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LIC-14-0107
August 22, 2014

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

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Subject: OPPD Response to NRC Request for Additional Information Regarding Fort Calhoun Station, Unit No. 1 Reactor Vessel Internal Component Aging Management Program

References:
1. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), "Submittal of Reactor Vessel Internal (RVI) Component Aging Management Program (AMP) for Fort Calhoun Station (FCS), Unit No.1," dated September 27, 2012 (ML12276A005) (LIC-12-0144)
2. E-mail from NRC (J. Rankin) to OPPD (M. Edwards / B. Hansher), "Request for Additional Information – Reactor Vessel Internal Component Aging Management Program (MF3412)," dated July 8, 2014 (ML14190A211)

This letter responds to a U. S. Nuclear Regulatory Commission (NRC) request for additional information (Reference 2) regarding the Fort Calhoun Station, Unit No. 1 Reactor Vessel Internal (RVI) Aging Management Program (Reference 1). Attached is the Omaha Public Power District (OPPD) response to RAIs 4, 6, 7, 8, and 9. As indicated in the NRC e-mail (Reference 2), OPPD agreed to provide this response within 45 days with the remaining questions to be addressed in a future submittal.

This letter contains no regulatory commitments. If you should have any questions regarding this submittal or require additional information, please contact Mr. Bill R. Hansher, Supervisor-Nuclear Licensing, at 402-533-6894.

Respectfully,

Louis P. Cortopassi
Site Vice President and CNO

LPC/KGM/mle

Attachment: OPPD Response to NRC Request for Additional Information Regarding Fort Calhoun Station, Unit No. 1 Reactor Vessel Internal Component Aging Management Program

OPPD Response to NRC Request for Additional Information Regarding Fort Calhoun Station, Unit No. 1 Reactor Vessel Internal Component Aging Management Program

REQUEST FOR ADDITIONAL INFORMATION
AGING MANAGEMENT PROGRAM FOR THE
REACTOR VESSEL INTERNALS
FORT CALHOUN STATION, UNIT 1
OMAHA PUBLIC POWER DISTRICT
DOCKET NO. 50-285

RAI 4:

Chapter 7 of the MRP-227-A report addresses how a licensee will evaluate and disposition relevant plant-specific or generic operating experience (OE) that is applicable to RVI components at its PWR facility. The NRC staff requests that the licensee identify any and all generic and plant-specific OE that is applicable to the design of the RVI components at Fort Calhoun, including but not limited to OE that is applicable the following components at FCS:

> Panel to former bolts, core barrel bolting, thermal shields (including positioning pins), fuel alignment pins, guide lug inserts and bolts, guide lugs, core support barrel girth welds, in-core instrumentation flux thimble tubes, core barrel at the prior thermal shield bracket attachment areas, RV flow skirt, and, include the RVI components addressed in Appendix B of the September 27, 2012 submittal.

OPPD Response

Fort Calhoun Station (FCS) maintains and reviews both plant-specific and generic OE that is applicable to Reactor Vessel Internal (RVI) components. OE is one source of information used to develop the FCS aging management programs.

RVI OE review consists of components which include: Panel to former bolts, core barrel bolting, thermal shields (including positioning pins), fuel alignment pins, guide lug inserts and bolts, guide lugs, core support barrel girth welds, in-core instrumentation flux thimble tubes, core barrel at the prior thermal shield bracket attachment areas, RV flow skirt. Also included in the RVI components addressed in Appendix B of the September 27, 2012 submittal, are the Control Element Assembly (CEA) Shroud Assembly, Core Support Barrel Assembly, the Incore Instrument Assembly Guide Tubes, the Lower Support Structure, and the Upper Guide Structure.

The age-related degradation mechanism leading to the aging affect are Inter-Granular Stress Corrosion Cracking (IGSCC), Primary Water Stress Corrosion Cracking (PWSCC), Irradiation Assisted Stress Corrosion Cracking (IASCC), Flow-induced Vibration, Loss of Material, Irradiation-Induced Growth and other miscellaneous mechanisms.

A detailed review of the FCS Corrective Action Program (CAP) for the RVI components listed above, which failed due to one or more of the above degradation mechanisms was performed. The FCS CAP program includes generic OE from the nuclear industry. Previous failures of the FCS RVI components listed above have been corrected. No FCS RVI components are currently failed or are being evaluated for previous failures. OE review is a responsibility of the RVI program owner and is documented in the program owner's health reports each trimester.

In addition to the review of previous failures of RVI components at FCS and OE from the nuclear industry conducted for this response, future outage work, which includes the 10-year in-

service inspection (ISI) of RVI and MRP-227-A RVI inspection results will be evaluated by both OPPD and the vendor performing the inspections according to the requirements documented in MRP-227-A.

RAI 6:

The Fort Calhoun Updated Safety Analysis Report (USAR) includes Chapter 15, which summarizes the applicant's aging management programs, time-limited aging analyses, and license renewal commitments that will be implemented to manage or analyze the effects of aging during the period of extended operation. This USAR section includes Commitment Nos. 16, 17, and 18 relative to recommended inspections, evaluations, or analyses for RVI components at Fort Calhoun. However, these commitments were established well before the industry's development of the augmented inspection and evaluation guidelines for CE-designed plants in MRP-227-A. Thus, the staff seeks information on how implementation of the new MRP-based program relates to these commitments. Specifically, clarify and justify how implementation of the MRP-227-A based program for Fort Calhoun supersedes, augments, adjusts, replaces or fulfills the criteria for RVI components in Commitment Nos. 16, 17, and 18 of USAR Chapter 15.

OPPD Response

The following table from USAR Section 15.4, "License Renewal Commitment Listing," lists the three commitments (16, 17, and 18):

ITEM NUMBER	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
Reactor Vessel Internals Inspection Program			
16	Visual inspections of the core shrouds at Palisades and FCS in 1995 and 1993, respectively, revealed no panel separation and no missing bolts. Ten-year in-service inspections were performed at FCS in 1992 and will be performed again in 2003 and after the period of extended operation (Reference LIC-13-0059). The results of these inspections, the Palisades in-service inspection results, and the results of industry programs will be monitored to determine if additional action, such as ultrasonic inspection, is necessary. The EPRI MRP is developing an action plan to address potential SCC of reactor vessel internals. OPPD is participating in this program and will take action, as necessary, in response to any recommendations and findings coming from the evaluation.	10-year inspection of core shroud - ongoing beginning prior to the period of extended operation. Implementation of EPRI MRP recommendations - when recommendations are available.	Response to RAI B.2.8-1
17	OPPD has incorporated an augmented inspection of the thermal shield bolting or pins within the Reactor Vessel Internals Inspection Program. OPPD continues to monitor thermal shield vibrations as a task within the Reactor Vessel Internals Inspection Program (B.2.8).	Ongoing	Response to RAI 3.1.3-1

18	The following enhancements will be made to the Reactor Vessel Internals Inspection Program: A fluence and stress analysis will be performed to identify critical locations. A fracture mechanics analysis for critical locations will be performed to determine flaw acceptance criteria and resolution required to detect flaws. Appropriate inspection techniques will be implemented based on analyses. (For the RVI flow skirt)The fracture mechanics analysis committed to in Section B.2.8 of the LRA will be performed. OPPD has opted to implement EPRI MRP-227-A rather than complete FCS specific fluence and stress analyses for critical RV internals locations for a fracture mechanics analysis for the RVI flow skirt (LIC-13-0059).	Prior to the period of extended operation.	LRA Section B.2.8 and response to RAI 3.1.2-3

Commitment 16 specifically states that "the EPRI MRP is developing an action plan to address potential SCC of reactor vessel internals. OPPD is participating in this program and will take action, as necessary, in response to any recommendations and findings coming from this evaluation." The EPRI MRP action plan developed into MRP-227-A and MRP-228, the Aging Management Program (AMP) submitted to the NRC is based on these two documents.

Commitment 17 deals with thermal shield bolting or pins, which is discussed in the AMP. The thermal shield bolts will be inspected along with the other MRP-227-A components.

Commitment 18 deals with fluence and stress analysis and a fracture mechanics analysis. In OPPD letter LIC-13-0059, "Change to License Renewal Commitments Pertaining to Inspection of Reactor Vessel Internals," dated May 10, 2013, OPPD stated:

"In the period since OPPD submitted its LR application, the NRC has approved MRP-227-A as the industry methodology for inspection of RV internals. OPPD submitted the Reactor Vessel Internal (RVI) Component Aging Management Program (AMP) for Fort Calhoun Station, Unit No. 1 in September of 2012 (LIC-12-0144) (ML12276A005) noting that it was based on the guidance of MRP-227-A."

RAI 7:

For three of the "Primary" inspection category components applicable to FCS, MRP-227-A permits a demonstration of fatigue life versus a time-limited aging analysis (TLAA) instead of inspection. These components are the Core Support Barrel Assembly – Lower Flange Weld the Lower Support Structure – Core Support Plate, and, Upper Internals Assembly--Fuel Alignment Plate. Details of these plant-specific fatigue evaluations were not provided in the RVI Program Description.

Clarify whether the current licensing basis (CLB) or current design basis for these components includes either a cumulative usage factor analysis, implicit 7000 cycle maximum allowable stress range reduction analysis, fatigue flaw growth analysis or cycle-based flaw tolerance analysis, and if so, whether the applicable cycle-based TLAA was previously found to be

acceptable in accordance with the TLAA acceptance criterion in either Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1)(i) or (ii). Otherwise, justify why your program does not propose inspections of the components using the MRP-227-A recommended inspection bases if either the CLB does not include an applicable fatigue-based or cycle-based TLAA for each of these components, or if the TLAAs were not found to be acceptable previously in NUREG-1782, "Safety Evaluation Report Related to the License Renewal of the Fort Calhoun Station, Unit 1," in accordance with either 10 CFR 54.21(c)(1)(i) or (ii).

OPPD Response

A fatigue life versus TLAA analysis for the Core Support Barrel Assembly – Lower Flange Weld and the Lower Support Structure – Core Support Plate will not be performed as those components will be examined during the MRP-227-A inspections. Although the core shroud plates at Fort Calhoun Station are full-height, the terminology of full-height shroud plate used in the Applicability column of Table 4-2 in MRP-227-A for the fuel alignment plate applies only to non-bolted full-height core shroud plates as detailed in MRP-232. The Examination Coverage column of Table 4-2 refers to Figure 4-17 which is a figure of the Combustion Engineering (CE) System 80 design fuel alignment plate and not applicable to the Fort Calhoun Station design configuration.

RAI 8:

Table C-1 on page C-5, the licensee stated that the inspection requirements for deep beams in lower support structure are not required for FCS. Since FCS has full-height shroud plates, the inspection criteria should be applicable to FCS. Please confirm.

OPPD Response

Although the core shroud plates at Fort Calhoun Station are full-height, the terminology of full-height shroud plate used in the Applicability column of Table 4-2 in MRP-227-A for the deep beams applies only to plants with non-bolted full-height core shroud plates as detailed in MRP-232. The Examination Coverage column of Table 4-2 refers to Figure 4-19 which is a figure of the CE System 80 lower support structure containing deep beams.

RAI 9:

In Section 4.4.2 in MRP-227-A report, the MRP stated that for all CE units, the American Society of Mechanical Engineers (ASME) Code, Section XI inspection criteria applies to all of the following RVI components binned in "Existing" inspection category—(a) guide lugs and lug inserts and bolts; and (b) fuel alignment plates. However, in Table C-3 in the September 27, 2012 submittal, the licensee stated that the AMP for the aforementioned RVI components is not applicable to FCS, and this position is inconsistent with the MRP-227-A. Please provide an explanation for this inconsistency.

OPPD response

(a) Guide lugs, and lug inserts and bolts do not exist in the Fort Calhoun Station reactor internals design. These items are applicable to the CE plants with welded core shroud designs, not bolted core shroud designs.

(b) Fuel alignment pins do not exist in the Fort Calhoun Station reactor internals design. These items are applicable to the CE plants with welded core shroud designs, not bolted core shroud designs.