NSP

Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401-1927
Telephone (612) 330-5500

March 15, 1990

NRC Generic Letter 89-19

Director of Nuclear Reactor Regulation
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

Response to NRC Generic Letter 89-19
Unresolved Safety Issue A-47 "Safety Implication
of Control Systems in LWR Nuclear Power Plants"

Our response to Generic Letter 89-19 is attached.

Please contact us if you have any questions related to our response.

Thomas M Parker

Thomas M Parker
Manager
Nuclear Support Services

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

Attachments
1. Affidavit
2. Response to NRC Generic Letter 89-19

A019
A001 '1

Attachment 1

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT   DOCKET NO. 50-282
50-306

UNRESOLVED SAFETY ISSUE A-47 "SAFETY IMPLICATION
OF CONTROL SYSTEMS IN LWR NUCLEAR POWER PLANTS"

Northern States Power Company, a Minnesota corporation, with this letter is submitting information requested by NRC Generic Letter 89-19.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Thomas Parker

Thomas M Parker
Manager, Nuclear Support Services

On this 15th day of March 1990 before me a notary public in and for said County, personally appeared Thomas M Parker, Manager, Nuclear Support Services, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Judy L Klapperick

JUDY L. KLAPPERICK
NOTARY PUBLIC—MINNESOTA
ANOKA COUNTY
My Commission Expires Sept. 29, 1991

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

RESPONSE TO NRC GENERIC LETTER 89-19

The original design for steam generator overfill protection at Prairie Island Units 1 and 2 is defined by Group I on page 4 of Enclosure 2 to Generic Letter 89-19. This is one of the "Plants that have an overfill-protection system initiated on a steam generator high-water-level signal based on . . . a 2-out-of-3 initiating logic which is safety grade but uses one out of the three channels for both control and protection." This system isolates feedwater to the steam generators by tripping the main feedwater pumps and closing the feedwater main and bypass control valves.

The Generic Letter concludes this design is acceptable if:

A. The feedwater control system is not powered from the same source as overfill protection.

B. Overfill protection and feedwater control are not located within the same cabinets.

C. Overfill protection and feedwater control signals are routed such that a fire is not likely to affect both systems.

D. Plant procedures and Technical Specifications include requirements to periodically verify operability of overfill protection.

Each criterion is individually addressed below in the context of the original Prairie Island design; then the added benefit of the upgraded feedwater control system, with a Median Signal Select, is discussed.

A. System Power: Overfill protection is provided through trip bistables in the Reactor Protection analog instrumentation racks, which are powered from Red (Panel 112), White (Panel 111), Blue (Panel 113), and Yellow (Panel 114) 120 VAC instrument buses. Upon bistable actuation, Train A and Train B 120 VAC Protection relays (normally powered by the bistable) are deenergized, and the relay contacts (configured in a 2 out-of 3 matrix) close to energize 125 VDC safeguards relays, resulting in overfill protection actuation.

   In general, the control systems are powered from the same 120 VAC panels; however, different source breakers are used to provide separation from the protection supply. An instrument bus failure would cause the 120 VAC relay to deenergize, actuating one channel of overfill protection. This aspect of the licensed design of the plant has been discussed with Westinghouse personnel and it is believed the design is adequate regarding overfill protection.

B. Location: Overfill protection and feedwater control are physically located in separate cabinets.

C. Routing: No control signal cables are routed with protection cables (Red, White, Blue, Yellow, Train A, or Train B). This is the case for feedwater control/overfill protection; thus the likelihood of a fire affecting both systems is minimized.

D. Technical Specifications/Surveillance Procedures: Steam Generator level channel testing is required to be performed at a frequency specified by Technical Specifications (Table 4.1-1, Item 11). The only aspect of the channels required to be tested per Technical Specifications (Table 3.5-2, Item 12) is the Low-Low Level actuation functions. Thus, testing of the high level trip actuation (feedwater pump trip and valve closure) is not specifically addressed by Technical Specifications.

However, this function is presently tested by Prairie Island procedures. Individual protection channels are tested via monthly analog protection testing (SP1003). The 2-out-of-3 safeguards relay matrix is tested monthly via safeguards logic testing (SP1032A). Verification of relay energization is tested via the Integrated SI Test (SP1083) every outage. Field actuation is tested via feedwater isolation testing (SP1143) every outage.

Feedwater Control System Upgrade

Prairie Island has recently upgraded its feedwater control system on Unit 2 (4/89) and Unit 1 (1/90). The control system now uses all three narrow range steam generator level channels for protection and control, with the implementation of a Median Signal Select on steam generator level control channels to provide signal validation. This prevents protection/control interaction upon failure of a single level channel and is used to satisfy IEEE 27, Paragraph 4.7.3 criteria for reactor protection (low level trip, Reference: Westinghouse WCAP-11931). In addition, the NRC has accepted the use of a median signal select, implemented in control grade equipment, to address steam generator overfill concerns at another Westinghouse plant (Reference: Westinghouse WCAP-11484).

Although there are no licensing requirements to test the median signal select, it is implicitly tested through the proper operation of automatic steam generator level control while monthly surveillance of protection channels is occurring. Additionally, the median signal select function for narrow range steam generator level is specifically verified (and documented) by ensuring that a failed channel (high or low) will not be selected for control.

The reliability of the new control system is supported by redundant 120 VAC power supplies (from the aforementioned instrument buses). The failure of a single 120 VAC bus will have no effect on automatic feedwater control.

Thus, the new control system characteristics enhance the licensing basis of the plant with regard to steam generator overfill protection. This design has been accepted by the NRC via an approved License Amendment for Prairie Island; however, it should be noted that GL 89-19 does not recognize this design.

The following actions will be taken:

A. A License Amendment Request will be submitted to:

Revise Table TS 3.5-4 to include Feedwater Isolation, as shown below:

FEEDWATER ISOLATION

a. Hi-Hi Steam Generator Level
b. Safety Injection
c. Reactor Trip with Low T_{ave} (Main Valves only)

B. Feedwater isolation surveillance procedures will be revised to verify that control valves move from full open to full closed in 5 seconds or less, upon deenergizing either air supply solenoid valve. Each of these procedures will be revised prior to the next refueling outage for which the surveillance is performed.