Nebraska Public Power District

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June 22, 1979

Director, Division of Reactor Operations Inspection
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Response to IE Bulletin Nos. 79-01 and 79-01A
"Environmental Qualification of Class IE Equipment"
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

References: 1) Letter from J. M. Pilant (NPPD) to Director, Division of Operating Reactors, dated February 21, 1979

Dear Sir:

IE Bulletin No. 79-01 required that Nebraska Public Power District provide written evidence of the qualification of safety-related electrical equipment required to function under accident conditions. Enclosed as Attachment 1 please find information concerning equipment located outside of the primary containment. Attachment 2 addresses the equipment located inside the primary containment. Our review has determined that proper environmental qualification documentation exists for all installed safety-related electrical equipment required to function under accident conditions.

Item 2 of Bulletin No. 79-01 required a determination as to whether certain listed types of stem mounted limit switches are being used on safety-related valves located inside the containment. This determination was presented in Reference 1, and during the April 1979 refueling outage the applicable NAMCO switches were replaced with qualified models.

IE Bulletin No. 79-01A, Item 2a, required that a determination be made as to the qualification of ASCO solenoid valves subject to a LOCA environment. Cooper Nuclear Station does not have any ASCO solenoid valves subject to a LOCA environment. In response to item 2b, a preventive maintenance program is being established such that the solenoid coil, manual operator (if applicable), and the resilient parts of the ASCO solenoid valves will be replaced in accordance with the time period established by the manufacturer and documented as the qualified life of the assembled component. The preventive maintenance program, which includes the periodic inspection of each solenoid valve on a safety-related system, will be completed by the end of the station's 1980 refueling outage.

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If you have any questions, or require additional information on this matter, please contact me.

Sincerely,

Jay M. Pilant

Jay M. Pilant
Director of Licensing
and Quality Assurance

JDW/cmk

Attachment

cc: K. V. Seyfrit

ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT OUTSIDE OF PRIMARY CONTAINMENT

On December 18, 1972, and January 19, 1973, the Staff sent letters to NPPD requesting a detailed design evaluation to substantiate that the design of CNS is adequate to withstand the effects of a postulated rupture in any high energy fluid piping system outside the primary containment, including the double-ended rupture of the largest line in the main steam and feedwater system. Criteria for conducting this evaluation were included in the letters. Criterion 13 of the January 19, 1973 letter required that: "Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy fluid line break."

In response to the Staff's letters, Amendment No. 20 to the Safety Analysis Report (SAR) was filed by NPPD on April 13, 1973. NPPD submitted additional documentation in Amendment No. 25 to the SAR dated June 8, 1973. A partial summary of the criteria and requirements for the analysis is as follows:

1. Protection of equipment necessary to shutdown the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from all effects resulting from ruptures in pipes carrying high-energy fluid, up to and including a double-ended rupture of such pipes, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig.

2. In addition, protection of equipment necessary to shutdown the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from the environmental effects (including the effects of jet impingement) resulting in pipes carrying fluid routed in the vicinity of this equipment. The size of the cracks are assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

The analysis included the following piping systems containing high energy fluids.

- Main Steam System
- Feedwater and Condensate System
- Reactor Core Isolation Cooling System
- High Pressure Coolant Injection System
- Reactor Water Cleanup System
- Residual Heat Removal System
- Control Rod Drive System
- Heating Systems
- Sample Lines

The effects of high energy pipe breaks were evaluated on the following systems and components which would be necessary (in various combinations, depending of the effects of the break) to safely shutdown, cooldown, and maintain cold shutdown conditions:

. 282 228

1. General

   - Control and Instrument Cables
   - Electrical Distribution Systems
   - Emergency DC Power Supply (batteries)
   - Emergency AC Power Supply (diesels)
   - Heating and Ventilation Systems

2. Reactor Control

   - Control Rod Drive System
   - Neutron Monitors
   - Instrumentation for Reactor Temperature and Water Level
   - Standby Liquid Control System

3. Core Cooling Systems

   - Condensate and Feedwater
   - Residual Heat Removal System (Including LPCI Mode)
   - Reactor Core Isolation System
   - Automatic Depressurization System
   - High Pressure Coolant Injection System
   - Core Spray System

4. Service Systems

   - Service Water System
   - Residual Heat Removal Service Water System
   - Reactor Building Closed Cooling Water System

The results of this design evaluation determined that Cooper Nuclear Station can be shutdown safely and maintained in a safe shutdown condition assuming that no electrical equipment affected by a pipe break environment functions and that remedial action or utilization of redundant equipment in unaffected areas has been initiated in each case.

The Staff reviewed Amendments 20 and 25 and reached the following conclusion as contained in Supplement No. 1 to the Safety Evaluation of the Cooper Nuclear Station issued July 16, 1973:

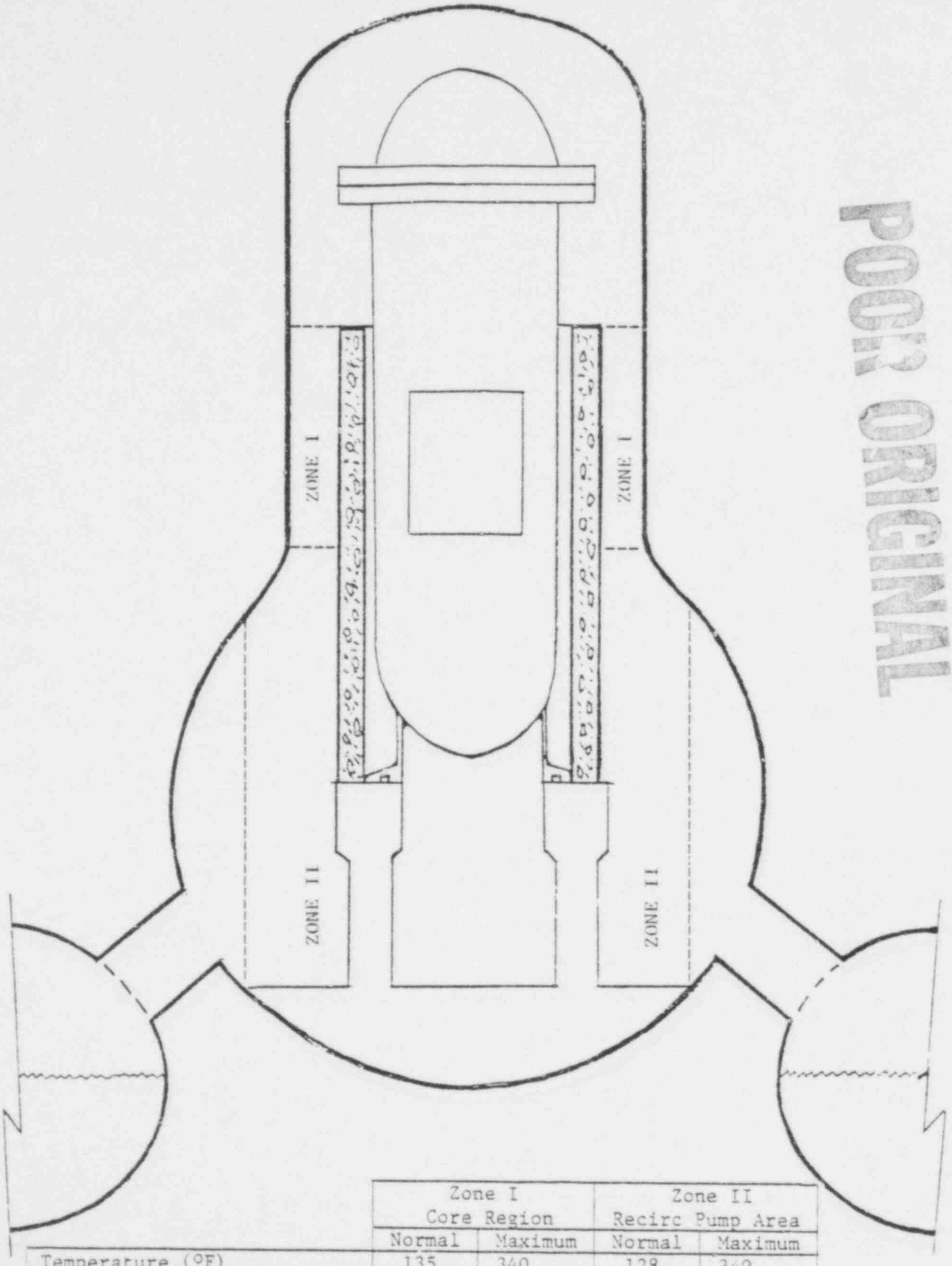
> "We have reviewed the information submitted to use and based on this review and our discussions with NPPD, find evidence of a thorough assessment of high-energy line failures outside containment. We concur in the conclusions of the applicant relative to the need for the modifications listed above, and will review the design details and implementation of the modifications prior to power operation. The other areas of the plant, wherein no modifications are proposed, have been reviewed and we concur with the applicant's evaluation that, in these areas, the potential consequences of postulated high-energy piping failures will not prevent the capability to achieve safe cold shutdown conditions consistent with the single-failure and redundancy requirements."

Based on the results of the design evaluation presented in Amendments 20 and 25, this response to IE Bulletin No. 79-01 will not further address the environmental qualification of safety-related electrical equipment located outside of the primary containment.

ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT INSIDE PRIMARY CONTAINMENT

Environment

The most extreme environment calculated for the CNS primary containment resulting from a design basis loss-of-coolant accident is an internal drywell pressure of 58 psig coincident with drywell and suppression chamber temperatures of 281°F. Safety-related electrical equipment located in the primary containment has been qualified for saturated steam conditions at 62 psig (340°F). Equipment located in radiation environments throughout the plant has been qualified to withstand an integrated gamma radiation dose of 1.0×10^7 Rads. The normal and maximum expected environmental conditions for the areas of the primary containment where safety-related electrical equipment is located are shown on Figure 1.



	Zone I Core Region		Zone II Recirc Pump Area	
	Normal	Maximum	Normal	Maximum
Temperature (°F)	135	340	128	340
Pressure (psig)	2.0	56	2.0	56
Relative Humidity (%)	55	100	55	100
Radiation Dose Rate (R/hr)	50	1.3E+06	25	1.3E+06

(Ref: GE Spec 22A2928)

282 232

Figure 1

ASSUMPTIONS USED IN
COMPILING QUALIFICATION DATA
FOR TABLE 1

1. All installed safety-related electrical equipment in the drywell was considered. Table 1 lists all components required to function for the prevention or mitigation of accidents which could affect the public health and safety <u>and</u> subject to potential damage by one or more of the accidents. Inactive components which are only required to remain de-energized and intact following an accident were reviewed for appropiate qualification documentation, but are not contained in the response to this IE Bulletin.

2. It has been verified that the electrical components listed in Table 1 installed at CNS are in fact the same components (i.e. model number, etc.) for which qualification documentation exists.

3. The Limiting Qualification Parameters Column lists only the parameters for the most severe temperature and pressure environment and the qualification parameters for the long term environment. Additional qualification parameters for intermediate short term conditions are not presented in Table 1.

TABLE 1

COOPER NUCLEAR STATION

ENVIRONMENTAL QUALIFICATION DATA FOR SAFETY-RELATED ELECTRICAL EQUIPMENT

RESPONSE TO IE BULLETIN NO. 79-01

Equipment Description	Functions	Drywell Zone	Limiting Qualification Parameters	Qualification Method	Qualification Reference Documentation
NAMCO Limit Switches Type EA180-31302	Position indication on Main Steam Isolation Valves (MSIV's)	II	340°F 70 psig 100% RH $2x10^8$ RADS 3 hours	Sequential	1
Automatic Valve Co. Solenoid Valves Model #C-5450-5 supplied by Target Rock P/N 54	Air to MSRV's for blowdown operation. Must be energized during blowdown.	I	340°F 250°F 62 psig 25 psig 100% RH $3x10^7$ RADS 2 hrs min 90 hours	Sequential	2
Limitorque Motor Operated Valves: Model No. Valve Number		I & II	340°F 212°F 104 psig 100% RH $1x10^7$ RADS 1 hour 18 hours	Sequential for Ref. Docs. 3 & 4. Separate effects for Ref. Docs. 5 through 11.	3 though 11
SMB-000-2 MO-274A	Bypass to RHR Testable Check Valves				
SMB-000-2 MO-274B	Bypass to RHR Testable Check Valves				
SMB-2 MO-18	Reactor Shutdown Cooling Suction				
SMB-00 MO-32	Reactor Head Spray Isolation				
SMB-1 MO-15A	HPCI Isolation Valve				
SMB-00 MO-15B	RCIC Isolation Valve				
SB-3-100 MO-53A,B	Recirculation Valves				
SMB-000-5 MO-54A,B	Recirculation Bypass Valves				
SMB-00 MO-15C	RWCU Isolation Suction				
SMB-000 MO-15A,B	Bypass to Core Spray Testable Check Valves				

282 234

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TABLE 1
COOPER NUCLEAR STATION
ENVIRONMENTAL QUALIFICATION DATA FOR SAFETY-RELATED ELECTRICAL EQUIPMENT
RESPONSE TO IE BULLETIN NO. 79-01

Equipment Description	Functions	Drywell Zone	Limiting Qualification Parameters	Qualification Method	Qualification Reference Documentation
Model No. / Valve Number SMB-000 MO-74	Main Steam Drain Isolation Valve				
SMB-000 MO-57A	RHR Discharge to Radwaste Inboard				
SMB-000 MO-17	Bypass to RCIC Testable Check Valve				
SMB-000 MO-57B	Bypass to HPCI Testable Check Valve				
Electrical Penetration GE type NS02, NS03, NS04	Various	II	340^oF 62 psig 100% RH $4x10^7$ RADS 15 min. 281^oF 63 psig 240 hours	Separate Effects for Ref. Doc. 12. Sequential for Ref. Doc. 13.	12, 13
Electrical Cable Cerro CLP Insulation Neoprene Jacket	Various	I, II	340^oF 62 psig 100% RH $1x10^7$ RADS 2 hours 160^oF 23 psig 24 hours	Sequential	14, 20
Electrical Cable Boston Insulation Bostran CSPE Jacket	Various	I, II	340^oF 62 psig 100% RH $2x10^8$ RADS 3 hours 200^oF 135 days	Sequential	15
Electrical Cable Boston Insulation Polyorgamosiloxame Insulation Chlorosulforated Polyethlene Jacket	Various	I, II	340^oF 62 psig 100% RH $2x10^8$ RADS 3 hours	Sequential	16

282 235

TABLE 1
COOPER NUCLEAR STATION
ENVIRONMENTAL QUALIFICATION DATA FOR SAFETY-RELATED ELECTRICAL EQUIPMENT
RESPONSE TO IE BULLETIN NO. 79-01

Equipment Description	Functions	Drywell Zone	Limiting Qualification Parameters	Qualification Method	Qualification Reference Documentation
Electrical Cable Kerite HT Insulation FR Jacket	Various	I, II	320°F 214°F 82 psig 23 psig 100% RH $1x10^8$ RADS 13 hours 18 days	Sequential	17, 19
Electrical Cable Raychem Rayolin F Insulation Flamtrol Jacket	Various	I, II	360°F 60 psig 100% RH $1x10^8$ RADS 27 hours	Sequential	18

282 236

QUALIFICATION DOCUMENTATION REFERENCE

1. NPPD P.O. 147467 to NAMCO Controls. Certificate of Compliance certifying compliance to IEEE 323-1974, 344-1975, 382-1972 per NAMCO report of 3-3-78.

2. GE Test Report PED MEMO 126-62.

3. Report No. C2232 "Test of a Limitorque Valve Operator Under a Simulated Reactor Containment Post-Accident Steam and Chemical Environment" dated November 1968 by Franklin Institute Research Labs.

4. Qualification Type Test by Limitorque No. B-0009-9-2-75.

5. Aero-Nav Report 5771 Seismic Test on SMB-000-5 dated 4-30-75.

6. Aero-Nav Report 5772 Seismic Test on SMB-0-25DC dated 7-28-75.

7. Aero-Nav Report 5773 Seismic Test on SMB-3-100 dated 7-22-75.

8. Aero-Nav Report 5774 Seismic Test on SB-0-25 dated 7-23-75.

9. Wyle Report 43059-02 Seismic Test on SMB-5 & Modutronic dated 10-30-75.

10. Aero-Nav Report 5770 Seismic Test on SB-3-100 dated 4-24-75.

11. Aero-Nav Report 5-6167-5 Seismic Test on SMB-1-25/H4BC dated 11-18-75.

12. Qualification Test for F01 Electrical Penetration Assembly dated 4-30-71 LOCA Simulation at 340°F, 56 psig, 100% RH.

13. Electrical Penetration Assembly Prototype Testing dated 3-16-70 No. EPAPTOS.

14. Cerro Wire and Cable Company Test Report dated Oct. 13, 1970 - Aging, Irradiation, LOCA Test to 340°F, 62 psig, Fire Test.

15. Boston Insulated Wire & Cable Test No. 75A025 dated 4-15-75 - Aging, Irradiation, LOCA Test to 340°F and 105 psig.

16. Boston Insulated Wire & Cable Test No. 8-21 dated 11-6-70 - Irradiation, LOCA Test to 340°F and 65 psig.

17. Report on the Effects of Gamma Radiation and Autoclaving on Kerite Power and Control - Irradiation, LOCA Test to 320°F and 82 psig.

18. Raychem Corp. Test Report EM#518 dated 5-4-72 - Aging, Irradiation, LOCA Test to 360°F and 60 psig.

19. Kerite Test Report - Long Term Autoclave - 1970.

282 237

20. Franklin Institute Report F-C2750 March 1970.