Northern States Power Company

Monticello Nuclear Generating Plant
2807 West Hwy 75
Monticello, Minnesota 55362-9637

July 3, 1996

10 CFR Part 2
Section 2.201

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Reply to Notice of Violation Contained in
NRC Inspection Report No. 50-263/96003

Pursuant to the provisions of 10 CFR Part 2, Section 2.201, our reply to the notice of violation contained in your letter of June 3, 1996, is provided in Attachment A.

Attachment A contains the following new NRC commitments:

Procedure 9019 will be further enhanced to provide appropriate emphasis on coordination with refuel floor activities. Specifically, the individual rod checklist will be revised to include a signoff referencing the procedural step governing refuel floor coordination. The Procedure will be revised by September 30, 1996.

This temporary change oversight will be discussed in Operator Continuing Training with emphasis on a "questioning attitude" with regard to the follow-up associated with operational decisions. The training will be completed by September 30, 1996.

By August 1, 1996, the electrical maintenance/construction pre-job briefing checklist will be revised to include an item that covers construction notes and/or other special installation requirements.

Your letter of June 3, 1996 also requested a discussion of our bases for assuming leak before break will provide sufficient notice to ensure future leaks would be detected prior to HELB pipe failure. The requested discussion is provided in Attachment B.

Please contact Mel Opstad at (612) 295-1653 if you require further information.

William J Hill

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Plant Manager
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c: Regional Administrator - III, NRC
NRR Project Manager, NRC
Sr Resident Inspector, NRC
State of Minnesota
Attn: Kris Sanda
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Attachment A: Reply to Notice of Violation
Attachment B: Use of "Leak Before Break" in Operability Determinations

Attachment A

REPLY TO NOTICE OF VIOLATION

Violation Identified in 6/3/96 NRC Letter

10CFR50 Appendix B, Criterion V stated that activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings.

1. *Administrative work instruction, 4-AWI-02.02.05, "Temporary Change Process," revision 3, section 4.1.2 required temporary changes to procedures to be reviewed and approved prior to implementation.*

   Procedure 9019 "Changeout Selected Control Rod Drives - Operations," step 4 required the operator to notify personnel on the refuel floor to halt any activity over the reactor cavity.

   Contrary to the above, between April 16 and 18, 1996, operators did not notify personnel on the refuel floor prior to manipulating control rods. Operations management failed to temporarily change procedure 9019 to not require this notification prior to commencing control rod changeouts.

2. *Administrative work instruction, 4-AWI-05.01.4, "General," revision 2, section 4.16 required items such as field installation interference problems be resolved using engineering change requests.*

   Procedure QCIM-EL-01 (Rev. 3), "Inspection Requirements and Acceptance Criteria for Electrical Activities," Section E.4.3 required quality control (QC) inspectors to verify that the configuration of work was as specified on drawings/procedures.

   Contrary to the above, a mechanic did not install two speed sensing panels for the #12 emergency diesel generator on a safety related panel C-92 as specified in the modification drawings. Also, a QC inspector did not identify that the work was not in accordance with the drawings.

This is a Severity Level IV violation (Supplement 1).

NSP Response to Item 1

NSP acknowledges item 1 of the above violation.

Reason For Item 1 of Violation:

The cause of the failure to implement a temporary change to Procedure 9019, " Changeout Selected Control Rod Drives - Operations" was due to cognitive errors. On April 16, 1996 a Shift Manager, who was augmenting the duty shift, approved the commencement of Procedure 9019 and provided first line supervision of the control rod drive changeout activity from an operational perspective. It was correctly established that Step 4, which required the operator to notify personnel on the refuel floor to halt any activity over the reactor cavity, could be waived. The reactor operator performing the procedure was so advised, however, the Shift Manager made a cognitive error in not recognizing a temporary change was required prior to implementing the waiver. On April 18, 1996 the Shift Supervisor, who in turn reaffirmed the applicability of the Step 4 waiver, also made a cognitive error in not requiring a temporary change. This second error was due in part to the practice of the Step 4 waiver that had occurred the previous two days. Both of these cognitive errors were facilitated by the fact that Step 4 signoff was not required in either the main body of the procedure or on the individual control rod checklists.

Corrective Action Taken and Results Achieved:

1. A temporary change to Procedure 9019, "Changeout Selected Control Rod Drives - Operations " regarding Step 4 requirements was approved on April 18, 1996.

2. A permanent change to Procedure 9019 was approved on May 13, 1996 as Revision 13. Revision 13 clarified the conditions under which refuel floor personnel are to be notified and the actions to take prior to control rod withdrawal.

3. The event investigator interviewed involved personnel and determined that this was an isolated incident of failure to implement a temporary change and no programmatic deficiencies were identified. The investigator noted that the Shift Manager had properly documented a temporary change to Procedure 9019 on the date this oversight occurred.

Corrective Action to be Taken to Avoid Further Violations:

1. Procedure 9019 will be further enhanced to provide appropriate emphasis on coordination with refuel floor activities. Specifically, the individual rod checklist will be revised to include a signoff referencing the procedural step governing refuel floor coordination. The Procedure will be revised by September 30, 1996.

2. This temporary change oversight will be discussed in Operator Continuing Training with emphasis on a "questioning attitude" with regard to the follow-up associated with operational decisions. The training will be completed by September 30, 1996.

NSP Resp[illegible]e to Item 2

NSP acknowledges item 2 of the above violation.

Reason For Item 2 of Violation:

A pre-job briefing checklist was implemented for the 1996 refueling outage to enhance pre-job briefings for the electrical maintenance/construction personnel. An item not included on this checklist was special installation requirements such as those given in construction notes on drawings. The cause of this violation was failure of the installation crew and the QC inspector to recognize the requirement specified in a construction note (on the modification drawings) to use existing mounting holes to install the speed sensing panels. As a result of not recognizing this requirement, the persons involved did not initiate an Engineering Change Notice (ECR) as required to resolve a discrepancy with the mounting holes. An effort was made to determine if the construction note requirements were specifically discussed in the pre-job briefing, but this could not be determined.

Corrective Action Taken and Results Achieved:

Installation Aspects

1. The NSP Responsible Supervisor and the Project Engineer interviewed the crew foreman and the electrician involved with the incident. It was determined that the crew foreman and the electrician were not aware of the construction note requiring that the speed sensing panels be installed using the existing mounting holes. They knew, however, that an ECR must be initiated when an installation cannot be performed in accordance with the drawing requirements.

2. The NSP Responsible Supervisor provided training on this incident for the electrical construction craft personnel involved with outage related activities. The training stressed the importance of following installation requirements and the necessity for attention to detail.

3. ECR No. 93Q415-05 was written to assess and document the acceptability of drilling and tapping mounting holes in panel C-92 for mounting the speed sensing panels.

4. A preoperational test was performed as part of the design change to assure that the portion of the Emergency Diesel Generator (EDG) system affected by the design change functioned properly. In addition, applicable plant surveillance procedures were performed to ensure the EDG system functioned properly.

QC Inspection Aspects

1. The Quality Supervisor discussed the incident with QC Inspector to determine if there was a lack of understanding of the acceptance criteria. It was determined that the inspector knew the criteria and specifically the requirement for an ECR to resolve field installation interference problems.

2. In order to reinforce the importance of the QC function and the necessity for attention to detail, the Quality Supervisor discussed the details of this incident with each of the QC inspectors. The Supervisor specifically emphasized the Quality Control Inspection Manual acceptance criteria for verifying proper installation in accordance with drawings.

Corrective Action to be Taken to Avoid Further Violations:

By August 1, 1996, the electrical maintenance/construction pre-job briefing checklist will be revised to include an item that covers construction notes and/or other special installation requirements.

Date When Full Compliance Will Be Achieved

Full compliance has been achieved.

Attachment B

USE OF "LEAK BEFORE BREAK" IN OPERABILITY DETERMINATIONS

As discussed in NRC Integrated Inspection Report 50-263/96003, plant engineering personnel discovered an error in the Monticello high energy line break (HELB) thermal-hydraulic model used for the Turbine Building. The error was discovered shortly before the scheduled beginning of the 1996 refueling outage. The error was found as a consequence of pro-active upgrading of computer modeling techniques used to analyze HELB effects.

A Justification for Continued Operation (JCO) safety evaluation was prepared by engineering personnel and reviewed by the plant Operations Committee. This evaluation documented the basis for continued plant operation until full compliance with the Monticello HELB design and license basis could be achieved. The evaluation relied upon the "leak before break" (LBB) principle. The evaluation was later augmented with additional engineering analyses showing that serious operability concerns did not exist for equipment required to be operable in the event of a HELB.

Modifications made during the 1996 refueling outage restored the plant to full compliance with the HELB design and licensing basis described in the Monticello Updated Safety Analysis Report (USAR).

Integrated Inspection Report 50-263/96003 requested that Northern States Power Company provide a discussion of the basis for assuming leak before break will provide sufficient notice to ensure future leaks would be detected prior to HELB pipe failure.

Response:

Leak-Before-Break Analysis

The LBB analysis used to support continued plant operation was performed for the Monticello plant by Impell Corporation in April, 1986, when an earlier nonconformance with the Monticello HELB design and licensing basis was discovered. Since the same piping was involved in both the 1986 event and this event, the 1986 LBB analysis was applicable.

A copy of this analysis was provided to the Commission as an attachment to our letter dated April 9, 1986 (Reference 1).

The Impell analysis demonstrated that a large double-ended break of the affected feedwater piping was extremely unlikely. The more likely scenario would be the initiation of a small stable crack with leakage of less than 0.5 gpm. This leakage would have a negligible impact on equipment operability and would be detected during walk-throughs of the affected area. Plant procedures would direct operators to expeditiously shutdown the plant and isolate the affected piping when leakage was detected.

Refer to Reference (1) for details of this analysis.

Basis for Leak-Before-Break Technology

The basis for use of LBB technology in JCO safety evaluations is established by:

A. Regulatory Requirements and Guidance

B. Limited Application of LBB Technology

C. Prior Acceptance of LBB

A. Regulatory Requirements and Guidance

The current regulatory requirements and guidance applicable to LBB technology are contained in the following documents:

- Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability"

- Generic Letter 88-07, "Modified Enforcement Policy relating to 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

- "Modification of General Design Criterion (GDC) 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," Federal Register Notice. Volume 52, Number 207, dated October 27, 1987.

- "Policy Statement on Additional Application of Leak-Before-Break Technology," Federal Register Notice, Volume 54, Number 83, dated May 2, 1989.

Generic Letter 91-18 contains guidance for evaluating equipment operability and performing JCO evaluations. The Monticello Administrative Work Instruction which governs preparation of JCO evaluations references this NRC Generic Letter. Two NRC Inspection Manual sections are attached to the Generic Letter. Key provisions of these NRC Inspection Manual sections applicable to evaluation of environmental qualification (EQ) nonconforming conditions include:

1. Items which are appropriate for consideration in a licensee's development of a JCO include:

- Availability of redundant or backup equipment
- Compensatory measures including limited administrative controls
- Safety function and events protected against
- Conservatism and margins, and
- Probability of needing the safety function.
- Probabilistic Risk Assessment (PRA) or Individual Plant Evaluation (IPE) results that determine how operating the facility in the manner proposed in the JCO will impact the core damage frequency

2. The full scope of the current licensing basis, including the TS and FSAR commitments, should be examined to establish the conditions and performance requirements to be met for determining operability. The operability decision may be based on analysis, a test or partial test, experience with operating events, engineering judgements, or a combination of these factors taking into consideration equipment functional requirements. An initial determination regarding operability should be revised, as appropriate, as new or additional information becomes available.

3. The scope of an operability determination needs to be sufficient to address the capability of the equipment to perform its safety function(s). Operability determinations should therefore include the following actions:

   - Determine safest plant configuration including the effect of transitional actions.

   - Determine the basis for declaring the affected system operable, through:

     a. analysis
     b. test or partial test,
     c. operating experience, and
     d. engineering judgement

The JCO evaluation appropriate to EQ issues is specifically addressed by Generic Letter 88-07. Generic Letter 88-07, contains the following guidance:

1. When a potential deficiency has been identified by the NRC or licensee in the environmental qualification of equipment (i.e., a licensee does not have an adequate basis to establish qualification), the licensee is expected to make a prompt determination of operability (i.e., the system or component is capable of performing its intended design function),

take immediate steps to establish a plan with a reasonable schedule to correct the deficiency, and have written justification for continued operation, which will be available for NRC review.

2. The licensee may be able to make a finding of operability using analysis and partial test data to provide reasonable assurance that the equipment will perform its safety function when called upon. In this connection, it must also be shown that subsequent failure of the equipment, if likely under accident conditions, will not result in significant degradation of any safety function or provide misleading information to the operator.

GDC 4 requires systems, structures, and components to be designed to accommodate the effects of a variety of events including accidents. Historically, GDC 4 has been implemented by assuming a non mechanistic double-ended guillotine rupture of a variety of pipes.

Based on research conducted by the industry and the NRC in the 1980's, it was shown that piping used in nuclear applications was not likely to crack in ways that result in instantaneous guillotine ruptures. GDC 4 was revised in 1987 to permit the exclusion of dynamic effects associated with a broad range of postulated pipe ruptures from the design basis when it can be demonstrated to the Commission that the probability of fluid system piping rupture is extremely low. The technical basis for the revision to GDC 4 is summarized in Reference (2). The NRC acknowledged, at that time, that an inconsistency was created between the design basis for dynamic effects and the design basis used for emergency core cooling system (ECCS) and containment design and EQ. It was stated that:

> When leak-before-break technology is applied to dynamic effects design bases, these effects are reduced to zero; there are no replacement dynamic effects postulated. However, environmental qualification design bases cannot be reduced to zero when leak-before-break technology is applied to piping. The postulated pipe rupture has served as a convenient and conservative umbrella covering many sources of environmental qualification design bases, such as breaches in the fluid system pressure boundary from failed pump seals, leaking valve packing, flanged connections, bellow, manways, rupture disks and through wall cracks. Thus, in applying leak-before-break technology to environmental qualification, the Commission faces the task of developing a replacement environmental qualification design basis.

In its 1989 Policy Statement on Additional Applications of Leak-Before-Break Technology, the Commission decided not to extend the applicability of LBB to ECCS or EQ. The Commission did, however, remain open to future rulemaking which would permit the application of LBB in these areas.

Therefore a review of available regulatory guidance indicates that use of LBB analysis in JCO evaluations is not prohibited. Use of LBB to demonstrate compliance with EQ design and licensing basis requirements is a different matter, however, and is clearly not permitted unless an exemption is granted by the Commission.

Available NRC guidance also indicates that a complete JCO includes a discussion of alternate or backup safe shutdown paths, compensatory measures, safety margins, and probabilities. These issues should be evaluated in an EQ JCO, in addition to LBB considerations, for completeness. NRC guidance also makes it clear that engineering judgement has a place in operability determinations and JCO evaluations.

B. Limited Application of LBB Technology

LBB analysis is a mechanical engineering technology that can often be used to provide reasonable assurance that equipment will perform its safety function following the discovery of a condition that is outside the established EQ design and licensing basis. The JCO evaluation documents the basis for continued plant operation, for a limited period of time, until corrective actions are complete and full compliance with EQ design and licensing basis is once again achieved.

The utilization of LBB technology in a JCO evaluation is not intended to demonstrate compliance with EQ design or licensing basis requirements.

In this case leak-before-break technology was utilized in accordance with long established Monticello administrative controls for the prompt determination of operability required by the Monticello Technical Specifications. It was not used to demonstrate compliance with 10 CFR 50.49.

C. Prior Acceptance of LBB

LBB technology has been utilized at Monticello and other nuclear generating plants in JCOs prepared to address both dynamic and environmental effects of HELBs.

Specifically, in 1986, when unanalyzed potential pipe break locations were identified in the Turbine Building pipe chase, a JCO evaluation was prepared which relied almost entirely on LBB technology to permit short-term operation of the plant until the next refueling outage (less than 30 days). During the outage modifications were made to achieve full compliance with NRC requirements. The JCO was documented in Monticello Safety Review Item 86-014. Discussions were held with the NRC Staff and a followup notification was provide by letter dated June 9, 1986 (Reference 1).

The procedure used in addressing the recent HELB analysis non-conformance parallelled the evaluation and corrective action developed in response to the 1986 event.

Conclusions

The initial JCO evaluation prepared in this case lacked the detail and rigor that is customary at Monticello. The initial operability determination and documented JCO was developed within an hour of concluding that a HELB design and licensing basis nonconformance existed. The JCO was later supplemented with additional engineering evaluation including information from a 14-hour thermal-hydraulic computer analysis completed the next day.

The long standing Monticello administrative process related to HELB practices was followed.

We recognize that LBB has not been accepted by the NRC as a permanent solution to environmental qualification concerns. We also believe, however, that past NRC guidance and precedent permit LBB to be used as an essential element of a JCO evaluation to document the basis for continued plant operation on a limited bases following the discovery of plant HELB design and licensing basis deviations.

References:

1. Letter dated April 9, 1986, from David Musolf, NSP, to Director, Office of Nuclear Reactor Regulation, NRC, "Failure to Provide High Energy Line Break Protection for the Turbine Building Pipe Chase"

2. NUREG-1061, "Report of the US Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," Vols 1 - 5