October 3, 2005

Mr. R. T. Ridenoure
Vice President - Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - RELIEF REQUEST - ALTERNATIVE TEST REQUIREMENTS FOR CONTAINMENT REPAIRS (TAC NO. MC4653)

Dear Mr. Ridenoure:

By letter dated October 11, 2004, Omaha Public Power District (OPPD/licensee) requested relief from Section XI of the American Society of Mechanical Engineers (ASME) Code for the Fort Calhoun Station, Unit 1. Pursuant to Section 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), OPPD has requested an alternative to the test requirement of ASME Section XI, paragraph IWE-5221, to demonstrate the leak-tight integrity of the repaired containment liner.

The Nuclear Regulatory Commission staff has concluded that the proposed local leak rate test, in conjunction with the planned containment pressure test at the design-basis accident pressure of 60 psig, will provide an acceptable level of quality and safety for demonstrating the leak-tight integrity of the repaired containment liner and welds. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

All work under TAC No. MC4653 is complete.

Sincerely,

/RA/
Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure: Safety Evaluation

cc w/encl: See next page

October 3, 2005

Mr. R. T. Ridenoure
Vice President - Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
Post Office Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - RELIEF REQUEST - ALTERNATIVE TEST REQUIREMENTS FOR CONTAINMENT REPAIRS (TAC NO. MC4653)

Dear Mr. Ridenoure:

By letter dated October 11, 2004, Omaha Public Power District (OPPD/licensee) requested relief from Section XI of the American Society of Mechanical Engineers (ASME) Code for the Fort Calhoun Station, Unit 1. Pursuant to Section 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), OPPD has requested an alternative to the test requirement of ASME Section XI, paragraph IWE-5221, to demonstrate the leak-tight integrity of the repaired containment liner.

The Nuclear Regulatory Commission staff has concluded that the proposed local leak rate test, in conjunction with the planned containment pressure test at the design-basis accident pressure of 60 psig, will provide an acceptable level of quality and safety for demonstrating the leak-tight integrity of the repaired containment liner and welds. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

All work under TAC No. MC4653 is complete.

Sincerely,

/RA/

Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure: Safety Evaluation

cc w/encl: See next page

DISTRIBUTION:
PUBLIC
PDIV-2 Reading
RidsNrrDlpmLpdiv (HBerkow)
KManoly (NRR/DE/EMCB)
RidsNrrPMAWang
WBateman (NRR/DE/EMCB)
RidsNrrLALFeizollahi
RidRgn4MailCenter (DGraves)
JMa (NRR/DE/EMCB)
RidsOgcRp
RidsNrrDlpmLpdiv2
JDixon-Herrity
RidsAcrsAcwMailCenter
DCollins

ADAMS Accession No.: ML051960074 *** Memo dated** **NRR-028**

OFFICE	PDIV-2/PM	PDIV-2/LA	EMCB *	OGC	PDIV-2/(A)SC
NAME	AWang:sp	LFeizollahi	KManoly	MWoods	DCollins
DATE	6/2/05	8/2/05	6/28/05	8/1/05	10/3/05

DOCUMENT NAME: E:\Filenet\ML051960074.wpd

OFFICIAL RECORD COPY

cc:

Winston & Strawn
ATTN: James R. Curtiss, Esq.
1400 L Street, N.W.
Washington, DC 20005-3502

Chairman
Washington County Board of Supervisors
P.O. Box 466
Blair, NE 68008

Mr. John Hanna, Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 310
Fort Calhoun, NE 68023

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Ms. Sue Semerera, Section Administrator
Nebraska Health and Human Services
Systems
Division of Public Health Assurance
Consumer Services Section
301 Centential Mall, South
P.O. Box 95007
Lincoln, NE 68509-5007

Mr. David J. Bannister, Manager
Fort Calhoun Station
Omaha Public Power District
Fort Calhoun Station FC-1-1 Plant
P.O. Box 550
Fort Calhoun, NE 68023-0550

Mr. Joe L. McManis
Manager - Nuclear Licensing
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

Mr. Daniel K. McGhee
Bureau of Radiological Health
Iowa Department of Public Health
Lucas State Office Building, 5th Floor
321 East 12th Street
Des Moines, IA 50319

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUESTS FOR RELIEF FOR CONTAINMENT REPAIR LEAK TESTING

FORT CALHOUN STATION, UNIT NO. 1

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

1.0 INTRODUCTION

During the 2006 refueling outage for Fort Calhoun Station, Unit 1 (FCS), Omaha Public Power District (OPPD) will replace the steam generators, pressurizer, and reactor vessel head. Because these components are larger than the existing equipment hatch, OPPD will cut an access opening through the post tensioned containment concrete and metallic liner. Following the component replacement, OPPD will restore the containment concrete and metallic liner. Paragraph IWE-5221 of Section XI of the American Society of Mechanical Engineers (ASME) Code requires that Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), Appendix J, Type A test be performed after such a repair. OPPD has proposed to perform a local leak rate test on the new weld of the containment metallic liner in lieu of the Type A test. This alternative test will be performed subsequent to the containment pressure test specified in IWL-5200, which is to be performed at the design-basis accident pressure to verify the structural integrity of the containment. By a letter dated October 11, 2004, pursuant to 10 CFR 50.55a(a)(3)(i), OPPD requested this alternative to the test requirement of ASME Code Section XI, paragraph IWE-5221, to demonstrate the leak-tight integrity of the repaired containment liner.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of the ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b)

twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ASME Code of record for the FCS third 10-year interval ISI program, which began on September 26, 1993, is the 1989 Edition of Section XI of the ASME Code, with no addenda.

3.0 TECHNICAL EVALUATION

OPPD stated that it would cut out an access opening in the containment for the upcoming outage and upon completion of the modifications, replace the opening with new tendons and concrete. Cutting of the access opening, will require the tendons in the access opening area to be de-tensioned first and later replaced with new tendons. The replacement tendons will be tensioned after the concrete reaches the required strength, in accordance with ASME Section XI, IWL-4000 requirements.

Once the new concrete meets the design strength requirements of the original design requirements in accordance with ASME Section XI, paragraph IWL-4000, it will be tested in accordance with IWL-5000. Prior to the placement of the concrete, the outside surface of the tendon sheathing, the metallic liner, the reinforcing steel and the surfaces of existing concrete will be visually examined to assure proper surface preparation. After placement of the concrete and tendon tensioning, the containment will be pressure tested at the design basis accident pressure of 60 psig in accordance with IWL-5220 and examined in accordance with IWL-5250 to demonstrate containment structural integrity. A 100 percent VT-1C visual examination of the exterior surface of the new concrete will be conducted prior to, during, and following pressurization.

The exposed reinforcing steel, after the concrete is removed, will be 100 percent VT-1 visually inspected by qualified personnel. The reinforcing steel will be repaired or replaced to meet the original design requirements or ASME Section III, Division 2, requirements.

To cut out the access opening, metallic liners in the opening area will be cut and removed and later re-welded in place. ASME Section XI, Subsection IWE-5221 requires that repair/replacement activities performed on the pressure retaining boundary of Class MC or Class CC components be subjected to a pneumatic leakage test in accordance with the provisions of Title 10, Part 50 of the Code of Federal Regulations, Appendix J, Paragraph IV.A. Part 50 of 10 CFR, Appendix J, Paragraph IV.A states that any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage, be followed by either a Type A, Type B, or Type C test, as applicable for the area affected by the modification. The licensee requested a relief from the Appendix J pressure test, and instead proposed a local pressure test. Prior to the local pressure test, the licensee will test the removed section of the metallic liner with a vacuum box, the channel attachment welds with soap bubbles, perform a 100 percent surface (liquid penetrant or magnetic particle) examination and perform spot volumetric examinations (radiograph at 50-foot intervals at locations specified by the examiner) of the containment metallic liner repair welds. After the containment pressure test is completed, the local leak rate test of the welds will be performed, using a channel welded over the new repair welds. The local leak rate tests will meet American Nuclear Society (ANS) 56.8, “Containment System Leakage Testing Requirements,” which are the same requirements that

Type A, Type B, and Type C tests of 10 CFR, Part 50, Appendix J must meet. The welds to attach the channel over the new repair welds will be performed and inspected in accordance with the ASME Section XI repair program.

The licensee states that the local leak rate test is a superior test for determining leakage at the repaired area as compared to the specified Type A test, because the local leak rate test will directly measure the leakage at the repair area, while Type A test measures total containment leakage. The licensee concluded that the local leak rate test, in conjunction with the planned containment pressure test at the design-basis accident pressure of 60 psig, will provide an acceptable level of quality and safety for the containment metallic liners and welds.

Based on the above discussion, the NRC staff concludes that the licensee's proposal contains the proper inspection and examination provisions for containment liners and welds, and that the proposed local leak rate test, in conjunction with the planned containment pressure test at the design-basis accident pressure of 60 psig, will provide an acceptable level of quality and safety for demonstrating the leak-tight integrity of the repaired containment liner and welds.

4.0 CONCLUSION

Based on its review of the licensee's submittal, dated October 11, 2004, for FCS relief request, the NRC staff finds that the proposed local leak rate test, in conjunction with the planned containment pressure test at the design-basis accident pressure of 60 psig, will provide an acceptable level of quality and safety for demonstrating the leak-tight integrity of the repaired containment liner and welds. The NRC staff authorizes the alternative test requirements for containment repairs as requested pursuant to 10 CFR 50.55a(a)(3)(i). All other requirements of the ASME Code, Sections III and XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: J. Ma

Date: October 3, 2005