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OPPD
Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

June 4, 2003
LIC-03-0083

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference: 1. Docket No. 50-285
2. Letter from OPPD (R. L. Phelps) to NRC (Document Control Desk) dated May 1, 2003, Relief Request: Visual Inspection of Inaccessible Piping and Components (LIC-03-0062)

Subject: Revised Relief Request Pertaining to Visual Inspection of Inaccessible Piping and Components

This letter replaces Omaha Public Power District's (OPPD's) relief request, "Visual Inspections of Inaccessible Piping and Components", submitted in Reference 2. This revised request proposes inspections and/or provides additional information in response to NRC requests made during a May 12, 2003 phone call.

Pursuant to the provision specified in 10 CFR 50.55a(a)(3)(ii), OPPD requests relief from certain requirements of the ASME Boiler and Pressure Vessel Code. These relief requests pertain to IWA-5240, Visual Examination. During Code required 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. The specifics of the revised relief requests are detailed in the attachment to this letter and are intended to be applied to the performance of the inservice inspection (ISI) examination for the Third and Fourth Ten Year ISI Interval.

No commitments are made to the NRC in this letter. If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833.

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/rlj

A047

Attachment: Fort Calhoun Station Revised Relief Request for Visual Inspection of Inaccessible Piping and Components

c: Thomas P. Gwynn, Acting Regional Administrator, NRC Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector

ATTACHMENT

Fort Calhoun Station

Revised Relief Request for Visual Inspection of Inaccessible Piping and Components

ISI PROGRAM RELIEF REQUEST
NUMBER: RR-8

System: Reactor Coolant, Safety Injection and Chemical and Volume Control

Class: Class 1, 2 and 3 piping

Test Requirements:

IWA-5240, Visual Examination: During Code required 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.

Basis for Justification:

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.

Areas for which relief is requested:

1) Area under the Reactor Vessel

Radiological Conditions:

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 1R. 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.

Information Gained by Visual 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.

The area under the FCS reactor vessel is designed as a sump area and used as a water collection point for plant operations and maintenance.

Other Inspections, Practices and or Design Features:

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.

Alternative Testing:

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.

2) Safety Injection Piping in "Sub-hulls" (SI-9 & SI-10)

Other Inspections, Practices and or Design Features:

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 10CFR50 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 code required IWE inspection of the sub-hulls did not find deterioration or other problems.

Alternative Testing:

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.

Reference P&ID: E23866-210-130 3

3) Ion Exchanger Room 62 & Purification Filter Vault

Radiological Conditions:

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 1R. General Area dose rate has been estimated at 800R/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.

Other Inspections, Practices and or Design Features:

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.

Alternative Testing:

FCS proposes to conduct VT-2 visual examination of piping in this area at a frequency of every 10 years (once per interval). 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.

Reference P&ID: E-23866-210-120 2A

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**

Other Inspections, Practices and or Design Features:

This piping is contained in a piping trench covered by 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.

Alternative Testing:

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.

Reference P&ID: E-23866-210-120

Period for which Relief is requested:

This relief is requested for the remainder of the Third Interval and for the Fourth Interval as described in ASME Section XI Code. Code of reference for 3rd and 4th Ten Year intervals are ASME XI 1989 Edition and 1998 Edition, 2000 Addenda respectively. Proposed inspection frequencies will commence when relief is granted.