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AUTH. NAME AUTHOR AFFILIATION
HINTZ, D. C. Wisconsin Public Service Corp.
RECIP. NAME RECIPIENT AFFILIATION
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SUBJECT: Forwards relief request for inservice insp program, per 10CFR50.55. Info supports determination that conformance w/ ASME code requirements impractical for facility. Fee paid.

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600 North Adams • P.O. Box 19002 • Green Bay, WI 54307-9002

March 20, 1987

10 CFR 50.55a(g)(5)(iii)

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

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Inservice Inspection Plan Relief Request

Wisconsin Public Service Corporation hereby submits the attached relief request for the Inservice Inspection Program at the Kewaunee Nuclear Power Plant (KNPP). Pursuant to 10 CFR 50.55a(g)(5)(iii), we have provided as an attachment to this letter the information which supports our determination that conformance with the ASME Code requirements is impractical for our facility.

In accordance with 10 CFR 170.12 please find enclosed a check for $150.

Sincerely,

D. C. Hintz
Vice President - Nuclear Power

DSN/jms

Attach.

cc - Mr. Robert Nelson, US NRC
US NRC, Region III

A047 w/check $150 #0387-2335
1/1

RELIEF REQUEST NO. RR-1-5

1. Components Affected

Two Class 1 circumferential pipe welds:

	Isometric	Description
RC-W24	M-1144	Weld on 27.5" ID Reactor Coolant Cold Leg Piping
SI-W118	M-1145	Weld on 12" Safety Injection Piping between Valves SI-21A and SI-22A

2. Section XI Requirements

Volumetric and surface examination per 1980W1981, Table IWB-2500-1, Category B-J, Item B9.11.

3. Basis for Requesting Relief

Weld RC-W24 is located inside a penetration which passes through the concrete shield around the reactor vessel. The pipe surface is covered with insulation, and is inaccessible. Weld SI-W118 is located inside a penetration which passes through the floor grating which completely surrounds the weld. Since these welds are inaccessible, volumetric and surface examinations cannot be performed.

4. Alternative Method of Examination

The integrity of these welds will be verified during the system leakage tests which are conducted each refueling outage as required by Table IWB-2500-1, Category B-P, Item B15.50.

RELIEF REQUEST NO. RR-1-6

1. Components Affected

One Class 1 socket weld:

	Isometric	Description
LD-W1S	M-1157	Weld on 2" Letdown Line from Loop B Reactor Coolant Intermediate Leg

2. Section XI Requirements

Surface examination per 1980W1981, Table IWB-2500-1, Category B-J, Item B9.40

3. Basis for Requesting Relief

This weld is located in a floor penetration directly under the "Loop B" intermediate reactor coolant leg. Little clearance (a few inches) exists between the bottom of the reactor coolant piping and the floor penetration therefore, access is severely restricted due to location. Due to access limitations, surface examinations cannot be performed on this weld.

4. Alternative Method of Examination

The integrity of this weld will be verified during the system leakage tests which are conducted each refueling outage as required by Table IWB-2500-1, Category B-P, Item B15.50.

RELIEF REQUEST NO. RR-1-7

1. Components Affected

Two Class 1 Support Brackets:

	Isometric	Description
RV-CS5	M-1194	Reactor Vessel Support Bracket at 90° Location
RV-CS6	M-1194	Reactor Vessel Support Bracket at 270° Location

2. Section XI Requirements

Volumetric or surface, as applicable per 1980W1981, Table IWB-2500-1, Category B-H, Item B8.10. Visual per Table IWF-2500-1.

3. Basis for Requesting Relief

The 1980 edition through winter 1981 addenda of the ASME Code requires that either a volumetric or surface examination be performed during the first and second inspection intervals. A visual or surface examination of this weld cannot be performed due to the access restrictions caused by the reactor vessel concrete shield wall. We plan to volumetrically examine the attachment welds and the vessel wall base metal beneath the two bracket supports from inside the vessel when the core barrel is removed at the end of the second 10-year interval. Table IWB-2500-1, Category B-H, Item B8.10 of the 1980W1981 Code states that deferral of this inspection to the end of interval is not permissible. Performance of this weld examination at any time other than when the core barrel is removed for the vessel weld examination would severely impact a refueling outage.

4. Alternative Method of Examination

A volumetric examination of these welds from the inside surface of the reactor vessel will be performed at the end of the interval when the core barrel is removed for the reactor vessel weld examination. This examination requires the use of a remote inspection tool.

RELIEF REQUEST NO. RR-1-8

1. Components Affected

Five Class 1 Nozzles:

	Isometric
Pressurizer Safety Nozzle	M-1160
Pressurizer Safety Nozzle	M-1160
Pressurizer Relief Nozzle	M-1159
Pressurizer Spray Nozzle	M-1161
Pressurizer Surge Nozzle	N/A

2. Section XI Requirements

Volumetric Examination of nozzle inner radius per 1980W1981, Table IWB-2500-1, Category B-D, Item B3.120.

3. Basis for Requesting Relief

The pressurizer inner radius sections cannot be ultrasonically examined for the following reasons:

1) Course grain found in castings causes sound to be attenuated.
2) Cannot maintain a perpendicular scan to the inner radius section.
3) Difficult to differentiate flaws from normal geometry (clad roll).
4) No ASME Code qualified examination procedures exist to perform this inspection on castings.
5) Section V of the ASME B & PV Code provides no guidance for the design and fabrication of calibration blocks for examination of nozzle inner radius.

WPSC has reviewed several vendor "best effort" outer surface volumetric methods, and we have determined that these procedures do not provide for inspecting all of the Code required examination volume at code sensitivities for casted materials.

4. Alternative Methods of Examination

The surge line (at bottom of pressurizer) is inaccessible for visual exam even when the manway (at top of pressurizer) is removed, therefore, no alternative exam on the pressurizer surge nozzle can be performed. The design of the pressurizer is such that access to the inside of the safety, relief and spray nozzles requires removing the manway and entering through the access port. If the pressurizer manway is removed for purposes of maintenance, the performance of a visual (VT-3) examination of the pressurizer spray, relief and safety nozzle inner radius sections will be considered by the plant staff.

The integrity of these nozzles will be verified during the class 1 system leakage test which is performed after each refueling outage during startup as required by Table IWB-2500-1, Category B-P, Item B15.20.

RELIEF REQUEST NO. RR-1-9

1. Components Affected

Four Class 1 Nozzles:

	Isometric
Steam Generator 1A Hot Leg Nozzle	M-1201
Steam Generator 1A Intermediate Leg Nozzle	M-1201
Steam Generator 1B Hot Leg Nozzle	M-1201
Steam Generator 1B Intermediate Leg Nozzle	M-1201

2. Section XI Requirements

Volumetric Examination of nozzle inner radius per 1980W1981, Table IWB-2500-1, Category B-D, Item B3.140

3. Basis for Requesting Relief

The design of the steam generator is such that access to the steam generator requires removing the manway and entering through the access port. No practical method exists for volumetrically examining the inner radius of the steam generator reactor coolant hot leg nozzles and the reactor coolant intermediate leg nozzles from inside the S/G due to extremely high radiation levels and limited accessibility. The inner radius sections cannot be adequately examined from the steam generator outer surface for the following reasons:

1) Course grain found in castings causes sound to be attenuated.
2) Cannot maintain a perpendicular scan to the inner radius section.
3) Difficult to differentiate flaws from normal geometry (clad roll).
4) Section V of the ASME Code provides no guidance for the design and fabrication of calibration blocks of nozzle inner radius sections.

WPSC has reviewed several vendor "best effort" outer surface volumetric methods, and we have determined that these procedures do not provide for inspecting all of the Code required examination volume at Code sensitivities for casted materials.

4. Alternative Method of Examination

The hot leg and intermediate leg nozzle inner radius can be accessed when the steam generator manway is removed. Quality control checks are periodically performed when the manways are removed and reinstalled. These checks, will verify that no major defects exist.

The integrity of these nozzles will be verified during the class 1 system leakage test which is performed after each refueling outage during startup as required by Table IWB-2500-1, Category B-P, Item B15.30.

RELIEF REQUEST NO. RR-2-5

1. Components Affected

One Class 2 circumferential pipe weld:

	Isometric	Description
RHR-W48	M-1169	Weld on 8" x 10" expander on RHR take off line supplying the RHR pumps

2. Section XI Requirements

Surface examination per 1980W1981, Table IWC-2500-1, Category C-F, Item C5.11.

3. Basis for Requesting Relief

Access to weld RHR-W48 is restricted by pipe restraint #225 and pipe restraint #205 as shown on drawing S-274-13. Weld RHR-W48 cannot be inspected without disassembly of these restraints. Due to access restriction and the unwarranted burden of restraint disassembly, surface examinations cannot be performed on this weld.

4. Alternative Method of Examination

The integrity of this weld will be verified during the system leakage tests which are conducted each inspection period (approx. 3-1/3 years) as required by Table IWC-2500-1.

RELIEF REQUEST NO. RR-2-6

1. Components Affected

   Class 2 heat exchanger brackets:

	Isometric
Regenerative Heat Exchanger Brackets	M-1208

2. Section XI Requirements

   Visual (VT-3) examination per 1980W1981, Table IWF-2500-1.

3. Basis for Requesting Relief

   These support brackets provide support for the regenerative heat exchanger. Relief Request No. RR-2-1 provides relief from volumetrically examining the twelve circumferential welds on the regenerative heat exchanger. The basis for Relief Request No. RR-2-1 and this Relief Request are the same. Radiation levels adjacent to the heat exchanger are between six and seven R/Hr. In order to perform a VT-3 exam of the support bracket assembly, insulation must be removed on the heat exchanger as a portion of the support assembly is covered by the insulation. This will require erection of scaffolding since the heat exchanger is over seven feet tall. The total time for erection of scaffolding, removal of insulation, cleaning, and restoration of insulation could take three to four hours. Personnel could be subjected to a total accumulated dose of up to 28 rem. It is felt that the potential personnel exposure to complete these examinations is impractical in light of the high radiation levels, especially when the associated piping connected to the heat exchanger is exempt from volumetric examination because of line size.

4. Alternative Method of Examination

A visual (VT-3) examination will be performed on the accessible portion of the support bracket without removing the insulation.