UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 3, 1989

Dockets Nos. 50-282
and 50-306

Mr. D. M. Musolf, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

SUBJECT: AMENDMENTS NOS. 87 AND 80 TO FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60: ELIMINATION OF STEAM/FEEDWATER MISMATCH FLOW AND LOW FEEDWATER REACTOR TRIPS (TACS NOS. 69175 AND 69174)

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. DPR-42 and Amendment No. 80 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 18, 1988 as supplemented by letters dated September 15, 1988 and March 10, 1989.

The amendments change the TSs by eliminating the steam/feedwater mismatch flow in coincident with the low feedwater flow reactor trip.

A copy of the Safety Evaluation supporting these amendments is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Additionally, in support of the amendments applications, you submitted Westinghouse Corporation Topical Report WCAP-11931, describing the advanced digital feedwater control system. Our acceptance of the advanced digital feedwater control incorporating the median signal selector described in WCAP-11931 is contingent upon undergoing an acceptable audit of the verification and validation program dealing with the separation requirements of IEEE 279 (1971) at the Westinghouse Electric Corporation Offices.

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The issuance of these amendments completes our work effort under TACs Nos. 69174 and 69175.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 87 to
   License No. DPR-42
2. Amendment No. 80 to
   License No. DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

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Sincerely,

Dominic C. DiIanni, Project Manager
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cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated July 18, 1988 as supplemented by letters dated September 15, 1988 and March 10, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 3, 1989

ATTACHMENT TO LICENSE AMENDMENTS NOS. 87 AND 80

FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60

DOCKETS NOS. 50-282 AND 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
TS.2.3-3	TS.2.3-3
TS.2.3-6	TS.2.3-6
Table TS.3.5-2 (Page 2 of 2)	Table TS.3.5-2 (Page 2 of 2)
Table TS.4.1-1 (Page 2 of 5)	Table TS.4.1-1 (Page 2 of 5)

K_6 = 0.002 for T > T', 0 for T < T'

τ_3 = 10, sec

f(ΔI) as defined in d. above.

f. Low reactor coolant flow per loop - ≥90% of normal indicated loop flow as measured at loop elbow tap.

g. Open reactor coolant pump motor breaker.

(1) Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

(2) Reactor coolant pump bus underfrequency - ≥58.2HZ

h. Power range neutron flux rate.

(1) Positive rate - ≤15% of rated power with a time constant ≥2 seconds

(2) Negative rate - ≤15% of rated power with a time constant ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

*c. Low steam generator water level - ≥15% of narrow range instrument in coincidence with steam/feedwater mismatch flow - ≤1.0 x 10^6 lbs/hr.

d. Turbine Generator trip

(1) Turbine stop valve indicators - closed

(2) Low auto stop oil pressure - ≥45 psig

e. Safety injection - See Specification 3.5

B. Protective instrumentation settings for reactor trip interlocks shall be as follows:

1. "At Power" reactor trips that are blocked at low power (low pressurizer pressure, high pressurizer level, and loss of flow for one or two loops) shall be unblocked whenever:

a. Power range neutron flux is ≥12% of rated power or,

b. Turbine load is ≥10% of full load turbine impulse pressure.

* The low steam generator water level in coincidence with steam/feedwater mismatch trip may be deleted following installation of the digital feedwater control system incorporating the median signal selector function. An acceptable verification and validation program shall be demonstrated to the satisfaction of the NRC Staff.

The other reactor trips specified in A.3. above provide additional protection. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of-load transient.

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip compliments the power range nuclear flux high and low trip to assure that the criteria are met for rod ejection from partial power.

The negative power range rate trip provides protection satisfying all IEEE criteria to assure that minimum DNBR is maintained above 1.30 for Exxon Nuclear fuel and above 1.17 for Westinghouse fuel for all multiple control rod drop accidents. Analysis indicates (Section 14.1.3) that in the case of a single rod drop, a return to full power will be initiated by the automatic reactor control system in response to a continued full power turbine load demand and it will not result in a DNBR of less than 1.30 for Exxon Nuclear fuel or 1.17 for Westinghouse fuel. Thus, automatic protection for a single rod drop is not required. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

References:

(1) FSAR 14.1.1
(2) FSAR Page 14-3
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.3
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11

INSTRUMENT OPERATING CONDITIONS FOR REACTOR TRIP

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS(1)	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
13. Undervoltage 4KV RCP Bus	1/bus	1/bus		Maintain hot shutdown
14. Underfrequency 4KV Bus	1/bus	1/bus		Maintain hot shutdown
15. Control Rod Misalignment Monitor				
a. Rod position deviation	1	-		Log data required by
b. Quadrant power tilt	1	-		TS 3.10 I and TS 3.10 J
16. RCP Breakers Open	2	1		Maintain hot shutdown
17. Safety Injection Actuation Signal	2	1		Maintain hot shutdown
**18. Lo Feedwater Flow	1/loop	1/loop		Maintain hot shutdown
19. Automatic Trip Logic including Reactor Trip Breakers	2	1		Notes 3, 4

Note 1: Automatic permissives not listed

Note 2: When bypass condition exists, maintain normal operation

Note 3: With the number of operable channels one less than the minimum operable channels requirement, be in at least hot shutdown within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1, provided the other channel is operable.

Note 4: When in the hot shutdown condition with the number of operable channels one less than the minimum operable channels requirement, restore the inoperable channel to operable status within 48 hours or open the reactor trip breakers within the next hour.

F.P. - Full Power

* - One additional channel may be taken out of service for low power physics testing

** - This trip may be deleted following installation of the digital feedwater control system incorporating the median signal selector function.

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
9.	Analog Rod Position	S(1) M(2)	R	T(2)	NA	1) With step counters 2) Rod Position Deviation Monitor Tested by updating computer bank count and comparing with analog rod position test signal
10.	Rod Position Bank Counters	S(1,2) M(3)	NA	T(3)	NA	1) With analog rod position 2) Following rod motion in excess of six inches when the computer is out of service 3) Control rod banks insertion limit monitor and control rod position deviation monitors
11.	Steam Generator Level	S	R	M	NA	
*12.	Steam Generator Flow Mismatch	S	R	M	NA	
13.	Charging Flow	S	R	NA	NA	
14.	Residual Heat Removal Pump Flow	S(1)	R	NA	NA	1) When in operation
15.	Boric Acid Tank Level	D	R(1)	M(1)	NA	1) Transfer logic to Refueling Water Storage Tank
16.	Refueling Water Storage Tank Level	W	R	M(1)	NA	1) Functional test can be performed by bleeding transmitter
17.	Volume Control Tank	S	R	NA	NA	
18a.	Containment Pressure SI Signal	S	R	M(1)	NA	Wide Range Containment Pressure 1) Isolation Valve Signal
18b.	Containment Pressure Steam Line Isolation	S	R	M	NA	Narrow Range Containment Pressure

* Following installation of the digital feedwater control system, only steam flow channels are checked, calibrated, and tested in accordance with this Table.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated July 18, 1988 as supplemented by letters dated September 15, 1988 and March 10, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 80, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 3, 1989

ATTACHMENT TO LICENSE AMENDMENTS NOS. 87 AND 80

FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60

DOCKETS NOS. 50-282 AND 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
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K_6 = 0.002 for T > T', 0 for T < T'

τ_3 = 10, sec

f(ΔI) as defined in d. above.

f. Low reactor coolant flow per loop - ≥90% of normal indicated loop flow as measured at loop elbow tap.

g. Open reactor coolant pump motor breaker.

(1) Reactor coolant pump bus undervoltage - ≥75% of normal voltage.

(2) Reactor coolant pump bus underfrequency - ≥58.2HZ

h. Power range neutron flux rate.

(1) Positive rate - ≤15% of rated power with a time constant ≥2 seconds

(2) Negative rate - ≤15% of rated power with a time constant ≥2 seconds

3. Other reactor trips

a. High pressurizer water level - ≤90% of narrow range instrument span.

b. Low-low steam generator water level - ≥5% of narrow range instrument span.

*c. Low steam generator water level - ≥15% of narrow range instrument in coincidence with steam/feedwater mismatch flow - ≤1.0 x 10^6 lbs/hr.

d. Turbine Generator trip

(1) Turbine stop valve indicators - closed

(2) Low auto stop oil pressure - ≥45 psig

e. Safety injection - See Specification 3.5

B. Protective instrumentation settings for reactor trip interlocks shall be as follows:

1. "At Power" reactor trips that are blocked at low power (low pressurizer pressure, high pressurizer level, and loss of flow for one or two loops) shall be unblocked whenever:

a. Power range neutron flux is ≥12% of rated power or,

b. Turbine load is ≥10% of full load turbine impulse pressure.

* The low steam generator water level in coincidence with steam/feedwater mismatch trip may be deleted following installation of the digital feedwater control system incorporating the median signal selector function. An acceptable verification and validation program shall be demonstrated to the satisfaction of the NRC Staff.

The other reactor trips specified in A.3. above provide additional protection. The safety injection signal trips the reactor to decrease the severity of the accident condition. The reactor is tripped when the turbine generator trips above a power level equivalent to the load rejection capacity of the steam dump valves. This reduces the severity of the loss-of-load transient.

The positive power range rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip compliments the power range nuclear flux high and low trip to assure that the criteria are met for rod ejection from partial power.

The negative power range rate trip provides protection satisfying all IEEE criteria to assure that minimum DNBR is maintained above 1.30 for Exxon Nuclear fuel and above 1.17 for Westinghouse fuel for all multiple control rod drop accidents. Analysis indicates (Section 14.1.3) that in the case of a single rod drop, a return to full power will be initiated by the automatic reactor control system in response to a continued full power turbine load demand and it will not result in a DNBR of less than 1.30 for Exxon Nuclear fuel or 1.17 for Westinghouse fuel. Thus, automatic protection for a single rod drop is not required. Administrative limits in Specification 3.10 require a power reduction if design power distribution limits are exceeded by a single misaligned or dropped rod.

<u>References</u>:

(1) FSAR 14.1.1
(2) FSAR Page 14-3
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.3
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11

TABLE TS.3.5-2 (Page 2 of 2)

INSTRUMENT OPERATING CONDITIONS FOR REACTOR TRIP

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS(1)	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
13. Undervoltage 4KV RCP Bus	1/bus	1/bus		Maintain hot shutdown
14. Underfrequency 4KV Bus	1/bus	1/bus		Maintain hot shutdown
15. Control Rod Misalignment Monitor				
a. Rod position deviation	1	-		Log data required by
b. Quadrant power tilt	1	-		TS 3.10 I and TS 3.10 J
16. RCP Breakers Open	2	1		Maintain hot shutdown
17. Safety Injection Actuation Signal	2	1		Maintain hot shutdown
**18. Lo Feedwater Flow	1/loop	1/loop		Maintain hot shutdown
19. Automatic Trip Logic including Reactor Trip Breakers	2	1		Notes 3, 4

Note 1: Automatic permissives not listed

Note 2: When bypass condition exists, maintain normal operation

Note 3: With the number of operable channels one less than the minimum operable channels requirement, be in at least hot shutdown within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.1, provided the other channel is operable.

Note 4: When in the hot shutdown condition with the number of operable channels one less than the minimum operable channels requirement, restore the inoperable channel to operable status within 48 hours or open the reactor trip breakers within the next hour.

F.P. - Full Power

* - One additional channel may be taken out of service for low power physics testing

** - This trip may be deleted following installation of the digital feedwater control system incorporating the median signal selector function.

TABLE TS.4.1-1 (Page 2 of 5)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
9.	Analog Rod Position	S(1) M(2)	R	T(2)	NA	1) With step counters 2) Rod Position Deviation Monitor Tested by updating computer bank count and comparing with analog rod position test signal
10.	Rod Position Bank Counters	S(1,2) M(3)	NA	T(3)	NA	1) With analog rod position 2) Following rod motion in excess of six inches when the computer is out of service 3) Control rod banks insertion limit monitor and control rod position deviation monitors
11.	Steam Generator Level	S	R	M	NA	
*12.	Steam Generator Flow Mismatch	S	R	M	NA	
13.	Charging Flow	S	R	NA	NA	
14.	Residual Heat Removal Pump Flow	S(1)	R	NA	NA	1) When in operation
15.	Boric Acid Tank Level	D	R(1)	M(1)	NA	1) Transfer logic to Refueling Water Storage Tank
16.	Refueling Water Storage Tank Level	W	R	M(1)	NA	1) Functional test can be performed by bleeding transmitter
17.	Volume Control Tank	S	R	NA	NA	
18a.	Containment Pressure SI Signal	S	R	M(1)	NA	Wide Range Containment Pressure 1) Isolation Valve Signal
18b.	Containment Pressure Steam Line Isolation	S	R	M	NA	Narrow Range Containment Pressure

* Following installation of the digital feedwater control system, only steam flow channels are checked, calibrated, and tested in accordance with this Table.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENTS NOS. 87 AND 80 TO

FACILITY OPERATING LICENSES NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated July 18, 1988, as supplemented by letters dated September 15, 1988 and March 10, 1989 Northern States Power Company (the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Units Nos. 1 and 2. The proposed amendments would change the Technical Specifications by eliminating the reactor trip device associated with steam/feedwater mismatch flow and low feedwater flow. Specifically, the proposed changes would impact the technical specification in the following areas

1. Specification 2.3.A.3(c) dealing with the reactor trip setpoints of "low steam generator water level - $\geq$ 15% of the narrow range instrument in coincidence with steam/feedwater mismatch flow - $\leq 1.0\times10^6$ lbs/hr" would be deleted.

2. Specification Table TS.3.5-2, item 18 dealing with low feedwater flow reactor trip, would be deleted.

3. Specification Table TS 4.1-1, item 12, Steam Generator Flow Mismatch, would be modified so that surveillance would be performed on steam flow channels only since feedwater flow channels would no longer be used in the protection circuit.

The associated TS bases would also be changed to reflect the removal of the low feedwater flow reactor trip described above.

The proposed changes would become effective after installing the digital feedwater control system incorporating the median signal selector function for each unit.

In support of the amendments requested, the licensee submitted by letter dated September 15, 1988, a report prepared by Westinghouse Electric Corporation (WCAP-11931 and non-proprietary version WCAP-11932) that describes the advanced digital feedwater control system containing the median signal selector.

2.0 EVALUATION

The Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 has two Westinghouse designed reactors that trip if a low-low water level is reached in any one of the steam generators. The reactor trip derived from the low-low water level in the steam generator protects the reactor from loss of the heat sink in the event of sustained steam/feedwater mismatch flow or low feedwater flow resulting from the loss of normal feedwater caused by a system pipe break inside or outside of containment. In the event of loss of feedwater for any reason, the reactor would trip when the water level in the steam generator falls to the low-low level setpoint in the reactor trip circuitry. Therefore, the low-low steam generator water level reactor trip circuit is provided for each steam generator which bounds the reactor trips initiated by the steam/feedwater mismatch flow and the low feedwater flow. The low-low steam generator water level reactor trip also ensures that sufficient initial heat removal capabilities (water inventory) is available in the steam generator to protect the core at the start of the transient. A review of the Prairie Island updated safety analysis report shows that no credit is taken for the reactor trip initiated by steam/feedwater mismatch flow or low feedwater flow in mitigating the consequences of any of the analyzed accidents. In cases such as loss of main feedwater, steam or feedwater pipe break inside or outside of containment, or loss of offsite power credit is taken for the low-low steam generator water level reactor trip to ensure safe shutdown.

The steam/feedwater mismatch flow and the low feedwater flow reactor trip were installed to satisfy the requirements of the Institute of Electric and Electronics Engineers Standard 279, 1971 (IEEE-279) paragraph 4.7.3 which is endorsed by the Code of Federal Regulations, 10 CFR Part 50.55a. Paragraph 4.7.3 of IEEE-279 states in part that a single random failure in a control system shall not prevent or block the proper action of a protection system from occurring. Specifically, the criteria of Paragraph 4.7.3 are not met when a single failure in the main feedwater control system prevents the low-low steam generator water level channels from tripping the reactor. Therefore, the low feedwater flow and the steam/feedwater mismatch flow reactor trip was installed in order to achieve an adequate substitute for meeting the separation criteria of Paragraph 4.7.3 of IEEE-279.

The staff has reviewed the report supporting the amendments requested prepared by Westinghouse Electric Corporation titled "Advanced Digital Feedwater Control System Median Signal Selector for Northern States Power Prairie Island Units 1 and 2" (WCAP-11931) describing the details of the enhanced feedwater control system. The feedwater control system is enhanced by the installation of the median signal selector which effectively eliminates the concern regarding a single random failure causing a control system action that results in a condition requiring protective action and preventing proper operation of a protection system channel designed to protect against this condition. Therefore, the mechanism for providing acceptable control and protection system interactions is achieved between the steam generator low-low water level protective function and the feedwater control system in accordance with the requirements of IEEE Standard 279 (1971). The isolation devices providing protection for the steam generator low-low water level reactor protective function are Foxboro Model 66

BC-O style C current repeater that were subjected to an extensive testing program by Westinghouse Electric Corporation. The results of this testing program has been reported in the Westinghouse Electric Corporation Topical Report WCAP-7508-L (December 1970) and the staff found the test results acceptable as discussed in our letter dated June 6, 1973. By letter dated March 10, 1989 the licensee provided supplemental information confirming that the isolation devices are covered within the scope of the testing program for the maximum credible faults. As reported in the Westinghouse Topical Report WCAP-7685-A, (May 1975), the isolation amplifiers provide an effective electrical barrier for the input (protection side) signal when the output (control side) signal was subjected to faulted conditions. In addition, the licensee verified that the maximum and minimum signal levels allowed to pass through the isolation devices will in no way degrade the median signal selector such that damage would occur to the main feedwater flow control system. The verification and validation program performed at Westinghouse does include an actual testing program supporting the claim of adequate operability of the median signal selector upon the receipt of the maximum and minimum signal levels that normally passes through the isolators. While the staff has not yet performed an audit of the verification and validation program, our review is complete to a point where we find the application of the advanced digital feedwater control system containing the median signal selector acceptable. This acceptability is contingent upon our finding an adequate verification and validation program during the forthcoming audit at the Westinghouse offices.

On this basis, the reactor trip initiated by steam/feedwater mismatch and low feedwater flow are no longer necessary or required.

In conclusion, the safety analysis shows that no credit is taken for the reactor trip initiated by the steam/feedwater mismatch flow and low feedwater flow in mitigating the consequences of any of the analyzed design bases accidents. The initial installation of this trip was for the purpose of satisfying the single random failure requirement specified in IEEE 279 (1971) paragraph 4.7.3 for control and protective system interactions. The advanced feedwater control system provides an acceptable method of resolving the interaction concern between the feedwater control and steam generator low-low water level protective function. The control and protection system interaction meets the requirements specified in paragraph 4.7.3 of IEEE 279. The acceptability of the protective function meeting the separation requirement of IEEE 279 (1971) is predicted upon an acceptable audit of the manufacturer's verification and validation program associated with the software for the median signal selector. On this basis, the staff finds the proposed changes involving the elimination of the steam/feedwater mismatch flow and the low feedwater flow reactor trip acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on April 3, 1989 (54 FR 13445).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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