U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-282/87018(DRS); 50-306/87017(DRS)

Docket Nos. 50-282; 50-306 Licenses No. DPR-42; DPR-60

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Facility Name: Prairie Island Nuclear Generating Plant Units 1 and 2

Inspection at: Prairie Island Site, Red Wing, Minnesota
Fluor Engineers Inc., Chicago, Illinois

Inspection Conducted: December 16, 1987, and January 13, 1988, at Fluor
December 17-18, 1987, at the site

Inspector: J. A. Gavula — 3/18/88 Date

Approved By: D. H. Danielson, Chief — 3/18/88 Date
Materials and Processes Section

Inspection Summary

Inspection from December 16, 1987 through March 8, 1988 (Report No. 50-252/87018(DRS); 50-306/87017(DRS))

Areas Inspected: Special announced inspection of activities on IE Bulletin 79-14 (92703) and licensee action on previously identified items (92701).

Results: No violations or deviations were identified.

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DETAILS

1. Persons Contacted

Northern States Power Company (NSP)

*D. Mendele, General Superintendent, Engineering and Rad. Prot.
*J. Goldsmith, Superintendent, Nuclear Technical Services
*L. Anderson, Shift Supervisor
°*C. Baltos, Engineering Associates
°*G. Gore, System Engineer
*G. Rolfson, NTS Engineer
°G. Miller, Superintendent of Operations Engineering

Fluor Engineers, Inc. (FEI)

*B. Dickerson, Principal Mechanical Engineer
W. Brennen, Project Engineering Manager
C. Agan, Project Manager
G. Bartholomees, Quality Assurance Manager

*Denotes those attending the interim exit meeting at Prairie Island on December 18, 1987.

°Denotes those participating in the final telephone exit interview on March 8, 1988.

2. Licensee Action on Previous Inspection Findings

a. (Closed) Violation (282/85015-01A, B; 306/85012-01A, B):

NSP did not investigate the cause of the observed steam generator snubber (SGS) hydraulic fluid leakage nor the cause of the fluid contamination. After identifying this problem, no other snubbers were tested to determine the extent of the problem. Also, steam generator snubber design loads were increased by Westinghouse (W), but were not evaluated.

As a result of the above violation, all sixteen of the steam generator snubbers were functional tested and inspected. No hydraulic fluid leakage was noted during this process. All snubbers, except one, passed all the acceptance criteria. The one "failure" noted was a slightly high bleed rate. The investigation into this indicated that the problem was related to a slight warpage of the bleed valve seat. After relapping the valve seat, the snubber met all functional acceptance criteria.

Also during this time, the snubber hydraulic fluid was analyzed by the NSP Testing Laboratory. The results of the tests were inconclusive regarding fluid contamination. Based on the success of the previously discussed functional tests, the degree of contamination was considered inconsequential. In a letter dated January 27, 1988, NSP committed to monitor the hydraulic fluid condition for particulates and viscosity changes in order to address future concerns.

A review of the snubber design load calculation was also made. Based on this review, it was determined that the new design load for the steam generator snubbers was within the original design specification for the snubbers.

Based on the above information, the corrective actions identified in NSP's response dated October 18, 1985 have been adequately implemented and this item is considered closed.

b. (Closed) Violation (282/85015-02A, B, C; 306/87012-02A, B, C): NSP did not report: (a) steam generator snubber leakage and fluid contamination, (b) the need to replace steam generator snubber control valves in order to increase the locking velocities and (c) the significant increase in snubber loads well above the design capacity of the snubbers, in accordance with the requirements of 10 CFR 50.73.

The NRC inspector reviewed Administrative Control Directive, No. 5 ACD 3.6, "Reporting", Revision 5. This directive governs the identification, notification, investigation and reporting of events including those required by 10 CFR 50.73.

Based on the review of the above directive and its inclusion in the Operational Quality Assurance Program, the corrective actions identified in NSP's response dated October 18, 1985 have been adequately implemented and this item is considered closed.

c. (Closed) Unresolved Item (282/85015-04; 306/85012-04): The results of the W dynamic analyses concerning the potential consequences of low locking velocities for the steam generator snubbers could not be verified.

The following additional documentation was reviewed by the NRC inspector:

W letter (NSP-85-625), dated July 10, 1985
W letter (NSP-85-639), dated August 5, 1985

Based on the information presented in these letters as well as the associated calculations for determining the maximum steamline break loads, this item is resolved and considered closed.

d. (Closed) Unresolved Item (282/85015-05; 306/85012-05): The FEI Specification No. 287 was not compared to the W E-Specification for compliance with critical parameters on the steam generator snubbers.

The following documentation was reviewed by the NRC inspector:

W letter (NSP-85-692) dated November 27, 1985
W letter (NSP-87-247) dated December 11, 1987
W letter (NSP-87-193) dated August 27, 1987

Based on the initial comparison between the two specifications, the following three areas were found to have potentially significant differences:

(1) Design radiation levels of 1.2 Rads/hr specified by FEI were lower than the 25 Rad/hr generically specified by W.
(2) Snubber stiffness value verification was not specified by FEI.
(3) Requirements for establishing the snubber lock-up velocity were not specified by FEI.

To address these differences, the following actions were taken by the licensee:

(1) Actual radiation levels were measured in the areas of the steam generator snubbers. Based on these plant specific measurements, the radiation levels at the snubber seals were calculated to be less than 1.2 Rads/hr.

(2) Actual snubber stiffnesses were measured during recent refueling outages. The resulting stiffness of 7900 Kips/inch was reconciled by W in their primary loop support system. According to W (letter NSP-87-193) the change in loadings was readily accommodated by the margins available in the system structural analyses. The primary loop piping stresses, primary equipment support loads and nozzle loads have been reconciled.

(3) Based on concerns regarding potentially low lock-up velocities, the control valve assembly was modified to increase the lock-up velocities to approximately 1 inch/minute. Recent snubber tests results show that the locking velocities are in the 2 inches/minute to 3 inches/minute range. The maximum thermal transient velocity is calculated to be slightly less than 0.6 inches/minute. Therefore snubber lock-up during a thermal transient is not likely.

Based on the above actions, the FEI specified steam generator snubbers should perform adequately. This item is considered closed.

e. (Open) Unresolved Item (282/85016-02; 306/85015-02): Spherical bearings in the steam generator snubbers were found cracked.

Based on evaluations of the cracked bearings by FEI dated March 31, 1986 and by W dated July 1986, it was concluded that bearing misalignment during proof testing was the probable caused of the observed failures. In addition, some concern was expressed by W that cracking in the bearing material was induced by "cold cracking" due to trapped hydrogen.

A finite element stress analysis was performed on the spherical bearing inner ring by FEI. This analysis concluded that the bearings will continue to transfer the load from the pin to the snubber even if a crack develops. On this basis, continue use of the exiting bearings is justified and no safety concern exists. However, it was recommended that all cracked bearings be replaced. Additionally, to alleviate the cold cracking concern, it was recommended that replacement bearings should be baked at 400°F for at least 24 hours to remove any trapped hydrogen in the bearing material.

The cracked bearings in SGS-1 were replaced and the unit was proof tested to 900 kip in compression and 450 kips in tension. Subsequent dye penetrant inspections found no indications of any cracks. All other steam generator snubber bearings were also examined using dye penetrant. Two additional bearings were discovered with cracks following full load tests during the Unit 1 Spring outage in 1986. All cracked bearings were replaced with new bearings, however, the replacement bearings had not undergone the recommended 24 hour bake-out period. Pending a commitment to replace the existing bearings with "baked-out" bearings or additional justification as to why the existing bearings with no remedial action are acceptable, this item will remain open.

f. (Open) Unresolved Item (282/85018-03; 306/85015-03): Because of a cracked spherical bearing and structural interferences on an ITT-Grinnel steam generator snubber, NSP procurement and ITT-Grinnell design control measures were questioned. In addition, the following aspects of the ITT-Grinnell snubbers required additional review:

(1) Seal material certifications did not state that all seals have met TS No. 287 Code requirements.

(2) Seal life expectancies were not stated.

(3) Design engineer review and approval of the snubber's seal design and selection in accordance with TS No. 287 was not apparent.

(4) ITT Grinnell test procedures with instrument calibration data were not available for review.

(5) Evaluation and resolution of the cracking of the spherical bushing, that occurred during qualification tests was not included in the test report package.

Based on the information presented in the Teledyne Engineering Services Technical Report, TR-6860-1, "Summary of NSP Responses to US NRC Report," Items 1, 2, and 3 above have been adequately addressed by ITT Grinnell. Item 4 is no longer relevant since functional tests have been performed as documented in NRC inspection report No. 50-282/85018; 50-306/85015. Item 5 above has been addressed as part of item 282/85018-02; 306/85015-02 (Paragraph 2.f of this report). No further reviews are required for the above five items.

Pending reviews of the NSP procurement process and the ITT Grinnell design control measures as they relate to the receipt of a snubber with a cracked bearing and why the noted structural interferences were not caught in the design process, this item will remain open.

3. Licensee Action on I. E. Bulletins

a. (Closed) I. E. Bulletin 79-14 (282/79014-BB; 306/79014-BB, 282/79014-B1; 306/79014-B1, 282/79014-B2; 306/79014-B2, 282/79014-B3; 306/79014-B3) Seismic analysis for as-built safety related piping systems.

(1) Background

Two previous NRC inspections have reviewed portions of the licensee's actions for IEB 79-14. As documented in NRC inspection report 50-282/79022; 50-306/79017, applicable procedures for implementing the requirements I.E.B. 79-14 were reviewed. The procedures addressed the quality assurance and personnel qualification requirements for conducting as-built walkdowns, the attributes for piping or pipe supports included in these walkdowns, the acceptance criteria for these attributes, and the evaluation of the seismic analyses. During this time, portions of the walkdown inspection records were also reviewed. No adverse comments were made regarding the above items except for a lack of guidance on the timing of any nonconformance evaluation. This aspect was only critical during an interim period between the nonconformance discovery and modification implementation. Since all IEB 79-14 modifications have been completed, this aspect is no longer relevant.

The second NRC inspection was conducted by the NRC Vendor Branch and was documented in NRC inspection report 9900523/79-01. This special inspection was conducted specifically to review the activities of the architect-engineering organization for IEB 79-14. This inspection reviewed the guidelines used to identify nonconformances and the schedule for completing these activities. The inspection also reviewed the identification of seismic analysis input, the documentation for nonconformance analyses/results, the training and indoctrination of project personnel and the planning for conducting the overall program. Six analytical packages were reviewed in detail during this inspection. No deviations or unresolved items were identified during the inspection. However, one followup item was noted regarding a recent modification to the Unit 1 cooling water system. Further information was needed to address the affects of adding a 4" pipe to the existing system.

(2) Current Inspection Activities

(a) Procedure Review

The NRC inspector reviewed relevant portions of the following procedures/instructions to determine if they comply with licensee commitments and NRC requirements.

- "Guideline for the Review of the Original I.E. Bulletin 79-14 Reconciliation Packages," Revision 0, August 1987 and Revision A (Draft) January 1988.
- "Procedure for Inspection of Piping Floor and Wall Penetrations in Accordance with IE Bulletin 79-14," Walkdown Procedure PI-87-79-14, April 1987.
- "Procedure for the Review of Piping Floor/Wall Penetration Clearances," Procedure No. 832642-1, August 1987.
- "Installation and Construction Test Procedures," No. N1AWI 5.1.13, Revision 1, June 10, 1986.
- "Engineering Change Request," No. N1AWI 5.1.15, Revision 1, June 10, 1986.
- "Modification Close-out," No. N1AWI 5.1.17, Revision 1, June 10, 1986.

The first three procedures were recently issued for reviews of the original IEB 79-14 work. As a result of deficiencies identified at another site, FEI implemented a review of the original reconciliation work and also performed additional inspections at the site for specific attributes. To date no significant deficiencies have been identified at the Prairie Island site.

The second three procedures are part of the NSP Administrative Work Instructions at Prairie Island. These instructions implement the current requirements to meet I.E.B. 79-14 directives.

No adverse comments were made during the review of these procedures.

(b) Field Walkdowns

Portions of the following subsystems were walked down by the NRC inspector in order to verify conformance with the as-built data and applicable drawings. These walkdowns included verification of pipe lengths and orientations as well as support locations, types and directions.

Unit 2 Auxiliary Feedwater System
Unit 2 Safety Injection Pump Suction
Unit 2 Component Cooling

With the exception of the last subsystem all dimensions given on the as-built drawings for piping and support locations were within the given acceptance tolerances. For the last system, a vertical support next to snubber CCH-350 was not indicated on the drawing. Further investigation revealed that although this support was not shown on the drawing, this discrepancy had been noted in the field and had previously been included in the reconciliation of the system. No safety significance was associated with this problem and it appeared to be a documentation problem.

No violations or deviations were noted during this review.

(c) Seismic Analysis Reviews

During inspections at FEI, portions of the following analytical packages were reviewed by the NRC inspector. The reviews consisted of a comparison between the as-built piping information and the as-analyzed piping configurations. Overall pipe lengths, valve locations, support types and support locations were compared to documented field data. In those cases where differences were noted, the reconciliation calculations were also reviewed to verify compliance with applicable procedures and to determine if reasonable justifications were provided.

- Packages No. 103 and No. 124-Auxiliary Feedwater System, Unit 2.

  Supports No. AFWH-5 and AFWH-12 were noted as exceeding the acceptance criteria of one pipe diameter for support locations. The reconciliation evaluation was somewhat vague and relied on a degree of engineering judgement. Based on the NRC inspector's judgement, the discrepancies were not significant enough to cause any safety concerns.

- Package No. 7-Component Cooling system, Unit 1.

  This is the system noted in the Vendor Branch inspection as requiring followup. It was initially noted that the addition of the 4" diameter line to the 8" diameter pipe met the established moment of inertia decoupling criteria and therefore was not required to be accounted for. In response to the NRC's concerns, however, the system was reanalyzed to account for the 4" pipe addition. The resulting stresses were relatively low and easily met FSAR stress limitations.

- Packages No. 201 and No. 265-Component Cooling Water. This is a 1" diameter piping system that was initially analyzed using computerized techniques. On this basis it was necessarily included in the original I.E.B. 79-14 review program. No adverse comments were made by the NRC inspector.

It was pointed out to the NRC inspector that a review program is currently underway at FEI to reassess the original IEB 79-14 work. This effort resulted from problems identified at Kewaunee by the NRC, and FEI's recognition that some inconsistencies have been found in applying and documenting the original reconciliation criteria at Prairie Island. On this basis, a comprehensive review of all previous IEB 79-14 work was implemented. Pending the final outcome of this current effort, this will be considered an Open Item. (282/87018-01; 306/87017-01)

(d) Quality Assurance Reviews

For additional verification of the IEB 79-14 work, the NRC inspector reviewed the Quality Assurance audits performed at FEI by NSP and by FEI itself. Although no specific audit was performed to focus on IEB 79-14 work, several audits were performed that covered portions of the subject program.

An audit performed by NSP in May of 1980 addressed Design Control in general and looked at Residual Heat Removal work and Safety Injection Piping work. No significant deficiencies were identified.

An internal audit of FEI conducted in October of 1982 reviewed Design Verification. This audit reviewed specific Main Steam Piping calculations and drawings. Again, no significant deficiencies were identified.

An internal investigative committee was established in September of 1979 to review the generic applicability of uplift loads on rod hangers and the potential need to issue a 10 CFR Part 21 report. This committee concluded that no "substantial safety hazard" existed as a result of this issue and that a Part 21 report was not needed.

Based on the above information adequate management involvement was applied to the IEB 79-14 work.

No violations or deviation were identified.

(e) Current Modification Program

The procedures for implementing the current modification program were reviewed as previously documented in this report. As-built drawings are currently required in order to close-out any modification. The changes to design drawings are controlled under the Engineering Change Request (ECR) system and require engineering review for final resolution. As part of the ECR closure, the responsible engineer must confirm the application of changes in the as-built drawings.

Several recent ECR's were reviewed by the NRC inspector to confirm that all change requests were reviewed and approved by the engineering organization and that the changes were indicated as being incorporated in the design documents. In each case reviewed, all applicable procedures appeared to be met.

No violations or deviations were identified at this time.

(3) Conclusions

The implementation of the IEB 79-14 program at Prairie Island appears to have met the intent of the licensee's commitments and NRC requirements. Although some minor documentational weaknesses were note, these were apparently discovered prior to the NRC's recent inspection. A comprehensive program to review 100% of the previous IEB 79-14 work is currently underway and will address all of the weaknesses that have been disclosed so far. Based on the current inspection efforts as well as those previous efforts documented in the Background section of this report, this item is considered closed.

4. Open Items

Open items are matters which have been discussed with the licensee, will be reviewed further by the NRC inspector, and which involves some action on the part of the NRC or licensee or both. Open items disclosed during this inspection are discussed in Paragraph 3.a.2.c.

5. Exit Interview

The Region III inspector telephoned the licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on March 8, 1988. The inspector summarized the purpose and findings of the inspection. The licensee representatives acknowledged this information. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed during the inspection. The licensee representatives did not identify any such documents/processes as proprietary.