Northern States Power Company
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

January 19, 1996

10 CFR 50.55a(a)(3)

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

Request for Approval of Alternative to ASME Code Requirements

During reactor head assemblies during the current Unit 1 refueling shutdown, boric acid residue was noted at the base of the rod position indication stack at full length control rod drive mechanism (CRDM) for each of the following locations: H8, F6, and F8 (Attachment 3). The residue was confined to the upper coil housing on the CRDM stack; no boric acid residue was noted below the upper coil housing on the penetration or reactor vessel head insulation.

The repair options were evaluated and it was determined that the most appropriate repair was the use of a weld buildup rather than removing the defect and performing a weld repair. Weld buildup was an acceptable repair technique because the canopy seal weld does not provide the structural strength or the pressure boundary for the joint. A fracture mechanics analysis was performed to justify not removing the existing defect. Even though the canopy seal does not provide structural strength for the joint, the weld buildup over the canopy seal is considered a repair under the rules of ASME Section XI, IWA-4000 because the welding is performed on pressure retaining components. The need for NRC review and approval of the fracture mechanics analysis was discussed with the NRC Staff on June 1, 1995 and it was determined that no NRC review was required.

Based on N-518.4 of the 1968 ASME Boiler and Pressure Vessel Code, a liquid penetrant examination of the weld buildup is required. However, liquid penetrant examination of the canopy seal weld buildup would be difficult. The canopy seals being

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repaired are located in a high radiation area, with radiation fields of approximately 250 to 450 mr/hr. Additionally, access to the canopy seals being repaired is difficult due to the limited clearance between the adjacent control rod drive housings. The separation between the outer rod travel housings is approximately 7.2 inches. This is not adequate clearance to gain complete access to the inner rod travel housings to perform the liquid penetrant examination of the weld repairs. Final weld surface preparation (grinding), the liquid penetrant examination and the subsequent cleanup would be difficult and time consuming due to the limited access, and personnel performing these operations would incur substantial radiation exposure.

While the liquid penetrant examination specified by N-518.4 would provide indication of surface cracks, the processes used to perform the weld buildup and the visual examination of the welds provide the best measure of the intermediate canopy seal weld buildup acceptability due to the limited accessibility and high radiation fields. The surface to be repaired is examined with an 8x camera during weld surface preparation. The weld buildup is deposited using a fully automatic TIG process. All welding parameters are controlled within the qualified range from a remote panel. The weld puddle/deposit is observed via a 8x camera during every phase of the welding. A final visual examination of the weld surface is completed using the same 8x camera. Much of the welding is observed at the control panel by an NSP Level III inspector. In addition, the post outage hydrostatic test of the reactor coolant system will include a VT-2 inspection of the intermediate canopy seal weld area for leakage.

10 CFR Part 50, Section 50.55a(a)(3) allows the use of alternatives to the ASME Code requirements, when authorized by the Director of the Office of Nuclear Reactor Regulation, if it can be demonstrated that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or

2. Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

In accordance with the provisions of 10 CFR Part 50, Section 50.55a(a)(3), we are proposing the following alternatives to the liquid penetrant testing requirements of N-518.4 of the 1968 ASME Boiler and Pressure Vessel Code for the weld repairs described above:

1. Use of a controlled automatic welding process.

2. Observation of the weld puddle/deposit via a 8x camera during the welding process.

3. A final visual examination of the weld surface using the same 8x camera.

4. Performance of a VT-2 inspection of the canopy seal weld area for leakage during the post outage hydrostatic test.

5. Authorized Nuclear Inservice Inspector approval of alternative testing and NIS-2 acceptance.

We also propose using the above alternatives for the remaining two center intermediate canopy seal welds on Unit 1 during the current shutdown (locations H6 and G7 as preventive measures) and the remaining two center intermediate canopy seal welds on Unit 2 as needed during future shutdowns (locations F6 and G7) for which the same process parameters are used. *Prior to any use of this repair process on the remaining Unit 2 center intermediate canopy seal welds, we will inspect the previously similarly repaired welds to verify that they have provided an effective repair.*

A liquid penetrant examination would provide a more stringent verification of the final weld surface condition and therefore afford an added measure of the quality and safety of the completed weld buildup. However, the liquid penetrant examination does not provide a substantial increase in quality and safety above what is provided by the measures (controlled process, observation of weld process using 8x camera, final 8x visual inspection and hydrostatic test inspection) that have been and will be taken in lieu of the liquid penetrant examination.

An analysis was performed by Structural Integrity Associates to demonstrate that a through-wall flaw could be detected by visual examination which has a flaw size which is sufficiently smaller than the critical flaw size, thus assuring sufficient safety margins. The analysis demonstrated that, under a variety of conservative assumptions, the critical flaw size predicted for the repair geometry is in all cases of significant length. It is likely that a much smaller flaw could be credibly detected by visual examination under 8x magnification. The analysis results are summarized in Attachment 1.

In order to confirm the detectable flaw size, tests were performed by Welding Services Incorporated to evaluate the capabilities of the camera system used in the performance of the weld repair. This testing confirmed that the critical flaw sizes resulting from the Structural Integrity analysis are detectable with margin by the visual inspection technique. A summary of the tests performed and the test results are provided as Attachment 2.

In conclusion, the proposed alternatives (automatic weld process, observation of the process using 8x camera, final 8x visual examination and hydrostatic test inspection) to the liquid penetrant requirements of N-518.4 of the 1968 ASME Boiler Code and Pressure Vessel provide an acceptable level of quality and safety for weld repairs to the intermediate canopy seal welds. Furthermore, compliance with the liquid penetrant examination requirements of N-518.4 of the 1968 ASME Boiler and Pressure Vessel Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

We have made one new Nuclear Regulatory Commission commitment in this letter, indicated as the statement in italics. Please contact Jack Leveille (612-388-1121, Ext. 4662) if you have any questions related to this request.

Michael D Wadley

Michael D Wadley
Plant Manager
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg

Attachments:
1. Evaluation of Limiting Flaws for Structural Integrity in Canopy Seal Repairs at Prairie Island Nuclear Plants
2. Summary of Camera Testing
3. Control Rod Locations (Figure B5-2)

ATTACHMENT 1

EVALUATION OF LIMITING FLAWS FOR STRUCTURAL INTEGRITY IN CANOPY SEAL REPAIRS AT PRAIRIE ISLAND NUCLEAR PLANT

STRUCTURAL INTEGRITY ASSOCIATES, INC

Structural Integrity Associates, Inc.

3315 Almaden Expressway
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June 19, 1995
HLG-95-062

Mr. Richard Cooper
Northern States Power
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr., East
Welch, MN 55089

Subject: **Evaluation of Limiting Flaws for Structural Adequacy in Canopy Seal Repairs at Prairie Island Nuclear Plant**

Dear Mr. Cooper

At the request of the NRC and in support of the use of visual examination rather than dye penetrant examination of the completed weld overlay repairs to canopy seal welds at Prairie Island, Structural Integrity Associates (SI) performed several analyses to determine the critical flaw size in the repaired location. The purpose of these analyses was to demonstrate that a through wall flaw which has a flaw size which is sufficiently smaller than the critical flaw size , could be detected by visual examination thus assuring sufficient safety margins. NSP will review the critical flaw sizes determined in this calculation to confirm than the resulting sizes are detectable with margin by the visual technique.

The analysis results are summarized below.

1.0 Geometry

The design geometry of the repair is illustrated in Figure 1. For the purpose of the present evaluation, the component was modeled as a pipe with outside radius equal to that of the latch housing (3.7 inches) and wall thickness equal to the overlay thickness in the vicinity of the latch housing outside surface (0.31 inches minimum), as shown on Figure 1. Through wall axial and circumferential flaws were evaluated. These geometries are considered to be reasonable representations of the actual design geometry. The model geometries are shown in Figure 2.

2.0 Applied Stresses

For conservatism, the applied stress was assumed to be membrane stress at the Code allowable membrane level ($P_m = S_m$). No distinction was made between the hoop and axial directions in this regard, although realistically, the axial direction should be half of this value. Based upon discussions with plant personnel, there are no bending loads present. Therefore, bending stresses were not considered.

3.0 Material Properties

The allowable stress, S_m was taken to be 16.2 ksi at 650° F, which is typical of 304 stainless steel. The Alloy 625 material of the weld overlay repair has a significantly higher allowable stress at this temperature, so use of the stainless steel value is conservative. The flow stress for this component was taken as 3 S_m.

For linear elastic fracture mechanics evaluations, the K_{Ic} was taken as 135 ksi (in)^0.5, which is very conservative for this material at this temperature.

4.0 Analytical Approach

Two analysis methodologies were employed. The limit load (net section collapse) method is considered most appropriate for evaluation of through wall flaws in this very ductile material. This method is described in Appendix C of ASME Section XI. For comparison, linear elastic fracture mechanics (LEFM) methods were also applied. This approach is very conservative for this material due to the ductile behavior of the material.

A total of four cases were studied. These were:

1. Through wall axial flaw. Limit load.
2. Through wall axial flaw. LEFM
3. Through wall circumferential flaw. Limit load.
4. Through wall circumferential flaw. LEFM.

No Code safety margins were included in this evaluation, since the objective is to get a reasonable view of the relationship between detectable and critical flaw sizes.

The results from each case are summarized below.

5.0 Results

5.1 Through wall axial flaw. Limit load.

This case follows the methodology underlying ASME Section XI, Tables IWB-3641 and Appendix C. The SI program **pc-CRACK** was used to perform the analysis. The conclusion is that an axial flaw could be at least 4.3 inches long before leading to incipient collapse. This is much longer than is physically achievable, since cracking would be expected to be confined to the weld overlay material and vicinity, which, in the axial direction, extends approximately 1 inch.

5.2 Through wall axial flaw. LEFM.

This analysis assumes that brittle failure is the operative mechanism. The **pc-CRACK** program is used with this analysis. A fracture mechanics model of a through wall crack in a cylinder under internal pressure was used, together with an assumed fracture toughness K_{Ic} = 135 ksi (in)^0.5. The conclusion is that for this set of assumptions, the critical flaw length is greater than 5.0 inches.

5.3 Through wall circumferential flaw. Limit Load.

This analysis used hand calculations using the methods of Section XI Appendix C. The SI program **ANSC** was also used to perform a separate analysis of the same configuration. The analysis assumed a through wall circumferential flaw, and determined the critical flaw length using limit load techniques. The conclusion is that such a flaw could be 125° around the cylinder before reaching a critical size. This corresponds to a flaw approximately 8.1 inches long.

5.4 Through wall circumferential flaw. LEFM.

This analysis assumed that the failure mode was brittle failure. The **pc-CRACK** program was used with a through wall circumferential flaw in a cylinder under remote tension fracture mechanics model. A K_{Ic} = 135 ksi (in)^0.5 was conservatively assumed. The conclusion of this analysis was that the critical flaw length for this set of assumptions was 7.8 inches.

6.0 Conclusions

The above results demonstrate that, under a variety of conservative assumptions, the critical flaw size predicted for the repair geometry is in all cases of significant length. It is likely that a much smaller flaw could be detected by a visual examination under 8x magnification, as proposed by NSP. NSP should confirm their detectable flaw size, to complete the demonstration that the visual examination provides adequate assurance of safety.

Please call if you have any questions regarding the above.

Sincerely,

H L Gustin

H. L. Gustin, P. E.
Associate

/jj
attachment

cc: NSP-18Q-102

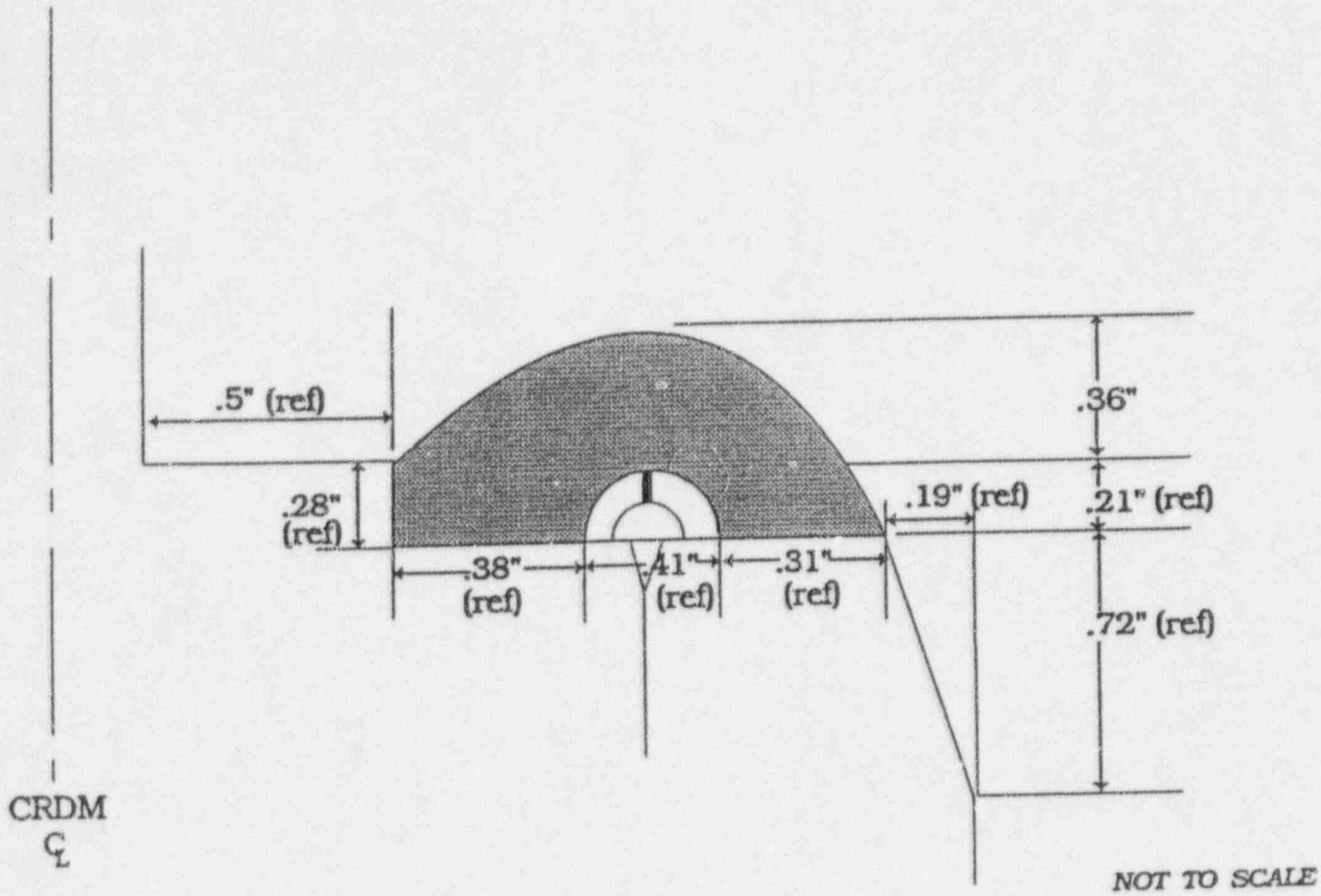
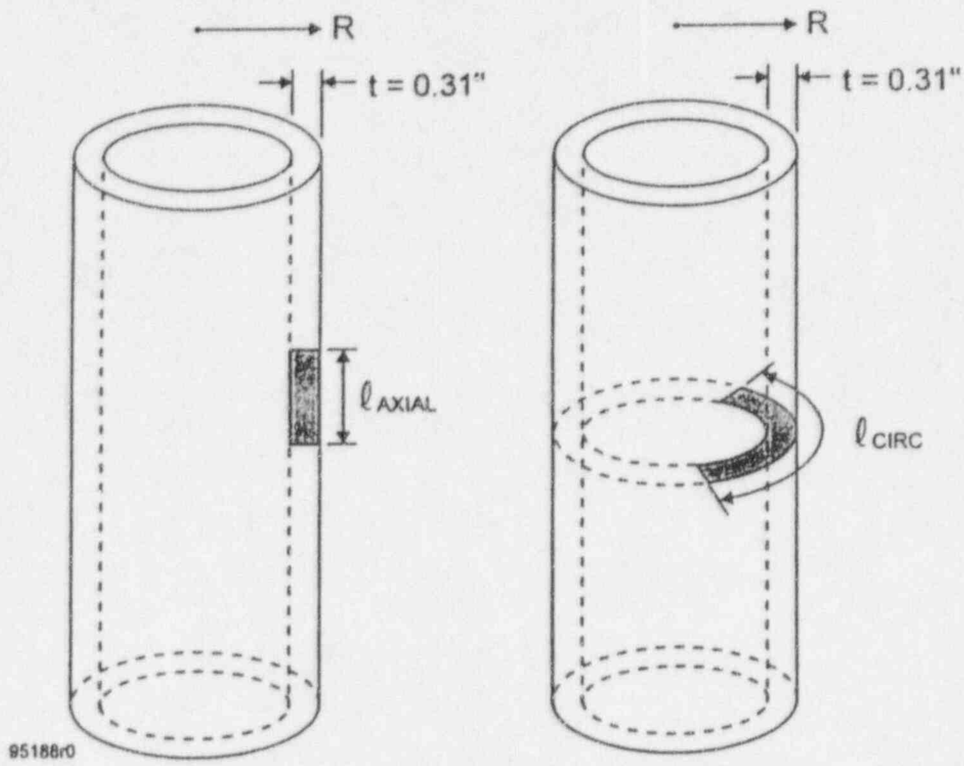


Figure 1. Canopy Seal Weld Overlay Design and Dimensions used in Analysis



$$R = \text{Outside Radius of Latch Housing} = \frac{7.386}{2}$$

Figure 2. Modeled Geometries

ATTACHMENT 2

SUMMARY OF CAMERA TESTING

WELDING SERVICES INCORPORATED

Welding Services Inc.
2225 SKYLAND COURT
NORCROSS, GEORGIA 30071
TELEPHONE (404) 452-0005
FAX (404) 729-8242

June 21, 1995

Mr. Dick Cooper
Northern States Power Company
1717 Wakonade Drive East
Welch, MN 55089
FAX: 612-330-7603

SUBJECT: Prairie Island Nuclear Plant
WSI Reference No.: 35049-2

Dear Mr. Cooper:

Per your request, we have performed several tests to evaluate the capabilities of the camera system used in the performance of the weld repair of Prairie Island's CRDM Seals. The intent of this testing is to provide data to be used by NSP to evaluate the adequacy of this camera for the performance of adequate visual inspection of the weld overlay. The testing described was not performed as a safety related procedure. The testing was performed as follows:

1. The video front end of the WSI weld head was connected to a VCR and monitor of the same make and model as the system used on site.

2. A mockup of a canopy seal housing similar in configuration to the Prairie Island design was overlayed in a similar configuration as the repair performed at the site. A .0005 inch diameter wire and a .001 inch diameter wire each, .4 inches long, were taped to the surface of the weld overlay on the housing.

3. The two wires were filmed using the weld head front end, and the weld head lighting for illumination.

4. WSI's site QC representative, Gary Caul, reviewed the tape and was able to see both wires on the surface of the weld.

A copy of the tape, samples of the two wires used, and additional camera information are included with this letter for your review. Please let me know if you need additional information.

Sincerely,

Pedro E. Amador

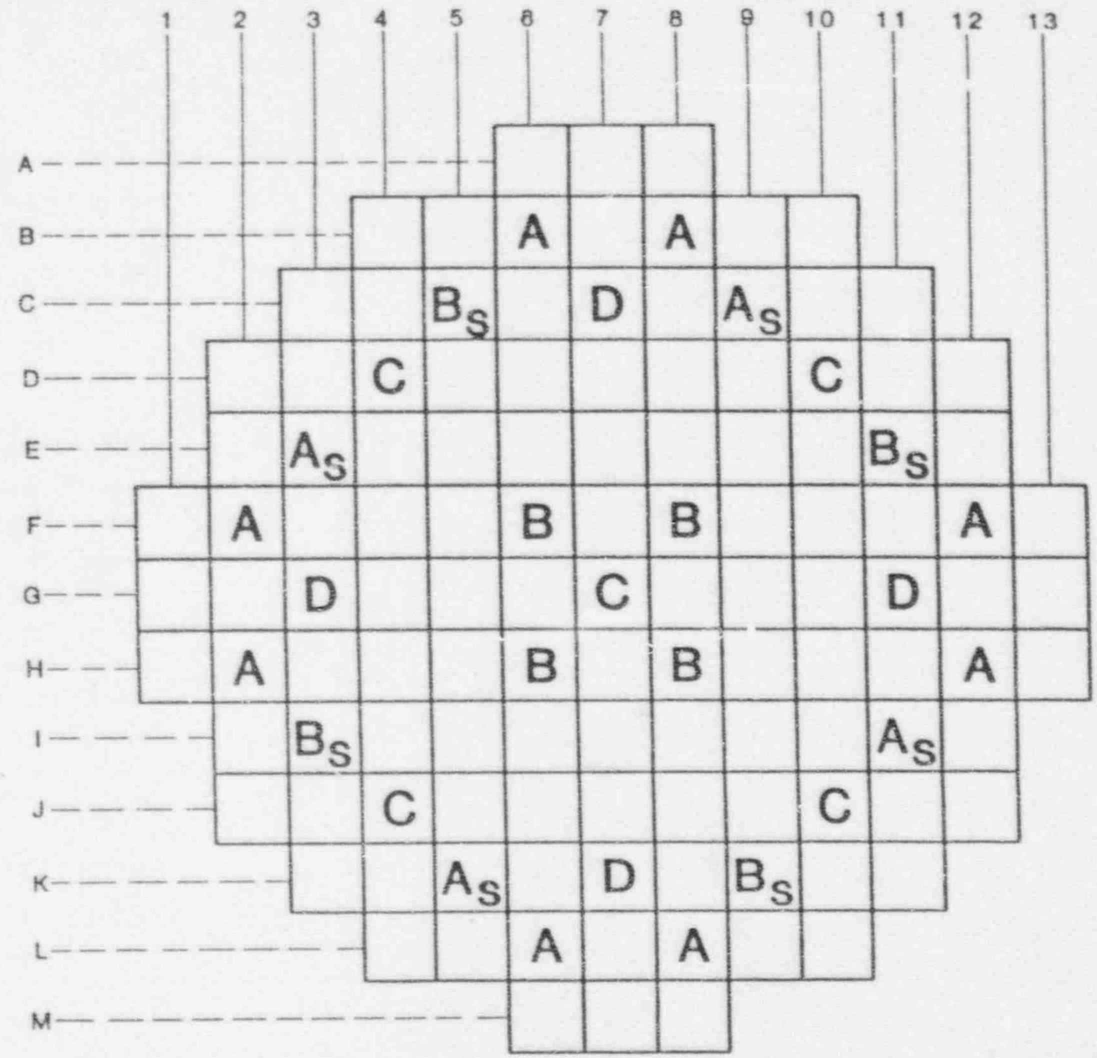
Pedro E. Amador
Senior Project Manager

35049-2.DOC

ATTACHMENT 3

CONTROL ROD LOCATIONS

SYSTEM DESCRIPTION FIGURE B5-2



ROD LOCATIONS

REVISION 0 B5-2