U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No.: 50-282/88017(DRS); 50-306/88017(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, MN

Inspection Conducted: September 21-22, 1988

Inspector: J. A. Gavula — J. A. Gavula — 10/7/88 Date

Approved By: D. H. Danielson — D. H. Danielson, Chief
Materials and Processes Section — 10/11/88 Date

Inspection Summary

Inspection on September 21-22,1988 (Reports No. 50-282/88017(DRS); 50-306/88017(DRS))

Areas Inspected: Special safety inspection of facility modifications which resulted from the ongoing IEB 79-14 reviews (37701); and licensee action on previously identified concerns (92701).

Results: One violation was identified (inadequate bases for safety evaluation Paragraph 4.a).

- Ongoing reviews of IEB 79-14 records have identified discrepancies which exceeded allowable stress values specified in the Safety Analysis Report.
- Modifications have been made to 20 supports to correct deficiencies resulting from Copes Vulcan center of gravity discrepancies and other IEB 79-14 related problems.
- Review process is approximately one-third completed.

DETAILS

1. Persons Contacted

Northern States Power Company (NSP)

*+J. Goldsmith, Superintendent, Nuclear Technical Services
+G. Eckholt, Licensing Engineer
*+G. Rolfson, Engineer
*+J. Donatell, Engineer
*+C. Baltos, Engineering Associate

Fluor Daniel, Inc. (FDI)

*+B. Dickerson, Project Piping Engineer
+J. Khanna, Engineer

Teledyne Engineering Services

+M. McKeown, Project Engineer

Nuclear Regulatory Commission (NRC)

+J. Hard, Senior Resident Inspector
*M. Moser, Resident Inspector
+D. DiIanni, Licensing Project Manager, NRR
+S. Hou, Senior Mechanical Engineer, NRR
+B. Burgess, Chief, Projects Section 2A, RIII

+Attended meeting at Prairie Island on September 21, 1988.
*Attended exit interview on September 22, 1988.

2. Licensee Action on Previously Identified Items

a. (Closed) Open Item (282/87018-01; 306/87017-01): Comprehensive review program of all previous IEB 79-14 work was implemented by the licensee to identify and correct inconsistencies in applying and documenting reconciliation criteria.

Based on the potential significance of the discrepancies being found and the number of re-analyses required to resolve the identified discrepancies, the acceptability of the original IEB 79-14 effort requires further evaluation. On this basis, the above item is closed and an Unresolved Item will be opened. (See Paragraph 4 of this report for details on this item (282/88017-02; 306/88017-02)).

b. (Closed) Open Item (282/88009-01; 306/88009-01): Criteria and methodology for establishing a justification for continued operation requires additional evaluation.

Based on the meeting discussed in Paragraph 3 of this report, the final criteria submitted by NSP was determined to be acceptable. This item is considered closed.

3. Meeting to Discuss Criteria for Justification for Continued Operation

A meeting was held between representatives of the licensee and the NRC (denoted in Paragraph 1) on September 21, 1988. The purpose of the meeting was to discuss details of NSP's recently submitted "Criteria for Determining Justification for Continued Operation when Encountering "As-Built" Discrepancies Which Cause Large Increases in Seismic Stress". A draft copy was issued on June 8, 1988, and a completely revised document was submitted for review during the September 21, 1988 meeting (See Enclosure 3). The major points of discussion during the meeting pertained mainly to administrative aspects of the criteria. Slight changes to some of the wording in criteria were made during the meeting to resolve these concerns. From a technical perspective, the criteria was determined to be acceptable for piping and pipe support operability stress limits. Based on the conclusions reached during the meeting, no additional evaluations of the criteria will be required. This issue is considered closed.

4. Facility Modifications Resulting from IEB 79-14 Reviews

a. Background

As documented in NRC Inspection Reports No. 50-282/87018; 50-306/87017, a review program is currently underway at FDI to reassess the original IEB 79-14 work. This effort resulted from FDI's recognition that some inconsistencies potentially existed in applying and documenting the original reconciliation criteria at Prairie Island. During the implementation of this program, the discrepancies on the Copes Vulcan valve weights and centers of gravity were discovered (See NRC Inspection Reports No. 50-282/88009; 50-306/88009 for additional details). As a result of these discrepancies, several piping systems were found to exceed code stress allowables and in one instance, the stresses were calculated to be approximately 160 ksi for the design basis earthquake (DBE) load. This was almost six times the allowable stress limit.

As a result of this situation, Safety Evaluation Report No. 259-1, Revision 1, "Justification for Continued Operation Due to Increased Valve Weights of Small Bore Copes Vulcan Control Valves" was issued on May 6, 1988. This evaluation provided the bases and criteria used to justify continued operation until the analyses could be completed and any resulting required modifications implemented.

During the review of this report, the NRC inspector noted the following problems:

- The original piping system was designed and constructed in accordance with ANSI B31.1, Power Piping, 1967, but allowable stress limits for ASME Section III, Class I piping were used. Although the report states that Class I piping requires a fatigue evaluation and also takes into account the plastic behavior of the piping material, no justification was given for differences between ASME Class I material and construction requirements and ANSI B31.1 material requirements.

- The report states that, "it is assumed for purposes of this evaluation that the seismic stresses calculated using non-linear techniques is less than 3 times the allowable stress intensity". No quantitative justification is provided for this assumption.

- A Class I fatigue evaluation and associated stress criteria was used to determine the operability of the piping system. No bases were given as to why primary plus secondary stress intensity ranges would be applicable to a strictly primary stress situation. The limitations for Level D service limits for faulted load conditions were exceeded by over a factor of three.

- Because of the concerns stated above, it cannot be concluded that the accumulator pressure boundary would survive a DBE. Furthermore, the bases for stating that the charging pumps for the safety injection system would keep up with a leak, if a portion of the piping system in question ruptured, was not provided in the evaluation.

As a result of the above, it was determined the bases for the above safety evaluation were not adequately established. This is an example of a violation of 10 CFR 50.59 (282/88017-01; 306/88017-01).

b. <u>Modification Implementation</u>

Even though NSP had completed the above justification for continued operation, immediate steps were taken to modify the pipe supports to bring all stresses within the code allowables. In addition, all of the other discrepancies which were identified prior to the Unit 1 fall outage were corrected before the unit was restarted.

For Unit 1, the following modifications have recently been made:

- Stress Report No. PI-216-II, Line 9.
  Support No. 1-RHRRH-21: Reinforced existing support.
  Support No. 1-RHRRH-24: Reinforced existing support.

- Stress Report No. PI-233-28, Line 39.
  Support No. 1-CCRH-41: Added U-bolt to existing support.

- Stress Report No. PI-206-I & III, Line 6.
  New anchor installed (never installed originally).

- Stress Report No. PI-206-43, Lines 38B & 38C.
  Support No. 1-RCVCH-857: Reinforced existing support.
  New support added to valve CV-31334.
  Pipe Whip No. 14CVCS-2: Modified into a 2-way guide.

- Stress Report No. PI-234-XIV, Lines 113/114.
  New support added to valve CV-31329.
  Support No. RCRH-14: Rod replaced to accommodate uplift.

Support No. RCRH-18: Rod replaced to accommodate uplift.
Support No. RCRH-19: Kicker added to existing support.
Support No. RPCH-2: Vertical modified into 2-way guide.
Support No. RCVCH-1451: Increase weld size.

- Stress Report No. PI-205-22, Line 14.
  New support added to valve CV31445.
  Support No. RSIH-32: U-bolt stiffened.

- Stress Report No. PI-205-VI, Line 10J.
  New support added to valve CV-31442.

- Stress Report No. PI-205-14, Line 1DB.
  New support added to valve CV-31447.
  New support added to valve CV-31448.
  Support No. 1-RSIH-423: Modified existing 1-way into 2-way.
  Support No. 1-RSIH-424: Modified existing 1-way into 2-way.

Drawings for the above modifications were briefly reviewed by the NRC inspector. Additional inspections of the design analyses associated with these modifications are planned in the future. The majority of the above modifications were walked down by the NRC inspector while at the site. No adverse comments were made relative to the general installation configuration or workmanship.

The discrepancies associated with the above modifications were caused by several factors. As previously mentioned, changes in the weights and centers of gravity for certain Copes Vulcan valves required re-analysis and subsequent modifications to be made. This was the case in most of the above modifications; however, several modifications were required which were not associated with the Copes Vulcan issue. In one case, an anchor was not installed as originally designed. Although the IEB 79-14 field walkdown indicated it was not installed, the review of the analyses did not identify this deficiency. In another instance, the wrong response spectra was used in the seismic analysis.

c. Scope of Future Modifications

The current review program has identified 91 subsystems which will require re-analysis to reconcile the as-built configuration to the design analysis. To date, approximately one-third of this re-analysis is completed. Since the worst cases are being addressed first, it is assumed that very few additional modifications will be required. There are currently four subsystems in Unit 2 which will require some degree of modifications but other than those, no additional modifications are anticipated.

Based on the current scope of the required re-analysis and the modifications required to date, the acceptability of the original IEB 79-14 effort will require additional evaluation by the NRC.

Pending the determination of the extent and significance of the as-built discrepancies, this will be considered an Unresolved Item (282/88017-02; 306/88017-02).

5. Unresolved Items

An unresolved item is a matter about which more information is required in order to ascertain whether it is an acceptable item, an open item, a deviation, or a violation. Unresolved items disclosed during this inspection are discussed in Paragraph 4.c.

6. Exit Interview

The Region III inspector met with the licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on September 22, 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.

ENCLOSURE 3

NSP

Northern States Power Company

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Minneapolis, Minnesota 55401
Telephone (612) 330-5500

September 26, 1988

Director of Nuclear Reactor Regulation
U S Nuclear Regul tory 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

Generic Criteria For Justification Of Continued Operation

On September 21, 1988, the NRC Staff met with NSP representatives at Prairie Island to discuss generic criteria to be utilized for the justification of continued operation when major discrepancies in "as-built" safety related piping are encountered. Attached is a copy of the generic criteria agreed to during the meeting.

Please contact us if you have any questions regarding the information provided.

David Musolf
Manager - Nuclear Support Services

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
G Charnoff

Attachment:
Criteria For Determining Justification For Continued Operation When Encountering Major Discrepancies in "As-Built" Safety Related Piping, September 21, 1988

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SEP 30 1988

CRITERIA FOR DETERMINING JUSTIFICATION

FOR CONTINUED OPERATION WHEN ENCOUNTERING

MAJOR DISCREPANCIES IN "AS-BUILT" SAFETY RELATED PIPING

SEPTEMBER 21, 1988

NORTHERN STATES POWER CO.

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

1717 WAKONADE DRIVE EAST

WELCH, MN 55089

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1.0 INTRODUCTION & SCOPE

These criteria are intended to assure the operability requirements of safety related piping and associated supports if it is determined that stresses exceed allowables presented in the Prairie Island USAR. These criteria permit operation for an interim period only. Modifications will be made which return the system to within USAR allowables by the next refueling outage or sooner if operation permits.

These criteria are intended to expeditiously perform necessary evaluations to determine interim operability and not to delay appropriate actions.

For cases involving components classified as ASME Code Class I where USAR allowables are exceeded, NSP shall be notified upon discovery and NSP shall evaluate reportability requirements per 10CFR50.

2.0 CRITERIA

2.1 Piping Operability Criteria

The piping analysis shall be in accordance with ASME, Section III NC-3600 service level D limits (Ref. 1). The design loading conditions to be applied in the analysis shall include the DBE earthquake.

Following is the pipe stress criteria for justifying continued operation of the plant:

$[S_{LP} + S_{WT} + S_{DBE} \leq 2.0\ Sy]$ (Ref. 1 equation 9)

Where: S_{LP} = Longitudinal Pressure Stress

S_{WT} = Dead Weight Stress

S_{DBE} = Stresses Resulting From Design Basis Earthquake

S_Y = Material Yield Stress (Reference 1 Appendices)

Code Case N-411 allows for increased damping values, independent of pipe diameter, for seismic analysis. Therefore, increased damping values, in accordance with reference 2, will be acceptable when performing these analyses to meet operability. Should the piping stress analysis exceed the value of 2.0 Sy, or pipe supports do not meet their operable limits (see Sect. 2.2), then additional iterative analysis of the piping may be required. The iterative analysis may use the knowledge that a support is not capable of withstanding the loads, and can be removed from the analysis. Where feasible, the actual support stiffness may be included in the iterative analysis, along with other refinements.

For cases where piping secondary stresses are determined to exceed USAR allowables, a specific case by case approach will be used to determine interim operability.

2.2 Pipe Support & Hanger Operability Criteria

As a first step in evaluating the support, a linear elastic analysis method will be used to determine the stress in the support members. In addition to the loading in Section 2.1, the support loads must include pipe thermal loads and results from free end displacement and anchor motion. Supports will be analyzed using the allowables listed below to meet operability requirements.

Structural Steel

Tension	F_t = 1.20 Sy but $\leq$ 0.70 Su
Bending	F_b = 1.20 Sy but $\leq$ 0.70 Su
Shear	F_v = 0.72 Sy but $\leq$ 0.42 Su
Compression	F_a < F_t but not to exceed 2/3 Pcr
Combined Stress	For axial compression and bending <u>or</u> axial tension and bending, use AISC 1.6., (Ref. 6)
Web Crippling	= 1.0 Sy

Weld Stress	F_w = 0.42 Su (of weld material)
Anchor Bolts	Use Factor of Safety of 2 against ultimate tension and shear values.
Snubbers	
Hydraulic:	Load < manufacturers one time load capacity. Movement < total travel
Springs	Load within catalog range without bottoming out
Struts	FS = 2 and < 2/3 Pcr

All remaining Catalog Items	Use manufacturers published faulted load rating. Where level D allowables are not given, and the factor of safety is specified in the catalog, use design allowables but with FS = 2. (Typical catalog FS = 5, therefore use 2.5 x catalog capacity).

Where: F_t = Allowable Tensile Stress

F_b = Allowable Bending Stress

F_v = Allowable Shear Stress

F_a = Allowable Axial Compressive Stress

F_w = Allowable Weld Stress

Pcr = Maximum Strength of Axially Loaded Compression Member

Sy = Specified Minimum Field Strength at Temperature (See Note 1)

Su = Specified Minimum Tensile Strength Temperature

FS = Factor of Safety

NOTE 1: Actual yield strength may be used where CMTR's are available for the material.

If a support fails using the linear elastic method, then a more refined analysis may be performed using plastic analysis techniques. The plastic analysis will follow the design rules of ASME Section III, Appendix F, (Ref. 1).

3.0 CONCLUSION

If the above criteria cannot be met, reportability per 10 CFR 50 must be evaluated and system operability requirements per Plant Technical Specifications must be evaluated and appropriate actions taken.

4.0 REFERENCES

1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Codes, Section III, 1983 Edition, through Winter 1985 Addenda.

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Codes, Case N-411, Dated 9/17/84.

3. NRC-IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Revision No. 1 (Supplement No. 1), Dated 8/20/79.

4. USAS B31.1.0-1967, Power Piping Code.

5. Updated Safety Evaluation Report for PINGP.

6. "Manual of Steel Construction," American Institute of Steel Construction, Inc., Eighth Edition, 1980.