September 12, 2003

Mr. R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station, FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 – RELIEF REQUEST - THIRD AND FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN - REQUEST FOR RELIEF RR-8 (TAC NO. MB8717)

Dear Mr. Ridenoure:

By letter dated May 1, 2003, Omaha Public Power District (OPPD) submitted Request for Relief (RR) RR-8 for the third and fourth 10-year inservice inspection interval at the Fort Calhoun Station, Unit No. 1. The licensee provided additional information and revised RR-8 by letters dated June 4 and August 19, 2003. OPPD has cited 10 CFR 50.55a(a)(3)(ii) as the basis for requesting relief for the use of an alternative VT-2 visual examination of exposed surfaces of pressure retaining components for evidence of leakage during pressure testing.

For RR-8, the staff concludes that the licensee’s proposed alternatives are authorized because they provide an acceptable level of quality and safety (area under the reactor pressure vessel) or because compliance with the Code requirements would result in a significant hardship without a compensating increase in quality and safety (piping in sub-hulls, Ion Exchange Room 62 and Purification Filter Vault, and entrenched piping). The licensee’s proposed alternative VT-2 visual examination performed when the subject systems are not pressurized or when the reactor is off-loaded provide reasonable assurance of structural integrity of the subject components. Therefore, the licensee’s proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the area under the reactor vessel and 10 CFR 50.55a(a)(3)(ii) for the piping in sub-hulls, the Ion Exchange Room 62 and Purification Filter Vault, and the entrenched piping for the third and fourth 10-year ISI intervals. All other requirements of the ASME Code, Section III and XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

The staff's evaluation and conclusions are contained in the enclosed safety evaluation. All work under TAC No. MB8717 is complete.

Sincerely,

/RA/

Stephen Dembek, 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

The staff's evaluation and conclusions are contained in the enclosed safety evaluation. All work under TAC No. MB8717 is complete.

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cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD AND FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

REQUEST FOR RELIEF NO. RR-8

FT. CALHOUN STATION, UNIT NO. 1

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

1.0 INTRODUCTION

By letter dated May 1, 2003, Omaha Public Power District (OPPD) submitted a request for relief from the 10 CFR 50.55a requirements as implemented through the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). The staff has reviewed the information concerning the inservice inspection (ISI) program Request for Relief (RR) RR-8 submitted for the Fort Calhoun Station's (FCS) third and fourth 10-year ISI interval. The licensee provided additional information and revised RR-8 by letters dated June 4 and August 19, 2003.

2.0 REGULATORY REQUIREMENTS

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 preservice 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 ten-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 Fort Calhoun Station third 10-year ISI interval is the 1989 Edition and the fourth 10-year ISI interval is the 1998 Edition through 2000 Addenda of the ASME Code.

Inservice inspection of the ASME Code Class 1, 2 and 3 components is to be 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 Commission pursuant to 10 CFR 50.55a(6)(g)(i). Section 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the applicant demonstrates that

(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.

3.0 TECHNICAL EVALUATION

Code Requirement:

ASME Code, Section XI, IWA-5240 requires that during pressure testing, a VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage at normal operating pressure and temperature.

System/Component(s) for Which Relief is Requested:

Reactor coolant, safety injection and chemical and volume control Class 1, 2 and 3 piping.

Code Requirement from Which Relief is Requested:

Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the ASME Code required VT-2 visual examination when the plant systems are at operating pressure and temperature.

Area for which relief is requested:

(1) Area under the Reactor Vessel

Licensee's Proposed Alternative Examination (As stated):

> FCS proposes to conduct a VT-2 visual examination during each refueling outage. This inspection would be conducted while the reactor vessel is not pressurized and nuclear fuel is off-loaded. The inspection will check insulation surfaces for signs of leakage or residue. If signs of leakage or residue are found, additional inspection will be conducted to determine the source. Additional inspection may include removal of insulation to gain visual access to the vessel lower head. Leakage or water on the floor area is not indication of vessel leakage. This area is designed and used as a sump or liquid collection area and water may be expected on the floor of the area.

Licensee's Basis for Requesting Relief (As stated):

> This relief request is submitted pursuant to 10 CFR 50.55(a)(3)(ii). Several piping or component sections in these systems at Fort Calhoun Station (FCS) are considered to have inaccessible external surfaces during the time frame and plant conditions required to reach acceptable test pressure. Several factors are considered when evaluation of a piping inspection area results in an inaccessible area determination. These factors may include ALARA or radiological conditions

(both dose rate and contamination level), the amount of useful information gained should the visual inspection be conducted, amount of work and effort required to obtain access to the area to be inspected, other testing and/or practices which may contribute to assurances that plant piping systems or components are intact and not leaking.

Access to this area is posted as a Restricted High Radiation Area and a Surface Contamination Area. Access is currently prohibited when fuel is loaded in the core. Maintenance and inspection activities are scheduled and performed under the Reactor Vessel during periods when the fuel is off-loaded. Estimated exposure for conduct of a VT-2 visual examination of this area, with insulation in place and fuel in the vessel is 1REM. This is a recurring dose estimate and does not include additional intermittent exposure for decontamination and radiological monitoring of the area for routine entrance and inspection.

Direct visual access of the Reactor Vessel is not available without removal of insulation panels protected by stainless steel sheathing. The best information gained by the VT-2 visual examination would be discoloration on stainless steel sheathed insulation surfaces or evidence of boric acid residue on these surfaces.

Reactor Vessel welds are inspected both by visual and ultrasonic testing (UT) methods from the interior of the vessel at ASME code required intervals. The most recent UT examination showed no significant indications and no evidence of indication growth. The FCS Reactor Vessel has no penetrations in the lower area (including instrumentation penetrations). Unknown leakage from the Reactor Coolant System is closely monitored on a daily basis by surveillance test. Sump levels and alarms also provide a measure of unexpected leakage.

The lower vessel head is of insignificant risk to cracking for the following reasons:

a) The fluence to the welds and plate material is less than 10^{17} n/cm^2, which is below the Generic Aging Lessons Learned (GALL) Report threshold of evaluation.

b) A Pressurized Thermal Shock (PTS) analysis (Reference CEN-636, Revision 02, 7/19/00 which was reviewed and approved in FCS Technical Specification Amendment 199) concludes that the FCS Reactor Vessel beltline welds have conservative chemistry factors for PTS.

c) The pressure stresses that are governing, in a hemisphere are ½ of that of the cylindrical shell.

d) There are no bottom head penetrations which could create stress concentration factors/leakage sources or bimetallic effects.

Staff Evaluation:

ASME Code, Section XI, IWA-5240 requires that a VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage. The VT-2 visual examination is conducted during system leakage tests in accordance with IWA-5211. System leakage tests are conducted at a test pressure not less than the nominal operating pressure associated with 100 percent rated reactor pressure in accordance with IWB-5221. For the reactor vessel, system leakage tests are performed each refueling outage in accordance with Table IWA-2500-1, Examination Category B-P, Item B15.10.

The licensee is proposing an alternative to the Code requirement to perform a VT-2 visual examination in the area under the reactor vessel when at nominal operating pressure associated with 100 percent rated reactor pressure. Access to this area is posted Restricted High Radiation Area and a Surface Contamination Area. Access to this area is currently only permitted during refueling outages and when fuel is off-loaded. The fuel is off-loaded each refueling outage. Estimated exposure to conduct a VT-2 visual examination of this area, with insulation in place and fuel in the vessel, is 1REM and does not include exposure for decontamination and radiological monitoring of the area for routine entrance and inspection.

In order for the licensee to perform a direct visual examination of the area under the reactor vessel, the insulation panels that are protected by stainless steel sheathing are required to be removed. OPPD stated that the best information gained by the VT-2 visual examination would be discoloration on stainless steel sheathed insulation surfaces or evidence of boric acid residue on these surfaces.

OPPD proposed as an alternative to perform a visual examination of the area under the reactor vessel during each refueling outage while the reactor vessel is not pressurized and nuclear fuel is off-loaded. The visual examination will include the inspection of the insulation surfaces for signs of leakage or residue. Furthermore, if signs of leakage or residue are found, OPPD will conduct additional inspections to determine the source. Additional visual inspections may include removal of insulation to gain visual access to the vessel lower head.

In 1992, OPPD performed the Code required visual and ultrasonic examination of the reactor vessel welds from inside the reactor vessel and found no significant indications or evidence of indication growth. Furthermore, there are no penetrations in the lower area of the reactor vessel including instrumentation penetrations. Unknown leakage from the reactor coolant system is closely monitored on a daily basis by surveillance test sump levels and alarms provide a measure of unexpected leakage. OPPD also performed an evaluation regarding the possibility of the risk of the lower vessel head cracking and determined it to be an insignificant risk.

OPPD is proposing to perform an alternative inspection method on the same inspection frequency. OPPD has not requested relief from the inspection frequency in Table IWB-2500-1. The proposed inspection will look for evidence of leakage, as would be performed under the Code requirements, but while the plant is in a refueling outage and when the core is off-loaded. The inspectors will not be encumbered by high temperatures and the radiation field that would be present while the plant is at nominal operating temperature. Since the information gathered under the licensee’s proposed alternative will be essentially the same type of information as under the Code requirements and since the inspection will be performed at the same frequency as required by the Code, pursuant to 10 CFR 50.55a(a)(3)(i), the staff has determined that OPPD's proposed alternative provides an acceptable level of quality and safety.

(2) Safety Injection Piping in “Sub-hulls” (SI-9 and SI-10)

Licensee’s Proposed Alternative Examination (As stated):

> FCS proposes to inspect piping and components in the sub-hulls during periods when the sub-hulls are open for testing and/or maintenance activities. A VT-2 visual inspection will be conducted when the sub-hull is open concurrent with MOV maintenance and/or testing, currently scheduled every five years. Since this piping is sump suction piping it will not be pressurized during the inspection. Bolting and carbon steel surfaces in the sub-hull will be inspected for any indication of leakage or deterioration.

Licensee’s Basis for Requesting Relief (As stated):

> Sub-hulls are special enclosures for valves HCV-383-3 and HCV-383-4 (Containment Sump Suction Valves). These Containment vessels receive a Type B Leakage Rate Test in accordance with 10 CFR 50 Appendix J (at +-60 psig) each refueling outage. The access openings are large bolted manway covers. Removal of these covers solely for inspection would result in undue hardship with no corresponding increase in plant safety. Type B testing would have to be performed after closure of the manway. Additionally, Type B leakage rate testing is conducted on the piping from the sump strainer to the associated valve on a schedule determined by the FCS Containment Leakage Rate Test Program.
>
> Periodic Motor Operated Valve (MOV) testing is conducted on the valves in the sub-hulls at routine intervals, currently every five years.
>
> Sub-hulls are opened intermittently for maintenance and test. No evidence of leakage has been noted during these entries. Containment Leakage Rate Testing results and recent ASME code required IWE inspection of the sub-hulls did not find deterioration or other problems.

Staff Evaluation:

ASME Code, Section XI, IWA-5240 requires that during pressure testing a VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage under normal operating pressure and temperature.

OPPD requested relief from the ASME Code requirement to perform a VT-2 visual examination in the safety injection piping located in sub-hulls (SI-9 and SI-10) when at normal operation pressure and temperature. Sub-hulls are special enclosures for valves HCV-383-3 and HCV-383-4 (containment sump suction valves). As an alternative, the licensee proposed to perform a VT-2 visual inspection when the sub-hull is open concurrent with MOV maintenance and/or testing, currently scheduled every five years. The system is sump suction piping and will not be pressurized during OPPD's alternative inspection. In addition, bolting and carbon steel surfaces in the sub-hull will be inspected for any indication of leakage or deterioration.

The ASME Code-required inspections, MOV testing, Appendix J, Type B leakage testing and OPPD's proposed alternative VT-2 visual examination provide reasonable assurance of the structural integrity of the piping and components contained in sub-hulls (SI-9 and SI-10). Pursuant to 10 CFR 50.55a(a)(3)(ii), the staff has determined that based on the difficulty to remove the large bolted manway covers solely to perform the ASME Code required VT-2 visual examination when at normal operation pressure and temperature, a significant hardship exists without a compensating increase in quality and safety; therefore, OPPD's proposed alternative is acceptable.

(3) Ion Exchanger Room 62 & Purification Filter Vault

Licensee's Proposed Alternative Examination (As stated):

> FCS proposes to conduct VT-2 visual examination of piping in this area at a frequency of every 40 months (once per period). The revised frequency is in accordance with the ASME Boiler and Pressure Vessel Code. The inspection criteria would include evidence of leakage from any piping with additional attention to bolted connections which may have carbon steel fasteners. The VT-2 visual examination would be conducted during convenient outage periods and would not require that piping be pressurized during the inspection.

Licensee's Basis for Requesting Relief (As stated):

> Access to this area is posted as a Restricted High Radiation Area. Estimated exposure for conduct of a VT-2 visual examination of this area is significantly greater than 1REM. General Area dose rate has been estimated at 800REM/hr. Operations (resin sluicing, backwash, etc.) result in intermittent pressurization of piping segments in this area. Several entries would be required to complete the inspection of all piping as required by the ASME Section XI code. Cost to discharge and dispose of resins, specifically to reduce radiation levels in the room, are estimated to exceed $250,000.
>
> Piping in this area is Class 3 and the normal VT-2 inspection requirement frequency is each period (Table IWD-2500-1). The subject piping is small bore (< 3 inches) operated at low pressure (< 240 psia). Planning is in progress to perform VT-2 inspection of the area's piping and components during the 2003 Refueling Outage to establish an inspection baseline. One attribute of the baseline inspection is to identify the presence of any carbon steel fasteners in piping system connections.

The area radiation monitor (RM-082) would detect and trend leakage in this area should it occur. The Reactor Coolant inventory daily monitoring surveillance testing would also lead to quick and positive identification of leakage from this piping.

In 2001 a minor leak was identified at a valve mechanical connection in the Purification Filter Vault. A general area inspection performed during the repair/maintenance of the connection, conducted by Radiation Protection and Maintenance Department personnel, did not find additional evidence of leakage.

Staff Evaluation:

ASME Code, Section XI, IWA-5240 requires that during pressure testing a VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage under normal operating temperature and pressure.

OPPD requested relief from the ASME Code requirement to perform a VT-2 visual examination of the piping located in Ion Exchanger Room 62 and Purification Filter Vault when at normal operation pressure and temperature. OPPD proposed to perform a VT-2 visual inspection of piping in this area at a frequency of every 40 months (once per period) as required by the ASME Code or during convenient outage periods with the piping not being pressurized during the inspection. OPPD is planning to perform VT-2 visual inspections of the area's piping and components during the Fall 2003 refueling outage to establish an inspection baseline to identify the presence of carbon steel fasteners in piping system connections. The piping with carbon steel fasteners is small bore (< 3 inches) operated at low pressure (< 240 psia).

The Ion Exchanger Room 62 and Purification Filter Vault is posted as a restricted high radiation area. The estimated exposure to perform the ASME Code required examinations is estimated to be greater than 1REM and the general area dose rate is estimated at 800REM/hr. To perform the ASME Code required examinations would require the licensee's personnel to make several entries in to the subject area and cause the plant personnel to receive high dosage of radiation. Leakage would be identified from this piping, because OPPD performs daily monitoring surveillance testing of the reactor coolant inventory therefore, providing reasonable assurance of leak tightness of the subject piping.

OPPD's proposed alternative to perform VT-2 visual examinations when the systems are not at normal operating pressure and temperature every inspection period (40 months) as required by the ASME Code provides reasonable assurance of leak tightness of the piping contained in the Ion Exchanger Room 62 and Purification Filter Vault. Pursuant to 10 CFR 50.55a(a)(3)(ii), the staff has determined that based on the high radiation exposure for the licensee to perform the ASME Code required VT-2 visual examination on the piping and components in the Ion Exchanger Room 62 and Purification Filter Vault when at normal operation pressure and temperature, a significant hardship exists without a compensating increase in quality and safety; therefore, OPPD's proposed alternative is acceptable.

(4) Entrenched Piping

- Between purification filters and volume control tank (VCT)
- Between charging pumps and regenerative heat exchanger
- Between reactor coolant pump (RCP) bleed-off and VCT
- Between TCV-211-2 and letdown strainer
- Between charging pumps and high pressure safety injection (HPSI) header
- Between ion exchangers and purification filters

Licensee's Proposed Alternative Examination (As stated):

> FCS proposes piping in the trench be treated similar to buried piping. FCS proposes the piping in the trench receive a VT-2 visual examination if opened for maintenance or modification and/or have a VT-2 visual examination at a maximum frequency of every 10 years (once per interval). All piping would not be pressurized during the inspection. Indications of flow decrease during operation in piping systems contained in the trench will be promptly investigated.

Licensee's Basis for Requesting Relief (As stated):

> This piping is contained in a piping trench covered by eleven large concrete plugs that, although removable, require in excess of 48 man-hours to lift and set aside. The plug weight is nominally 4500 pounds for each of the eleven. The blocks interlock such that a specific installation and removal sequence is required. The entrenched piping is approximately sixty feet in length. Block removal creates a significant disruption in the Corridor 26 area inside the Auxiliary Building since open trenches may create both safety and access problems. This creates a hardship with no corresponding increase in plant safety.
>
> Flow through these piping segments is routinely monitored during plant operations. Changes to volume control tank level are monitored and trended, ion exchanger flows are verified by daily coolant sampling and analysis, plant coolant and letdown operations are closely monitored to ensure the expected plant response is obtained. Significant leakage in any of the entrenched piping would be quickly noticeable.
>
> There are no mechanical joints or components contained in the trench. All piping is stainless steel and has welded joints. Recent (2001) inspection of the area revealed no evidence of past or current fluid leakage.

Staff Evaluation:

ASME Code, Section XI, IWA-5240 requires that during pressure testing a VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage under normal operating pressure and temperature.

OPPD requested relief from the ASME Code requirement to perform a VT-2 visual examination of the entrenched piping when at normal operation pressure and temperature. OPPD proposes that the piping in the trench be treated similar to buried piping and that the subject piping in the trench receive a VT-2 visual examination if opened for maintenance or modification and/or have a VT-2 visual examination at a maximum frequency of every 10 years. The piping would not be pressurized during the inspection. Any indications of flow decrease during operation in piping systems contained in the trench will be promptly investigated.

The trench has large concrete cover plugs that are removable and that require OPPD 48 man-hours to lift and set aside. There are eleven plugs, which weigh 4500 pounds each and have to be removed in sequence. The entrenched piping has no mechanical joints or components and the piping material is of stainless steel. In addition, the system is monitored for leakage during plant operations. OPPD in a recent 2001 inspection of the area found no evidence of past or current fluid leakage.

OPPD's proposed alternative inspection and its ability to identify leakage from this piping provides reasonable assurance of structural integrity of the piping contained in the subject trench. Pursuant to 10 CFR 50.55a(a)(3)(ii), the staff has determined that based on the difficulty to remove the concrete plugs covering the trench to perform the ASME Code required VT-2 visual examination when at normal operation pressure and temperature, a significant hardship exists without a compensating increase in quality and safety; therefore, OPPD's proposed alternative is acceptable.

4.0 CONCLUSION

For RR-8, the staff concludes that the licensee's proposed alternatives are authorized because they provide an acceptable level of quality and safety (area under the reactor pressure vessel) or because compliance with the Code requirements would result in a significant hardship without a compensating increase in quality and safety (piping in sub-hulls, Ion Exchange Room 62 and Purification Filter Vault, and entrenched piping). The licensee's proposed alternative VT-2 visual examination performed when the subject systems are not pressurized or when the reactor is off-loaded provide reasonable assurance of structural integrity of the subject components. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the area under the reactor vessel and 10 CFR 50.55a(a)(3)(ii) for the piping in sub-hulls, the Ion Exchange Room 62 and Purification Filter Vault, and the entrenched piping for the third and fourth 10-year ISI intervals. All other requirements of the ASME Code, Section 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: T. McLellan

Dated: September 12, 2003