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10 CFR Part 50
Section 50.55a

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Relief Requests for the 3rd 10-year Inservice
Inspection Intervals for Unit 1 and Unit 2

The purpose of this letter is to request Nuclear Regulatory Commission (NRC) authorization to utilize American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Cases N-533, "Alternative Requirements for VT-2 Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections" and N-566-1, "Corrective Action for Leakage Identified at Bolted Connections" during the Prairie Island Third Ten Year Inservice Inspection Intervals. We are requesting relief pursuant to 10 CFR Part 50, Section 50.55a(a)(3)(i) because the proposed alternatives would provide an acceptable level of quality and safety.

Attached are the two relief requests for both Prairie Island Unit 1 and Unit 2.

In this letter we have made one new Nuclear Regulatory Commission commitment. That commitment is in conjunction with the relief request to use Code Case N-533. In addition to the actions required by the Code Case, we will impose a four hour hold time on the system pressure test prior to the VT-2 visual examination. This commitment would become applicable if the Code Case is approved for use for Prairie Island. If the NRC approves either or both of these Code Cases generically, then we would void these relief requests (as applicable) and follow the NRC published guidance regarding them.

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Prairie Island requests approval by November 1, 2002 to support the refueling outage of Unit 1. Please contact Jack Leveille (651-388-1121, Ext. 4142) if you have any questions related to this letter.

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c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
Chief Boiler Inspector, State of MN
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Attachments:

- ISI Relief Request, Unit 1, No. 12 and Unit 2, No. 13 (Revision 0)
- ISI Relief Request, Unit 1, No. 13 and Unit 2, No. 14 (Revision 0)

ISI Relief Request
Unit 1, No. 12 and Unit 2, No. 13
(Revision 0)

Corrective Action for Leakage Identified at Bolted Connections
Nuclear Code Case N-566-1

Systems: All
Classes: 1, 2, 3
Categories: B-P, C-H, D-A, D-B
Items: B15.xx (all), C7.xx (all), D1.10, D2.10

Examination Requirements:

ASME Boiler & Pressure Vessel Code Section XI (1989 Edition, no addenda) paragraph IWA-5250 (a) (2) states:

> "If leakage occurs at a bolted connection, the bolting shall be removed, VT-3 examined for corrosion, and evaluated in accordance with IWA-3100"

This requires all bolts be removed and inspected from bolted connections when a VT-2 examination has determined there is evidence of leakage at the bolted connection.

Basis for Relief:

In certain situations, it is technically unnecessary to remove and inspect all the fasteners in a bolted connection that has evidence of leakage. The removal and subsequent inspection of all fasteners may not be necessary to maintain piping, component and system integrity. For example, certain fasteners may not be exposed to leaking fluid due to the orientation of the leak, the leaking fluid may have little corrosive effect on the fasteners, or maintenance including fastener inspection or replacement may have just occurred on the bolted joint. Avoiding complete disassembly of radioactive components when a less intrusive method of assuring fastener integrity is available can also reduce personnel radiation exposure. In these types of circumstances, an evaluation can determine the susceptibility of fasteners to corrosion and failure due to leakage and can provide an acceptable level of quality and safety.

ASME Code interpretation XI-1-92-01 states that new bolting or bolting that has received a visual examination prior to installation and has not been in service does not have to be evaluated in accordance with this section. This is recognition by the Code that leakage in this situation is not subject to the requirements of IWA-5250(a)(2).

This alternative is proposed pursuant to 10 CFR 50.55a(a)(3)(i).

Proposed Alternative:

In lieu of the IWA-5250(a)(2) Corrective Measures, Prairie Island Nuclear Generating Plant proposes to use Nuclear Code Case N-566-1, Corrective Action for Leakage Identified at Bolted Connections. Case N-566-1 would be selectively used for those instances where an engineering evaluation that addresses bolting susceptibility to corrosion and failure is more appropriate than removal and inspection of all fasteners.

Justification for Alternative:

Removal of pressure retaining bolting at mechanical connections for VT-3 visual examination and subsequent evaluation in locations where leakage has been identified is not always the most prudent course of action to determine the condition of bolting. Disassembly could have a detrimental effect on the bolted connection. For example, many bolted connections are studs threaded into valves and pumps. Removal of these studs can be difficult due to the length of time they have been installed. Damage to pump casings or valve bodies can occur if too much torque is required to remove a stud. This could require complete disassembly of the pump or valve in order to perform the examination, resulting in significant personnel radiation exposure and risking damage to the component. The Code requirement to remove, examine, and evaluate bolting in this situation does not allow the Owner to consider other factors which may indicate the condition of mechanical joint bolting without performing disassembly and inspection.

Other factors which should be considered when evaluating bolting condition when leakage has been identified at a mechanical joint include, but should not be limited to: joint bolting materials, service age of joint bolting materials, location of the leakage, history of leakage at the joint, evidence of corrosion with the joint assembled, and corrosiveness of the process fluid.

ASME Section XI is written primarily to address examination and testing during periods of plant or system shutdown. No guidance is given to address components that are examined or tested while the plant or system is in service. However, many Code Class 3 and a few Code Class 2 systems are pressure tested and VT-2 visually examined utilizing the "inservice test" requirements of IWA-5000.

Performance of the inservice test may identify leakage at a bolted connection that, upon evaluation, may conclude that the joint's structural integrity and pressure retaining ability is not challenged. It would not be prudent to negatively impact a safety system's availability by removing the system from service to address a leak that does not challenge the system's ability to perform its safety function.

In addition, it is common for Prairie Island Nuclear Generating Plant to perform complete replacement of bolting materials (studs, bolts, nuts, washers, etc.) at mechanical joints during plant outages. When the associated system process piping is pressurized during plant start-up, leakage is sometimes identified at these joints. The root cause of this leakage is most often the thermal expansion of the piping and bolting materials at the joint causing process fluid seepage at the joint gasket. Proper re-torquing of the joint bolting, in most cases, stops the leakage. Removal of the joint bolting to evaluate for corrosion would be unwarranted in this situation due to the new condition of the bolting materials.

The above alternative is proposed pursuant to 10 CFR 50.55a(a)(3)(i) because it provides an acceptable level of quality and safety.

The proposed alternative will provide an equivalent level of fastener integrity because Code Case N-566-1 requires either:

- the leakage must be stopped and an evaluation completed to determine the susceptibility of the fasteners to corrosion and failure or
- if the leakage is not stopped, the joint shall be evaluated in accordance with IWB 3142.4 for joint integrity and the fasteners shall be evaluated to determine their susceptibility to corrosion and failure.

Precedent:

Relief was granted to the Palisades Plant for application of ASME Code Case N-566-1 by letter dated August 26, 1999.

Implementation Schedule:

Implementation is planned to occur during the next refueling outage (November 2002, Unit 1).

ISI Relief Request
Unit 1, No. 13 and Unit 2, No. 14
(Revision 0)

Alternative Requirements for VT-2 Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections (Code Case N-533)

SYSTEM: RCS
Category: B-P

Class: 1
Items: All

Impractical Examination Requirements:

ASME Boiler & Pressure Vessel Code Section XI (1989 no addenda) paragraph IWA-5242 requires that insulation be removed from pressure-retaining bolted connections for visual examination VT-2 on systems borated for the purpose of controlling reactivity. The VT-2 examination is also required to be performed at normal operating pressure and temperature.

Basis for Relief:

Inspection of piping components at high temperatures are impractical because they pose significant safety hazards to personnel performing the examination, including heat stress, burns, and increased radiation exposure.

Performance of the System Leakage Test per the code would require an additional reactor coolant system (RCS) thermal transient and an extension of the refueling outage. Prior to performing the VT-2 visual examination required by the System Leakage Test, the code requires the RCS pressure and temperature to be raised to 2235 psig and 547 °F. The code does allow the pressure and temperature to be lowered after attaining test conditions, however cooling down to perform the examination and the subsequent heat-up imposes an additional transient on the RCS.

After the examination and before reactor startup the insulation has to be reinstalled. The additional cooldown and heat-up of the RCS and the replacement of insulation after the exam would result in an extension of the refueling outage.

Proposed Alternative:

In lieu of the IWA-5242 requirement to remove insulation from Class 1 pressure-retaining bolted connections to perform a VT-2 visual examination:

a) A system pressure test and VT-2 visual examination shall be performed each refueling outage for Class 1 connections without removal of insulation.

b) Each refueling outage, the insulation shall be removed from the bolted connections and a VT-2 visual examination shall be performed. The connections are not required to be pressurized. Any evidence of leakage shall be evaluated in accordance with IWA-5250.

These requirements are specified in Nuclear Code Case N-533 as approved by the Board of Nuclear Codes and Standards.

c) A four hour hold time on the system pressure test will be imposed.

Justification:

The above alternative is proposed pursuant to 10 CFR 50.55a(a)(3)(i) because it provides an acceptable level of quality and safety.

a) The proposed alternative will provide the same or greater level of leak detection/ RCS integrity because:

- A four hour hold time will be imposed on the System Leakage Test thereby increasing the likelihood that any leaking connections will be identified during the VT-2 visual examination.
- After being pressurized for the whole fuel cycle, all connections will undergo an examination under environmental conditions that allow a more detailed inspection than would be possible if the connections were inspected hot. Leakage indications (boric acid residue) will be equally visible at reduced temperature and pressure.

b) The proposed alternative will reduce personnel exposure to hazards such as heat stress, radiation, and burn hazards. Conduct of examinations in hot areas would result in decreased examination efficiency due to shorter examiner stay times and would require the wearing of special protective equipment to protect against heat stress.

c) The proposed alternative will take advantage of established refueling outage activities to inspect for leaks at the beginning of the outage so that any corrective measures could be made a part of the outage scope. The proposed alternative reduces the number of RCS thermal cycles and the refueling outage duration.

Precedent:

Relief was granted (by NRC letter dated April 26, 1999) to the Palo Verde Nuclear Generating Station for application of an alternative consistent with ASME Code Case N-533.

Implementation Schedule:

Implementation is planned to occur during the next refueling outage (November 2002, Unit 1).