UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 15, 2015

Mr. Louis P. Cortopassi
Site Vice President and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Blair, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 – SUPPLEMENTAL INFORMATION NEEDED FOR ACCEPTANCE OF REQUESTED LICENSING ACTION RE: REVISE CURRENT LICENSING BASIS TO USE AMERICAN CONCRETE INSTITUTE ULTIMATE STRENGTH REQUIREMENTS (CAC NO. MF6676)

Dear Mr. Cortopassi:

By letter dated August 31, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15243A167), the Omaha Public Power District (OPPD) submitted a license amendment request for Fort Calhoun Station, Unit No. 1 (FCS). The proposed amendment would revise the FCS Updated Safety Analysis Report (USAR) to change the structural design methodology for Class I structures at FCS with several exceptions. The exceptions are the containment shell, the spent fuel pool, and the foundation mat under the containment and auxiliary buildings, as well as the foundation mat under the intake structure.

The purpose of this letter is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this amendment request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

Consistent with Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), an amendment to the license (including the technical specifications) must fully describe the changes requested, and following as far as applicable, the form prescribed for original applications. Section 50.34 of 10 CFR addresses the content of technical information required. This section stipulates that the submittal address the design and operating characteristics, unusual or novel design features, and principal safety considerations.

The NRC staff has reviewed your application and concluded that the information delineated in the enclosure to this letter is necessary to enable the staff to make an independent assessment regarding the acceptability of the proposed amendment request in terms of regulatory requirements and the protection of public health and safety and the environment.

In order to make the application complete, the NRC staff requests that OPPD supplement the application to address the information requested in the enclosure by December 29, 2015. This will enable the NRC staff to begin its detailed technical review. If the information responsive to the NRC staff's request is not received by the above date, the application will not be accepted for review pursuant to 10 CFR 2.101, and the NRC will cease its review activities associated with the application. If the application is subsequently accepted for review, you will be advised of any further information needed to support the staff's detailed technical review by separate correspondence.

The information requested and associated time frame in this letter were discussed with Mr. B. Hansher, et al., of your staff on December 10, 2015.

If you have any questions, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure:
Supplemental Information Request

cc w/encl: Distribution via Listserv

SUPPLEMENTAL INFORMATION NEEDED

AMENDMENT REQUEST DATED AUGUST 31, 2015

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

By letter dated August 31, 2015 (Agencywide Documents Access and Management System Accession No. ML15243A167), the Omaha Public Power District (OPPD) submitted a license amendment request (LAR) for Fort Calhoun Station, Unit No. 1 (FCS). The proposed amendment would revise the FCS Updated Safety Analysis Report (USAR) to change the structural design methodology for Class I structures at FCS with several exceptions. The exceptions are the containment shell, the spent fuel pool, and the foundation mat under the containment and auxiliary buildings, as well as the foundation mat under the intake structure. Specifically, this LAR proposes the following changes:

1. Replace the working stress design (WSD) method with the ultimate strength design (USD) method from the American Concrete Institute (ACI) 318-63 Code for normal operating/service conditions associated with Class I concrete structures other than the containment shell, the spent fuel pool, and the foundation mats.

2. Use higher concrete compressive strength (f'c) based on cylinder break test data at select locations in the Class I structures.

3. Use higher reinforcing steel yield strength (fy) values for the containment internal structure (CIS) that includes the reactor cavity and compartments (RC&C) and the CIS beams, slabs and columns.

4. Add new methods for evaluating the RC&C walls. These new methods include the limit design method and the use of dynamic increase factors (DIF) for concrete analysis.

5. Adding a definition of control fluids to the dead load section. One-third increase in allowable stress values for structural steel is being added for operating basis earthquake (OBE) load combinations.

Review/Evaluation for Acceptance

Item 1: This change appears to be a first-of-a-kind request for operating plants (no other LAR was identified to request similar change). The change from WSD to USD is reasonable. However, please supplement the technical content of the LAR prior to acceptance of the LAR for the following items:

(a) The licensee refers to "containment shell." The proposed LAR should be clear that the proposed changes are not applicable to the containment structure, as a whole (i.e., containment shell may be interpreted as the containment cylindrical wall).

(b) The proposed load combinations should clearly state that soil dynamic pressure and hydrodynamic loading shall be accounted for, where applicable.

Item 2 and Item 3:

I. Using higher concrete compressive strength (f'c) based on historical test data was used for operability evaluation of the CIS beams/slabs/columns prior to FCS restart in December 2013. In this LAR, the licensee proposes to use the higher f'c for the Auxiliary Building (above Elevation 1007) and the CIS. The containment structure, intake structure, spent fuel pool, reactor cavity floor, and concrete around the reactor vessel will continue to use the f'c currently specified in the design basis documents.

II. Using higher reinforcing steel yield strength (fy) based on certified material test reports (CMTRs) was used for the operability evaluation of the CIS prior to FCS restart. In this LAR, the use of 44 ksi instead of the 40 ksi is proposed for the RC&C and the CIS beams, slabs, and columns.

The f'c, determined based on statistical evaluation of cylinder breaks test data, and the fy determined based on CMTRs, has historically been used for operability assessments and seismic margin assessments. The following items were identified requiring supplemental information. Please supplement the LAR and provide further information prior to acceptance of this portion of the LAR for review.

(a) This request should be accompanied by inspection results demonstrating no degradation or structural distress in the structures where the use of higher f'c and fy is requested. The LAR does not include any information regarding the structural health of these structures.

(b) There should be a more frequent structural inspection interval because the margin associated with the minimum specified f'c and fy in the ACI Code is being negated. This should be a license condition for this LAR.

(c) The LAR states that the higher f'c will be established based on concrete compressive strength test results of all samples environmentally controlled in the lab. Section 2.10 of the LAR states that numerous concrete test (i.e., core pour) records are available that accurately show test results and reference the specific location in the Class I structures where the concrete was placed. However, there is no indication in the LAR that the licensee intends to supplement the lab test data with the in-situ tests or nondestructive examination (NDE).

Item 4: Section 3.4 of the LAR states the following:

I. All applicable load combinations involve the compartmental pressure loads for the high-energy line break loading case. Ductility limits are as specified in Sections C3.3, C3.4, C3.5 and C3.7 of ACI 349-97, Appendix C.

II. OPPD requests that DIFs may be utilized as specified in ACI 349 C.2.1.

There are several technical deficiencies related to this request:

(a) This LAR requests to use ACI 349-97, Appendix C provisions for ductility ratios without recognizing that NRC Regulatory Guide (RG) 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," takes exceptions to the ductility ratios, as noted in Regulatory Position 10 (the USAR mark-up included in the LAR is not consistent with RG 1.142).

(b) Section 2.9.2 of the LAR states that OPPD takes exception to the ACI 349-97 reinforcing steel detailing due to the time of construction. Adopting ACI 349-97, Appendix C ductility ratios without addressing the inelastic deformation/rotation capability of FCS Class I structures is a technical deficiency in this LAR. Also, partial adoption of a more up-to-date code (ACI 349) is not consistent with industry practice.

(c) The USAR mark-up included in this LAR proposes to use, for the reactor cavity and compartment walls and transfer canal, a ductility demand of $0.05/(\rho - \rho')$ not to exceed 10 in regions where moment governs and compartment pressurization is not present, and up to a ductility demand of 3 in regions with compartment pressurization loading where moment governs the design. The statement "... ductility demand of up to 10 when compartment pressurization is not present ..." implies that this ductility ratio is proposed to be used for other loading conditions, including seismic loading. This is not consistent with the intent of ACI 349, Appendix C. In addition, the ACI 349 Code requires elastic design for seismic loading condition. Therefore, this request is not technically justified and not consistent with NRC staff's guidance (NUREG-0800, Standard Review Plan; SRP) and industry codes/standards.

(d) The USAR mark-up also indicates that the licensee intends to use an inelastic energy absorption factor of 1.25 for the CIS beams and columns. The technical content of the LAR did not discuss the use of this factor. The use of inelastic energy absorption factor has been historically used in seismic margin assessment and operability evaluations. The ACI 349 Code, referenced in this LAR, requires elastic design and does not allow the use of inelastic energy absorption factor. Therefore, this request is not technically justified and not consistent with NRC staff's guidance (SRP) and industry codes/standards.

In summary, the use of DIF, within the intent of ACI 349-97, Appendix C (i.e., expected strain rate for impulse/impact loading) is reasonable. The request to use ductility ratios of ACI 349, Appendix C, as described in this LAR, is not adequately justified.

Item 5:

Section 2.5 of the LAR states that clarity was added to address seismic load conditions. For operating basis, the allowable stresses can be increased by one-third when the load combinations include wind or seismic load(s). This is defined in American Institute of Steel Construction (AISC)-63, Section 1.5.6.

The SRP as well as the AISC N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities," have historically specified use of normal allowable stresses without the one-third increase for the OBE load combinations. Therefore, the NRC staff does not consider this change as a clarification that is added to the USAR. The LAR does not specifically provide a roadmap to establish the FCS Class I steel structures licensing basis (original safety evaluation report, original design calculations, etc.) regarding the one-third increase in stress allowable for OBE load combinations.

for review pursuant to 10 CFR 2.101, and the NRC will cease its review activities associated with the application. If the application is subsequently accepted for review, you will be advised of any further information needed to support the staff's detailed technical review by separate correspondence.

The information requested and associated time frame in this letter were discussed with Mr. B. Hansher, et al., of your staff on December 10, 2015.

If you have any questions, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure:
Supplemental Information Request

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NAME	RPascarelli	FLyon
DATE	12/14/15	12/15/15

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