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

February 13, 1998

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

Prairie Island Unit 2 shut down on January 24, 1998 to repair a small RCS leak. The source of this leakage has been identified as a flaw in the wall of the part length control rod drive mechanism (CRDM) at location G9 on the reactor vessel head (Attachment 3). The flaw is located on the CRDM motor tube base approximately 1.5 inches above the intermediate canopy seal weld.

Because the part length CRDMs are not used and are abandoned in place, it was decided that the part length CRDM at location G9, would be permanently removed. This would eliminate the need for repair of the flaw and would facilitate the metallurgical evaluation of the flaw. Removal of the G9 part length CRDM was completed on February 8, 1998. The part length CRDMs at locations I-7 and E-7 have also been removed and plans are to remove the last part length CRDM at G-5 during the current Unit 2 forced outage. Removal of the part length CRDMs requires capping of the associated reactor vessel head penetrations. The preferred method for capping of the penetrations is the installation of a cap which would be screwed onto the threaded end of the penetration and then seal welded.

Based on N-518.4 of the 1968 ASME Boiler and Pressure Vessel Code, a liquid penetrant examination of the seal weld is required. However, liquid penetrant examination of the seal weld would be difficult. The CRDM penetrations being repaired are located in a high radiation area, with radiation fields of approximately 1000 mr/hr. Additionally, access to the seal welds 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

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the seal welds. Final weld surface preparation, 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 seal welds and the visual examination of the welds provide the best measure of the seal weld acceptability due to the limited accessibility and high radiation fields. The surface to be seal welded is examined with an 8x camera during weld surface preparation. The weld 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 an 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 seal weld and CRDM penetration cap 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:

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 seal weld and CRDM penetration cap for leakage during the post outage hydrostatic test.

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

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 seal weld. 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 nave been and will be taken in lieu of the liquid penetrant examination. In addition, due to the time consuming nature of the examination r[illegible]ersonnel would incur substantial radiation exposure during the perform[illegible] penetrant examinations.

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 and Pressure Vessel Code provide an acceptable level of quality and safety for the seal welds on the part length CRDM penetrations. 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.

Because it may be necessary to remove the part length CRDMs from Unit 1 as well, we request that the NRC Staff approve the proposed alternative to ASME Code requirements for both Units 1 and 2.

Because the G9 part length CRDM has been removed from the reactor vessel head, the request for approval of alternate to ASME Code requirements, dated January 30, 1998, is no longer required and should be considered withdrawn.

We have made no new Nuclear Regulatory Commission commitments in this letter. Please contact Gene Eckholt (612-388-1121) if you have any questions related to this request.

Joel P. Sorensen

Joel P. Sorensen
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. Calculation Package: Evaluation of Limiting Flaws for Structural Adequacy in CRDM Repair Adapter Plug Fillet Weld Evaluation at Prairie Island Unit 2
2. Summary of Camera Testing
3. Control Rod Locations

ATTACHMENT 1

CALCULATION PACKAGE:

EVALUATION OF LIMITING FLAWS FOR STRUCTURAL ADEQUACY

IN CRDM REPAIR ADAPTER PLUG FILLET WELD EVALUATION AT

PRAIRIE ISLAND UNIT 2

STRUCTURAL INTEGRITY ASSOCIATES, INC

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February 12, 1998
MLH-98-006

Mr. Dick Cooper
Northern States Power Company
Prairie Island Nuclear Generation Plant
1717 Wakonade Drive East
Welch, MN 55089

Subject: Evaluation of Critical Flaw Sizes for CRDM Canopy Fillet Weld

Dear Mr. Cooper:

Structural Integrity Associates has performed an evaluation to determine the critical flaw size for the subject location. The results of this evaluation can be used to demonstrate that the critical flaw size is significantly larger than the flaw size observable using visual inspection techniques. This can serve to eliminate the need to perform dye penetrant testing of the fillet weld.

The evaluation was performed using limit load methods since the material is ductile and the fluence at this location is below that needed to impact the material fracture toughness. The evaluation was performed assuming a through-wall axial flaw and a through-wall circumferential flaw. The stress in the fillet weld was assumed equal to the design stress intensity of the material (S_m). The S_m for the stainless steel at 650°F was used in this calculation (16.2 ksi). A safety factor of 1.0 was used since the critical flaw size is being calculated.

The calculations were performed by assuming a pipe with radius and thickness equal to that at the fillet weld location (0.265" equivalent thickness, mean radius =3.3 inches). The results for the critical flaw lengths (in terms of fraction of circumference and characteristic length parameter, fraction of $\sqrt{Rt}$) are independent of the pipe geometry since the stress is set to S_m. The critical axial flaw was determined using the SI program, p...CRACK, and the critical circumferential flaw size was determined using the EPRI DLL program.

The results of the evaluation are shown below.

Critical Through-Wall Axial Flaw Length: 4.17 inch
Critical Through-Wall Circumferential Flaw Length: 6.95 inch

It should be noted that this calculation was performed using an applied stress equivalent to the S_m for the stainless steel material, which is conservative. If the actual primary stress at this location were used, the critical flaw lengths are expected to be even larger.

I have included the output listing for the two cases considered in this calculation. I will forward the entire calculation package to you shortly as soon as it is prepared.

Sincerely,

M. L. Herrera, P. E.
Senior Consultant

Attachments
cc: H. L. Gustin
R. A. Mattson
G. A. Miessi
NSP-27Q