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Docket Nos. 50-282
and 50-306

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

On February 2, 1983, the Commission issued Amendment Nos. 61 and 55 to Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant. The amendments revised the Technical Specifications concerned with Steam Generator Tube Surveillance, Steam Exclusion System, Auxiliary Feedwater System reporting requirements, Administrative Controls, and corrected typographical errors.

Unfortunately, three of the TS pages issued with Amendments 61 and 55 contained errors of a clerical/administrative nature, and two pages were omitted. Enclosed are corrected pages TS.3.1-3, Table TS.4.1-1 (Pg. 3 of 5), Table TS.4.12-1, and the missing pages TS.6.7-3a and TS.6.7-4.

Please accept our apologies for any inconvience these errors may have caused you.

Sincerely,

Original signed by

Dominic C. DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
As stated

cc: See next page

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OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL				
SURNAME	PMKreutzer	DCDiIanni/pn	RAClark				
DATE	2/ /83	2/14/83	2/15/83				

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

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Basis

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth $\geq$50% of the 0.050-inch tube wall thickness as being unacceptable for power operation. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.[2]

Part A of the specification requires that both reactor coolant pumps be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power physics tests, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load.[1]

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
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
18c.	Containment Pressure Containment Spray	S	R	M	NA	
18d.	Annulus Pressure (Vacuum Breaker)	NA	R	R	NA	
19.	Deleted					
20.	Boric Acid Make-up Flow Channel	NA	R	NA	NA	
21.	Containment Sump Level	NA	R	R	NA	Includes Sumps A, B, and C
22.	Accumulator Level and Pressure	S	R	R	NA	
23.	Steam Generator Pressure	S	R	M	NA	
24.	Turbine First Stage Pressure	S	R	M	NA	
25.	Emergency Plan Radiation Instruments	*M	R	M	NA	Includes those named in the emergency procedure (referenced in Spec. 6.5 A.6)
26.	Protection Systems Logic Channel Testing	NA	NA	M	NA	Includes auto load sequencers

Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
27. Turbine Overspread Protection Trip Channel	NA	R	M	NA	
28. Deleted					
29. Deleted					
30. Deleted					
31. Seismic Monitors	R	R	NA	NA	Includes those reported in Item 4 of Table TS.6.7-1
32. Coolant Flow - RTD Bypass Flowmeter	S	R	M	NA	
33. CRDM Cooling Shroud Exhaust Air Temperature	S	NA	R	NA	FSAR page 3.2-56
34. Reactor Gap Exhaust Air Temperature	S	NA	R	NA	FSAR page 5.4-2
35. Post-Accident Monitoring	M	R	NA	NA	Includes all those in FSAR Table 7.7-2 and Table TS.3.15-1 not included elsewhere in this Table
36. Steam Exclusion Actuation System	W	R	M	NA	See FSAR Appendix I, Section I.14.6
37. Overpressure Mitigation System	NA	R	R	NA	Instrument Channels for PORV Control Including Overpressure Mitigation System
38. Degraded Voltage 4KV Safeguard Busses	NA	R	M	NA	
39. Loss of Voltage 4KV Safeguard Busses	NA	R	M	NA	

TABLE TS.4.12-1

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC.	N/A	N/A

S=3%; When two steam generators are inspected during that outage.
S=6%; When one steam generator is inspected during that outage.

Unit 1 - Amendment No. 31, 61
Unit 2 - Amendment No. 25, 55

7. Report of Safety and Relief Valve Failures and Challenges. An annual report of pressurizer safety and relief valve failures and challenges shall be submitted prior to March 1 of each year.

B. Reportable occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date. Unless explicitly stated, the requirements of this section do not apply to the fire protection systems and measures contained in Sections 3.14/4.16, the radiological effluent limitations and measures in Sections 3.9/4.17, or the radiological environmental monitoring program in Section 4.10. Fire protection reporting requirements have been separately specified in those sections.

1. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Administrator of the appropriate Regional NRC Office or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

   (a) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

   Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items B.1(e), B.1(f), or B.2(a) below.

   (b) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

   Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item B.2(b) below.

   (c) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

   Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.