are used wherever possible, without reducing the necessary shielding, to allow pressurization of all compartments with the minimal differential pressure across walls and floors.

NARRATIVE

## EVENT DESCRIPTION

On January 24, 2012, during the modification preparation for pipe supports for component cooling water piping in containment, multiple discrepancies were identified between the design calculations and the design drawings for concrete beams in the steam generator bays, 1060 foot platform elevation, and the floor slab at the 1045 foot elevation of containment. Due to the problems identified with the design of various concrete internal structures, a Condition report (CR) was written on May 22, 2012, to track the extent of condition, determine the causes, and correct the problems.

On July 11, 2012, while performing the Extent of Condition, it was determined that the loading conditions for Beam B-22, a structural member of the containment internal structure at the 1013 foot elevation, were outside the allowable limits for both Working Stress and No Loss of Function load combinations as noted in the USAR Section 5.11. An updated analysis, as discussed below, later showed this not to be the case.

This condition was identified on July 11, 2012, while the unit was shut down and reported to the U.S. Nuclear Regulatory Commission (NRC) Operations Center the same day at approximately 1603 CDT under Event Notification Number 48094.

This condition is being submitted pursuant to:

- 10 CFR 50.73(a)(2)(ii)(B), Any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

## CONCLUSION

The Root Cause Analysis (RCA) completed December 21, 2012, and determined the condition described in this report was due to inadequate ownership review by Omaha Public Power District of plant construction architect/engineer produced calculations.

## CORRECTIVE ACTIONS

Beam 22 was identified as having potential loading conditions outside the allowable limits for the load combination for shutdown conditions. Therefore, additional analysis were performed that concluded that Beam 22 is functional when using provisions in EPRI NP-6041-SL, A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Revision 1, the increased ductility ratio in ACI-349-01, Building Code Requirements for Reinforced Concrete, an increased concrete compressive strength obtained from concrete break strengths on record, and reducing the live load seen by the beam from the current value of 200 psf to 140 psf. A walkdown of the area by Design Engineering estimated that the floor live load was approximately 100 psf when the walkdown was performed. To preserve this margin, compensatory actions were established that removed any equipment that was contributing to current live loading of the support beam and to isolate and post the affected area to ensure no equipment is stored in the area without engineering analysis.

The RCA team reviewed the current organizational and process procedures (SO-G-21, Modification Control, PED-QP-3, Calculation Preparation, Review and Approval, PED-QP-5, Engineering Analysis Preparation, Review and Approval, PED-GEI-3, Preparation of Modifications, and PED-GEI-28, Preparation of Construction Work Orders) and concluded that they are vastly improved from 1968 and no revisions are necessary to address the root cause.

In a related RCA, it was identified that FCS had subsequent unsuccessful opportunities to identify and correct this issue. Although this did not cause the problem, the problem could have been corrected

NARRATIVE

previously, and thus is a contributing cause. This condition is being addressed through RCA 2012-08125, Engineering Design/Configuration Control, AI-008, CAPR-2, "Implement a new Engineering organizational structure based on development of the following procedures consistent with best industry practices:

- Conduct of Engineering – Principles and Expectations,
- Engineering Division of Responsibilities, and
- Departmental Duties and Responsibilities".

SAFETY SIGNIFICANCE

Fort Calhoun Station was in Mode 5 at the time this condition was discovered.

Beam B-22, which supports safety injection tanks SI-6B and SI-6D, was determined to be susceptible to an overstressed condition during past operating cycles, for a hypothetical major break in the Reactor Coolant System, under the controlling "No Loss of Function" design load combination. Although this is what the condition of Beam 22 was thought to be at the time of discovery, the additional analysis described above shows that with a live load of 140 psf or less, Beam 22 was able to meet its design function. Therefore, there was no loss of function and the previously identified reporting criteria of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(D) no longer apply nor does this condition represent a safety system functional failure, as previously reported.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event does not result in a safety system functional failure in accordance with NEI-99-02.

PREVIOUS EVENTS

A cause analysis determined that there have been no previous events with the same underlying concern or reason of this event, such as the same root cause, failure, or sequence of events.