Central file

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

April 30, 1979

Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D.C. 20555

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

The following is submitted in response to IE Bulletin No. 79-01:

Stem-mounted limit switches of a type similar to those described in the bulletin (Namco Model SL3B2W) are used on safety-related valves inside containment at Monticello. However, these switches are not required to perform their safety function for events that would result in a LOCA environment. Therefore, no action is planned with respect to this bulletin.

The environmental qualification of safety-related electrical equipment required to function under accident conditions is described in Attachment 1 to this letter.

Yours very truly,

L. J. Wachter

L. J. Wachter
Vice President
Power Production and
System Operation

cc: Mr. G. Charnoff
Mr. James G. Keppler

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Qualification Data for Safety Related Electrical Equipment Required to Function in an Accident Environment

COMPONENT DESCRIPTION	ACCIDENT ENVIRONMENT	QUALIFICATION ENVIRONMENT	QUALIFICATION METHOD	QUALIFICATION DOCUMENT	REMARKS
MSIV Solenoid Valves (Automatic Valve Co. Model No. C-4988-15)	Note #1	345°F, 110 psig	Test/Analysis	Rockwell Report #2792-03-02, Rev. 1.	Note #2
SRV Solenoid Valves (Automatic Valve Co. Model No. C-5450)	Note #1	340°F, 65 psig, 3×10^7 Rads	Sequential Test	GE Plant Equipment Design Memo #126-62.	
Inboard Isolation Valve Actuators (Limitorque Type SMB)	Note #1	329°F, 90 psig	Test/Analysis	Franklin Institute Report #F-C2232-01	Note #4
Inboard Reactor Water Sample Isolation Valve Solenoid Valve (Asco Model No. THT-8317A23)	Note #1	Note #3	Analysis	SRI #168	Note #3
Recirc Pump Discharge valve Actuators (Limitorque Type SMB)	Note #1	210°F, 7" W.G.	Test/Analysis	Franklin Institute Report #F-C3271	Note #4
Containment Electrical Penetrations (General Electric Type NSO-2, NSO-3, NSO-4)	Note #1	340°F, 56 psig, 4×10^7 Rads	Sequential Test	Letter, G.G. Sherwood (GE) to USNRC, dated 12/2/77 and "Qualification Report for F01 Electrical Penetration Assembly"	
Control Cables (inside containment) (General Electric SI-58109)	Note #1	340°F, 62 psig, 4×10^7 Rads	Sequential Test	Wyle Labs Reports 44114-1 and 44114-2.	

Qualification Data (Continued)

COMPONENT DESCRIPTION	ACCIDENT ENVIRONMENT	QUALIFICATION ENVIRONMENT	QUALIFICATION METHOD	QUALIFICATION DOCUMENT	REMARKS
Power Cables (inside containment) (Rockbestos Firewall III)	Note #1	340°F, 104 psig, 20 x 10^7 Rads	Sequential Test	"Qualification of Firewall III Class IE Electric Cables" dated February 1, 1977 and "Class 1E Qualification of Raychem Splices" dated April 14, 1978 (The Rockbestos Company).	
Penetration Splices (Raychem Type WCSF-N)	Note #1	See Remarks	See Remarks	See Remarks	Qualified with power & control cables described above.
ECCS Pumps and Valves Outside Containment	Note #5	Note #5	Analysis	FSAR Section 14.10.1.3	
SBGT System Components	Note #6	Note #6	Analysis	FSAR Section 5.3.4	

NOTE 1: The primary containment pressure and temperature responses which determine the LOCA accident environment are shown in FSAR Figures 5-2-14 and 5-2-15, respectively. Pressure and temperature extremes from these figures are 41 psig and 281°F. For qualification purposes, the maximum temperature is extended to 340°F to account for possible superheating of the containment atmosphere as the result of a steam line break. Radiation conditions inside the drywell are shown in FSAR Table 14-10-4. The 180 day dose at the interior surface of the drywell is projected to be 3.3×10^7 Rads.

NOTE 2: The MSIV solenoid valves and motor operated valve actuators have not been tested under actual radiation conditions. Such testing is considered to be unnecessary since these components perform their function immediately upon detection of an accident and are not required to function thereafter. The anticipated radiation dose received by the components during operation (approximately 10^5 Rads) would not prevent them from performing their intended function.

NOTE 3: The inboard reactor water sample isolation valve solenoid valve has not been formally qualified for operation in an accident environment. There is assurance, however, that in the event of an accident, the valve would perform its intended function since it is actuated immediately upon detection of the accident and is not required to function thereafter. Subsequent failure of the valve would not cause the sample isolation valve to re-open.

However, to provide assurance that this valve could be reopened for post - LOCA coolant sampling, it will be replaced by a fully qualified model at the next refueling outage.

NOTE 4: The recirc pump discharge valve actuator motors have Class B insulation. Actuators with Class B insulation have been qualified in a steam environment for 6 hours as described in the referenced qualification document. The test revealed acceptable performance at 210°F.

The drywell temperature response to a LOCA (FSAR Figure 5-2-15) indicates a maximum temperature of 281°F is attained approximately 5 seconds from the start of the LOCA. Substantial conservatism exists in this prediction as discussed in paragraph 5.2.3.2 of the FSAR. In ECCS calculations, the recirc valves are assumed to perform their isolation function within approximately 43 seconds.

The actuators are equipped with a weatherproof housing which will delay the temperature response of the motor insulation with respect to drywell temperature.

Since the actuators will not fail until the insulation temperature exceeds at least 210°F, it is concluded that the valves will perform their function before the LOCA has a significant effect on the insulation.

However, to provide increased assurance that these valves will function in a LOCA environment, the actuators will be replaced with fully qualified models at the next refueling outage.

NOTE 5: The accident environment experienced by the ECCS electrical equipment within the secondary containment is described and evaluated in FSAR Section 14.10.1.3 using the extremely conservative assumptions of TID 14844. The anticipated dose levels would not be expected to affect equipment operability.

NOTE 6: The accident environment experienced by the SBGT System components is described and evaluated in FSAR Section 5.3.4 using the extremely conservative assumptions of TID 14844. The anticipated dose levels would not be expected to affect equipment operability.