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 AUTH. NAME AUTHOR AFFILIATION
 MUSOLF, D. Northern States Power Co.
 RECIP. NAME RECIPIENT AFFILIATION
 Office of Nuclear Reactor Regulation, Director

SUBJECT: Discusses resolution of SER open items re environ
 qualification of safety-related electrical equipment, per
 831212 meeting. General methodology used to establish aging
 & radiation qualification encl.

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Northern States Power Company

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

February 10, 1984

Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, D.C. 20555

Monticello Nuclear Generating Plant
Docket No. 50-263 License No. DPR-22

Resolution of Safety Evaluation Report for Environmental
Qualification of Safety-Related Electrical Equipment

On December 12, 1983 a meeting was held between members of the NRC staff and Northern States Power Company (NSP) to resolve all equipment qualification open items for the Monticello Nuclear Generating Plant. Included in our meeting were discussions concerning the following items: the equipment qualification program at Monticello; the general methodology used to resolve qualification deficiencies identified in the SER and attached Technical Evaluation Report (TER), dated January 4, 1983; the specific resolution of identified deficiencies for each equipment item on the master list; the schedule for completion and Justification for Continued Operation (JCO) for presently unqualified equipment items; and NSP's compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," which became effective February 22, 1983.

At the conclusion of our meeting, NSP was requested to prepare meeting minutes thereby documenting our discussions. This letter provides the general discussion minutes and is supplemented with two enclosures: Enclosure 1 provides the general methodology used to establish aging and radiations qualification. Enclosure 2 provides the specific resolution for each equipment item reviewed in the TER, and incorporates comments received from the Staff during our meeting.

Three categories of electrical equipment were identified in 10 CFR 50.49 as requiring environmental qualification. Equipment described in paragraph (b) (1) of 10 CFR 50.49 has been identified through a review of the accident analyses provided in the FSAR, a review of the emergency procedures, a review of safety system flow diagrams and the Q-List, and a review of the installed equipment locations with respect to postulated harsh environmental zones. This equipment has been identified in our previous submittals to IE Bulletin 79-01B and in our response to 10 CFR 50.49 dated May 19, 1983.

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NORTHERN STATES POWER COMPANY

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Paragraph (b) (2) of 10 CFR 50.49 requires that licensees identify "Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions..." The methodology that was used to identify such equipment is summarized below:

The elementary wiring diagrams of safety-related electrical equipment defined in paragraph (b) (1) of 10 CFR 50.49 were reviewed to identify any auxiliary devices electrically connected directly into the control or power circuitry of the safety-related equipment whose failure due to postulated environmental conditions could prevent the required operation of the safety-related equipment.

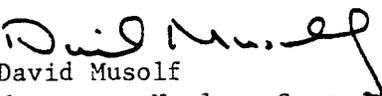
Nonsafety-related electrical circuits indirectly associated with the electrical equipment defined in paragraph (b) (1) of 10 CFR 50.49 by common power supply or physical proximity were considered by a review of the plant electrical design including the use of properly coordinated protective relays, circuit breakers, and fuses for electrical circuit fault protection.

The Monticello plant is now shutdown for an extensive refueling and maintenance outage. All qualification deficiencies will be resolved prior to startup following this outage. For this reason no justification for continued operation is provided.

Certain post-accident monitoring equipment was identified in our May 19, 1983 response to 10 CFR 50.49 in accordance with paragraph (b) (3). In addition, our response to NUREG-0737, Supplement 1 related to Regulatory Guide 1.97 was transmitted to the NRC on December 30, 1983. This letter identified the qualification requirements and implementation schedule for Regulatory Guide 1.97 equipment. This equipment will be qualified and added to the master equipment list prior to startup following the current refueling outage.

The master equipment list contains the equipment located in a harsh environment necessary to mitigate the consequences of the Design Basis Accidents (DBAs) identified in the FSAR, HELB's, and flooding resulting from these events.

Please contact us if you require any additional information.


David Musolf
Manager - Nuclear Support Services

cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
G Charnoff

Enclosure

ENCLOSURE 1

AGING QUALIFICATION METHODOLOGY

NRC IE Bulletin 79-01B established the DOR Guidelines as the criteria for evaluating the qualification and associated documentation of installed safety-related electrical equipment. The methodology and sequence used by Northern States Power to assess thermal aging qualification, and to establish a qualified life for safety-related equipment at Monticello Nuclear Generating Plant, are consistent with the DOR Guidelines.

The DOR Guidelines state that component type testing of equipment identical in design and construction to the installed component is the preferred qualification method; however, alternative methods are acceptable if justification for their use is provided.

Detailed analyses used to establish the qualified (thermal aging) life for equipment items are included in the November 1981 SER Response and the Monticello Central File. The methodology used to assess thermal degradation is outlined below.

1. Identify the specific component materials susceptible to age-related thermal degradation. Metallic components were assumed not to be susceptible. Where the specific material of construction could not be identified by the manufacturer, conservative data for typical materials was assumed.
2. Identify the thermal aging properties of susceptible materials. Data sources included those typically used throughout the nuclear industry, some of which were developed specifically to assist in the evaluation of thermal aging degradation. For example:
 - Appendix C of DOR Guidelines
 - Material manufacturer technical specifications
 - EPRI Report NP-1558
 - Test lab data
 - Other sources, as applicable

For each material, the most applicable data available was used with respect to both the tested configuration (bulk, thin film, o-ring, etc.) and the physical property pertinent to the function of the component. For example, the critical physical property for a diaphragm would be retention of elongation, flexure strength, or tensile strength, as appropriate. The aging data utilized incorporated conservative failure criteria of the material relative to retention of the critical property.

- b. Conservatively high normal operating temperatures were used to calculate the qualified lives of all materials.
 - c. A conservative activation energy of 0.5eV was assumed for materials where adequate aging data was unavailable.
 - d. Maximum total temperature (ambient + self-heating) was used to evaluate all materials in equipment which generates heat (energized solenoids, running motors, etc.).
4. Evaluate the effect of material degradation. When a qualified life for a component material was less than required, the effect of material failure was investigated. If it could be determined that failure of the material or subcomponent would not affect the operability of the equipment, the failed material was not considered limiting, and next most limiting material/component was used to establish the qualified life.

For equipment exposed to small and/or short duration accident temperature transients, normal age-related thermal degradation will not be significantly accelerated during post-accident conditions. Therefore, catastrophic failure due to thermal aging under these circumstances is very unlikely.

The methodology described above complies with NRC requirements established in the IE Bulletin 79-01B to provide reasonable assurance that the equipment will perform its safety function when required.

RADIATION QUALIFICATION METHODOLOGY

The DOR Guidelines specifically allow for qualification by analysis for radiation. Analysis was used for equipment under the following two circumstances: 1) to establish qualification when radiation type-testing was not performed, and 2) to supplement qualification of equipment for which the radiation testing performed did not envelop the required dose, but where no failure due to radiation occurred.

Equipment drawings and correspondence with the vendor/manufacturer were used to determine the non-metallic materials of construction and the pertinent failure criterion of each material. A literature search of widely used radiation studies on organic materials in nuclear applications was then made to establish the threshold gamma radiation dose for the failure criterion (e.g., % loss of tensile strength) for each constituent component. The limiting radiation damage threshold was compared to the specified radiation dose for the component. Only equipment for which the analyses dose exceeded the specified dose was considered qualified.

3. Determine a qualified life for thermal aging. Establishing a qualified life was initially based on extrapolation of type-test data, where available. The Arrhenius method was utilized, with an activation energy for the material which was consistent with the thermal aging property being evaluated, to extrapolate accident pre-aging test data.

Since pre-aging was not performed as part of the test sequence in many cases, and because significant margins often exist between the specified and tested environmental profiles, the Arrhenius method and material properties were used to draw a comparison between the test profile and the specified accident profile. The resultant "overtesting" was then used to establish a qualified life. In cases where type test data or an inadequate "overtest" were not available, the Arrhenius method was used to compare applicable materials test data to normal plant conditions in order to establish a qualified life.

Where pre-accident aging test data was not available for the component, accident "overtesting" was considered the next most conservative measurement of the equipment qualified thermal aging life, based on the following considerations:

- Materials of construction in the test specimen afford the most accurate and reliable indication of how the materials in the installed equipment will age.
- The atmosphere in the test chamber is acceptably similar to normal plant conditions to simulate normal aging. No accelerated aging test environment can accurately simulate all aspects of normal plant aging, and air oven aging is the most widely used accelerated aging method. However, air oven aging does result in an extremely dry environment which excludes the considerable aging effects of moisture. The test chamber atmosphere during LOCA/HELB accident testing also has disadvantages, in that it has a lower than normal concentration of oxygen. Oxygen is present in the chamber in some quantities at the start of testing and, as chamber temperature rises, gases (including oxygen) are driven out of solution from water present in the chamber in the form of spray solution or a vapor source pool. This oxygen is available to participate in aging reactions. Thus, test chamber conditions are adequate to provide reasonable assurance of the thermal aging capability of the equipment being tested.

Additional conservatism is applied to all Arrhenius calculations in the following manner:

- a. The lowest activation energy was used for all materials unless justification for use of a higher value could be provided.

ENCLOSURE 2

EQUIPMENT QUALIFICATION OPEN ITEM RESOLUTIONS

The table on the following pages provides the specific resolution for each equipment qualification open item identified in the TER. The resolutions provided are current to the date of this letter and incorporate comments received from the Staff during the December 12, 1983 meeting. Additional details of the resolutions are provided in the NSP/Monticello Response to the NRC TER, submitted April, 1983, or in the Monticello Central File.

Two important considerations were factored into each radiation analysis:

1. Specified radiation doses were determined in accordance with the DOR Guidelines, taking into account the time equipment was required to remain functional and its location. All radiation doses were computed based on an operating time of at least one hour.
2. Radiation data for equipment or materials which were type-tested but did not fail was considered to be overly conservative. Before performing a supplemental radiation analysis, type-test data was scrutinized to insure that critical material properties were monitored for failure.

Beta radiation qualification was established by demonstrating at least one of the following three conditions:

1. The total required gamma and beta radiation dose which penetrates existing shielding is less than (or equal to) 10% of the qualified gamma dose demonstrated in type testing.
3. The tested gamma dose exceeds the required gamma dose plus the unshielded beta dose.

This qualification methodology is also in accordance with the DOR Guidelines.

1.0 CONTROLS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

<u>TER Item No.</u>	<u>NRC Cat.</u>	<u>Equipment Description</u>	<u>Plant ID Number</u>	<u>Deficiencies</u>	<u>Response/Resolution</u>
122 123 124	II.c	Woodward Governor; m/ns 1680-622, 8271-083, 8270-811	NA	Aging; Qualified life/ replacement schedule	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1.
125	II.a	Woodward Governor; m/n R8250-133	NA	Documentation	The qualification file for this component has been revised to incorporate additional materi- als data from the manufacturer. The Central File documents resolution of all cited defi- ciencies. This equipment is fully qualified.

2.1 ELECTRICAL EQUIPMENT: CABLES

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
142	II.a	Carol Wire and Cable; m/n 1050	NA	Radiation	Additional analysis based on the materials of construction has been performed to address the effects of radiation on this cable. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
131	II.a	General Electric; m/n SI 58042		Similarity	This cable is considered qualified because it is constructed with the same insulation and jacket materials as other GE cable that has been successfully tested. However, because the radiation doses resulting from certain events that are not design basis events, but are required to be considered by the TMI Action Plan, exceed the qualification level of this cable, it is scheduled for replacement with fully qualified cable during the February 1984 outage.
132 133	II.a	General Electric; m/ns SI 58170, SI 58175		Documentation	The Central File documents resolution of all cited deficiencies. In addition, the qualified life for these cables has been established in a manner similar to that used for equipment items accepted as qualified by the NRC. This equipment is fully qualified.
134 139	I.a	General Electric; m/ns SI 57275, SI 58109	NA	None	Qualified
135 136	II.a	General Electric; m/ns SI 58007	NA	Similarity	Cables taken from Monticello were tested to provide adequate documentation for all environmental parameters except thermal aging. Qualification for thermal aging was addressed using test data for other GE cables made of the same materials as those installed. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.

2.1 ELECTRICAL EQUIPMENT: CABLES (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
140	II.a	General Electric; m/n SI 58136	NA	Radiation; Criteria regarding test failures/severe anomalies	The installed cables will be exposed to significantly less radiation ($3 \times 10^6 R$) than cables which failed during testing ($4 \times 10^7 R$). A materials analysis of the cable materials was performed to supplement qualification test data. Cable with identical materials of construction to those installed has been demonstrated to be serviceable after exposure to radiation in excess of the required dose for installed cables. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
138	I.b	General Electric; m/n SI 58081	NA	Documentation	This cable recently passed qualification testing successfully. Documentation of qualification will be included in the Central File upon receipt of the test report.
137	I.a	Raychem; m/n WCSF-N	NA	None	Qualified
141	I.a	Rockbestos; m/n Firewall III	NA	None	Qualified
144	II.a	Samuel Moore; m/n 1952		Similarity; Aging; Qualified life/ replacement schedule	Qualification test data for identical cables which was not available at the time of the FRC audit was used to establish a qualified life for the installed cable. This equipment is fully qualified.
143	I.b	Samuel Moore; m/ns 1802, 1852, 1862	NA	Criteria regarding steam exposure	Approximately half of this cable has been replaced. The remainder will be replaced during the February 1984 outage. Submergence testing and air oven aging provides reasonable assurance for functionality of this cable during an accident.

2.2 ELECTRICAL EQUIPMENT: TERMINAL BLOCKS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
113	II.a	Allen-Bradley; m/n Bulletin 1492CD3	NA	Radiation	Qualification type-test data which envelopes the required radiation dose for this equipment was used to establish radiation qualification for this component. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
114	I.b	General Electric Terminal Blocks; m/n CR151D3	NA	Documentation	Additional type-test data was used to resolve the documentation deficiency. The Central File documents resolution of this deficiency. This equipment is fully qualified.

2.3 ELECTRICAL EQUIPMENT: MOTORS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
102	II.c	Baldor; m/n 310406404	P-217	Aging; Qualified life/ replacement schedule	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
97	I.b	General Electric; m/n 5K145A1246	K-10(A,B)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
98	I.b	General Electric; m/n 5K43KG2802	K-11	Documentation; Aging; Qualified life/ replacement schedule; Radiation	This equipment has been replaced with a fully qualified Reliance motor.
99	I.b	General Electric Motor; m/n 5K254AK205	V-EF-17(A,B)	Documentation; Aging; Qualified life/ replacement schedule; Radiation	An equipment-specific radiation evaluation was performed to reduce excess conservatisms in the previous evaluation; radiation qualification has been established for the required dose, based on materials test data. A qualified life has been established for this equipment with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
100 101	II.a	General Electric; m/n K56329XC4A, 5K6338XC298	P-202(A-D)	Similarity; Aging; Qualified life; replacement schedule	A qualification report provided by GE for these particular motors was used to reevaluate environmental qualification of this equipment. A qualified life has been established for this equipment with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
103 104	I.b	US Electric; m/ns F132302268; s/ns	V-AC-4 V-AC-5 V-AC-8(A,B)	Documentation	This equipment has been replaced with fully qualified Reliance motors.

2.4 ELECTRICAL EQUIPMENT: MOTOR STARTERS/MOTOR CONTROL CENTERS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
107	I.b	Furnas Electric Motor Starter; m/n 14BA32BC	K-11	Documentation	Additional data regarding the materials of construction was obtained from the manufacturer which fully resolves the documentation deficiency. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
105 106	I.b	General Electric; m/n CR106	K-10(A,B)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in 11/81 SER Response.
109 110 111 112	I.b	General Electric; m/ns IC-7700, DA-7700	MCC D-31205 MCC D-312 MCC D-31104 MCC 143 MCC 143(A,B)	Documentation	Previous qualification testing is being evaluated for this equipment. JCOs were provided in the 11/81 SER Response.

2.5 ELECTRICAL EQUIPMENT: PENETRATIONS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

<u>TER Item No.</u>	<u>NRC Cat.</u>	<u>Equipment Description</u>	<u>Plant ID Number</u>	<u>Deficiencies</u>	<u>Response/Resolution</u>
129	II.a	General Electric; m/n 237X627G7	JX-105(A,C,D)	Similarity	Contact with the vendor provided additional information which fully resolves the similarity deficiency. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.

2.6 ELECTRICAL EQUIPMENT: MISCELLANEOUS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
108	I.b	Cutler-Hammer; m/n 6102	LC-1(A,B)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
127	I.b	Cutler-Hammer; m/n 4105H311H	LC-2(A,B)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
121	I.b	Hevi-Duty Electric Transformer; m/n SZ0	T1	Aging; Qualified life/ replacement schedule; Radiation	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. A materials analysis was performed which demonstrates that the radiation damage threshold of all non-metallic components is greater than the specified dose for this equipment. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
130	I.a	Honeywell; m/n T451A1132	E-34(A,B)	None	Qualified
126	I.b	ILG Industries Electric Heater; m/n H7133	E-34(A,B)	Documentation	These heaters have been exempted from qualification based upon an evaluation of system operating requirements. The heaters are only required during standby operation of the SGBT system. Upon system initiation, the heaters are turned off by the system logic. In addition, the heaters are isolated from the essential portions of the system by coordinated fuses.
128	I.a	Johnson; m/n 1080302001	NA	None	Qualified

3.0 ELECTRO/PNEUMATIC TRANSDUCERS

NORTHERN STATES POWER
 Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
119	I.b	Leeds and Northrup; m/n 109701	E/P 2942 E/P 2943	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
120	II.a	Fisher; m/n 546	E/P 1728 E/P 1729	Documentation	A plant-specific analysis of these transducers was made using applicable Fisher type-test data. This documentation provided was found to be acceptable during our December 12 meeting with the Staff; therefore, the deficiency for this item has been resolved. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.

4.0 MOTORIZED VALVE OPERATORS/ACTUATORS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
1,2, 5,7, 148, 149	II.a	Limitorque; m/ns SMB-000 thru SMB-4 operators/ AC/DC/various insulation classes	MO 2-53(A,B) MO 2-54(A,B) MO-2034 MO-2397 MO-2070 MO-2029 MO-2373 MO-2075 MO-2374 MO-2076 MO-2035 MO-2398 MO-2067 MO-2107 MO-2071	Documentation; Simi- larity, Aging; Quali- fied life/replacement schedule	Similarity for these components was resolved in a manner similar to other Limitorque valve motor operator evaluations which were previ- ously accepted by Franklin. Discussions with the vendor indicate that all Limitorque test- ing was conducted with operators mounted in the worst possible configuration. Therefore, testing envelopes all possible installed con- ditions. A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
4,8	II.a	Limitorque; m/ns SMB-4/DC/ Class B, SMB-2/AC/ Class H	MO-2068 MO-1751 through MO-1754	Documentation; Simi- larity, Aging; Quali- fied life/replacement schedule	These operators are being replaced with fully qualified operators during the February 1984 outage. A JCO was provided for the motor brakes in the 11/81 SER Response.
3,6, 32, 150	II.c	Limitorque; m/ns SMB-000 thru SMB-4 operators/ AC/DC/various insulation classes	MO-2036 MO-2030 MO-2012 MO-2013 MO-2014 MO-2015	Aging; Qualified life/ replacement schedule	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficien- cies. This equipment is fully qualified.
12	II.a	Rotork valve actuator; m/n 16AMKII	MO-2061 MO-2062 MO-2063	Criteria regarding test failures/ severe anomalies	Anomalies occurred only during the LOCA por- tion of testing as a result of steam exposure. Since these operators will not be subjected to a steam environment at any time during which their operation is required, this deficiency is considered resolved. The Central File docu- ments resolution of all cited deficiencies. This equipment is fully qualified.

4.0 MOTORIZED VALVE OPERATORS/ACTUATORS (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
9	I.a	Rotork; m/ns 12A, 14AMKII, 30A, 150A	MO-2407	Criteria regarding test failures/severe anomalies	These equipment items were originally categor- ized as II.b, but have been reclassified as I.a (Qualified). Ref. NRC Letter To: D. Musolf (NSP), From: D. Vassallo, dated June 16, 1983. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
10			MO-2032		
13			MO-2010		
14			MO-2011		
15			MO-2020		
17			MO-2021		
19			MO-2022		
			MO-2023		
			MO-2399		
			MO-1749		
			MO-1750		
	MO-1986				
	MO-1987				
	MO-1988				
	MO-1989				
	MO-2002				
	MO-2003				
11	II.b	Rotork; m/n 14AMKII	MO-1426	Criteria regarding test failures/severe anomalies	This component was identified in Table 4-1 of the TER as Category II.c. The evaluation for this item in the body of the TER identifies this actuator as Category II.b. The failure experienced during testing would not prevent this component from performing its safety function since it is required only to close and not reopen. Qualification of this operator is consistent with qualification provided for MO-2399 in NSP Letter To: Dir. of NRR, From: D. Musolf, dated March 3, 1983. The Central File documents resolution of all cited defi- ciencies. This equipment is fully qualified.

4.0 MOTORIZED VALVE OPERATORS/ACTUATORS (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
16	I.b	Rotork; m/n 35A	MO-2006 MO-2007	Criteria regarding test failures/severe anomalies	These actuators have been recategorized as I.b: Ref. NRC letter to: D. Musolf (NSP), From: D. Vassallo, dated June 16, 1983. The JCO provided for this equipment was accepted by the NRC in the referenced correspondence. Fully qualified torque/limit switches will be installed during the February 1984 outage.
18	II.c	Rotork; m/n 70NA4	MO-2008 MO-2009	Aging; Qualified life/replacement schedule	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.

5.0 RADIATION DETECTORS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

<u>TER Item No.</u>	<u>NRC Cat.</u>	<u>Equipment Description</u>	<u>Plant ID Number</u>	<u>Deficiencies</u>	<u>Response/Resolution</u>
145	II.a	General Atomic; m/n RD23	RE-7860(A,B)	Similarity; Qualified life/replacement schedule; Aging simu- lation, Radiation	Tested components are identical to the installed components; one of the qualified installation techniques is used at Monticello. The qualification test report addresses the effects of radiation and aging on the consti- tuent components and fully resolves the defi- ciencies cited for this equipment. The Central File documents resolution of all cited defi- ciencies. This equipment is fully qualified.

6.0 SOLENOID VALVES

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
20	I.b	ASCO; m/n THT 831723, THT8317 A23, THT83212, THP830081RU, 8300C64U, 8262A212	SV-2978	Documentation	The majority of these solenoid valves have been replaced with fully qualified valves. The remaining valves will be replaced during the February 1984 outage. JCOs were provided for valves which are still installed in the 11/81 SER Response.
21			SV-2979		
22			SV-2982(A,B)		
23			SV-2944		
24			SV-2945		
26			SV-2385		
27			SV-2791		
28			SV-1478		
29			SV-1994		
33			SV-1995		
37			SV-1996		
			SV-1997		
			SV-2065		
			SV-1728		
			SV-1729		
			SV-2386		
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	SV-3268				
	SV-3269A				
	SV-3307				
	SV-3308				
	SV-3311				
	SV-3312				
	SV-3313				
	SV-3314				
25	II.c	ASCO; m/n NP 8321A1E	SV-2790	Aging; Qualified life/ replacement schedule	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. This equipment is fully qualified.

6.0 SOLENOID VALVES (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
30	II.a	ASCO; m/n WPLBX	SV 3-140(A,B)	Aging; Qualified life/ replacement schedule; Aging program; Criteria regarding aging simulation	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficien- cies. This equipment is fully qualified.
31	II.a	ASCO; m/n HVA-90	SV 3-31(A,B) SV 3-13-117 SV 3-13-118	Aging; Qualified life/ replacement schedule; Criteria regarding aging simulation	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficien- cies. This equipment is fully qualified.
34 35 36	II.a	Automatic Valve Company; m/ns C498815, C5450	SV 2-80(A,D) SV 2-71(A,B,D) SV 2-71(A-H)	Similarity; Aging; Criteria regarding aging simulation; Criteria regarding peak temperature; Criteria regarding profile envelope; Criteria regarding steam exposure; Criteria regarding spray; Criteria regarding functional testing	At the time FRC reviewed the evaluation for this equipment, a limited amount of test data was available. The file has since been revised to incorporate additional test documentation which fully resolves the cited deficiencies. Correspondence with the equipment manufacturer was used to evaluate the impact of differences in construction between the tested and installed solenoid valves. It was determined that variations in construction enhance the installed valve design and, therefore, do not invalidate use of the test documentation. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
37	I.b	ASCO; m/n THT831723	SV-3305 SV-3306 SV-3309 SV-3310	Documentation	These solenoid valves have been removed from the system because of TMI modifications.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE

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Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
51 52 53 54 55 56 57	II.a	Barksdale; m/n B2TA12SS	PS 2-3-49(A,B) PS 2-3-50(A,B) PS 14-44(A-D) PS 2-3-53(A,B) PS 23-68(A-D) PS 2-3-51(A-D) PS 2-3-52A PS 5-14(A-D)	Aging; Criteria re- garding peak pressure; Criteria regarding peak temperature (TER Item 57 only)	Pressure qualification for these switches was established by evaluating the severity of industrial rating test requirements (e.g., NEMA 4 hosedown test). Interfaces have been installed which are equivalent to those tested. For TER Item 57 only, it was determined that the peak temperature specified for this component (214°F) is of sufficiently short duration that testing to 212°F provides reasonable assurance of the operability of the equipment during an accident. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
58 76 78 71	II.a	Barton; m/ns 278, 288, 288A	PS 2-3-52B dPIS 13-83 dPIS 13-84 dPIS 2-116(A-D) dPIS 2-117(A-D) dPIS 2-118(A-D) dPIS 2-119(A-D) dPIS 23-76(A,B) dPIS 23-77(A,B) FS 23-78	Documentation; Aging; Qualified life/re- placement schedule; Criteria regarding peak pressure; Cri- teria regarding profile envelope; Criteria regarding steam exposure	Additional test data provided by General Electric for NSSS safety-related electrical equipment supplied to Monticello were used to establish qualification of these switches which resolves cited deficiencies. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
77	III.b	Barton; m/n 288	dPIS 2-136(A,B) through dPIS 2-139(A,B) dPIS 2-129(A-D)	None	Qualified documentation is available in the Central File.
79	I.a	Barton; m/n 289A	dPS 2573 dPS 2572	None	Qualified
84	I.b	Chromalox; m/n AR2529	TS-3368 TS-3369	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
38	I.b	Cutler-Hammer; m/n Type L	A0-2982	Documentation	This equipment has been replaced with fully qualified equipment.
85	II.a	Fenwal; m/ns 17002-36, 17002-40	TS 2-121(A-D)	Similarity, Qualified life/replacement schedule; Criteria regarding steam expo- sure; Radiation	Correspondence with the vendor was performed to obtain additional data on the design and materials of construction of these switches. Tested and installed switches are constructed of the same materials and both are a hermetically sealed design. The tested switches were submerged in water at 200°F, which is adequate to simulate steam exposure. A materials analysis was also performed to demonstrate that the radiation damage threshold of all materials is greater than the specified radiation dose. Additionally, a qualified life has been established for this equipment with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
86			TS 2-122(A-D)		
87			TS 2-123(A-D)		
88			TS 2-124(A-D)		
			TS 23-101(A-D)		
			TS 23-102(A-D)		
			TS 23-103(A-D)		
			TS 23-104(A-D)		
	TS 13-79(A-D)				
	TS 13-80(A-D)				
	TS 13-81(A-D)				
	TS 13-82(A-D)				
59	III.a	Furnas; m/n 69GAV7	PS-3462	None	This component still performs a safety-related function in a harsh environment. Documentation of its qualification is included in the Central File. This equipment is fully qualified.
115 116	I.b	General Electric; m/n THN3361, Mod. 2	N3347 N4347	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
117	III.a	General Electric; m/n THN3361, Mod. 2	K-11	None	This component has been removed from the system.
74	I.b	Magnetrol; m/n 249C	LS 23-91(A,B)	Criteria regarding peak pressure; Criteria regarding steam exposure	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
72	II.a	McDonnell Miller; m/n AFIS	FS 2950 FS 2951	Similarity	Additional information from the vendor was used to establish similarity. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
60	II.a	Meletron; m/n 3726SS49A	PS 13-87(A-D)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
61	I.b	Mercoïd; m/n DAW23156	PS 10-105(A-D)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
62 63	II.a	Mercoïd; m/ns DAW44341, DA7043804	PS 23-84 PS 23-97(A,B)	Aging; Qualified life/replacement schedule; Criteria regarding peak pressure; Criteria regarding steam exposure	A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. A plant-specific analysis was performed which shows that accidents for which this equipment must operate result in a thermal-hydraulically mild environment at the equipment location. Therefore, radiation and operating time (based on radiation) are the only harsh environmental parameters for which qualification must be established. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
39	II.a	Microswitch; m/n OPAR	AO-2944 AO-2978 AO-2979	Radiation	The switching mechanism for this component has been radiation-tested in excess of the specified dose. Post-accident conditions are thermal-hydraulically mild for this component. All other non-metallic components are either not susceptible to radiation damage or not required for operation of this equipment in thermal-hydraulically non-harsh environments. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
40 41	I.b	Microswitch; m/n BZE62RN, BZE62RQ2	AO-2451(A,B) AO-2561(A,B) AO-2377 AO-2378 AO-2379 AO-2380 AO-2381 AO-2383 AO-2386 AO-2387 AO-2396	Criteria regarding peak pressure; Radia- tion; Criteria regard- ing steam exposure	TER Item 40: This equipment has been replaced with fully qualified equipment. TER Item 41: This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
42 43 47 48	I.b	NAMCO; m/ns D1200G, EA17014100, D2400XST, EA08021100	CV-3305 through CV-3314 CV-7440 CV-7956	Documentation	TER Items 43, 47: This equipment is scheduled for replacement during the February 1984 out- age. JCOs for these items were included in the the 11/81 SER Response. TER Items 42, 48: This equipment has been replaced with fully qualified equipment.
44	II.a	NAMCO; m/n EA170	HPCI Turbine Stop Valve Position	Documentation	The materials list, radiation threshold values, and data sources for radiation quali- fication which were not identified in previous submittals have been clarified in the revised calculation file. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
45	II.a	NAMCO; m/n EA740	AO 2-80(A-D)	Criteria regarding steam exposure	These limit switches are fully qualified; how- ever, the required conduit sealing devices have not been installed; they are scheduled for installation during the February 1984 outage. As discussed during the December 12 meeting, this situation is considered accept- able because these switches provide only posi- tion indication for the inboard MSIVs. There is a redundant set of isolation valves that would be unaffected by any event that causes a harsh environment at the switch location.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
46	II.a	NAMCO; m/n EA7408000	CV 2790	Criteria regarding steam exposure	Fully qualified limit switches, including the required conduit sealing devices, are scheduled for installation during the February 1984 outage. As discussed during the December 12 meeting, this situation is considered acceptable because these switches provide only position indication for the inboard recirc-loop sample valve. There is a redundant isolation valve that would be unaffected by any event that causes a harsh environment at the switch location.
49	II.a	NAMCO; m/n SL5C3L	CV 2384	Similarity	These limit switches have been replaced by fully qualified switches; however, the required conduit sealing devices have not been installed; they are scheduled for installation during the February 1984 outage. As discussed during the December 12 meeting, this situation is considered acceptable because these switches provide only position indication for the inboard torus vent valve. This valve is not required to mitigate any event that causes a harsh pressure, temperature, or humidity environment at the switch location.
50	II.a	NAMCO; m/n SL3B2W	AO 2-86(A-D)	Similarity	These limit switches have been replaced by fully qualified switches; however, the required conduit sealing devices have not been installed; they are scheduled for installation during the February 1984 outage. As discussed during the December 12 meeting, this situation is considered acceptable because these switches provide only position indication for the outboard MSIVs. There is a redundant set of isolation valves that would be unaffected by any event that causes a harsh environment at the switch location.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
73	I.b	Peeco; m/n HP-F	FS 10-121(A-D)	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
146	II.a	Pressure Controls; m/n A171P	PS-7356 PS-7463 PS-7465 PS-7307 PS-7358 PS-7466 PS-7359 PS-7464	Aging; Qualified life/replacement schedule; Criteria regarding spray	Pre-aging analysis of these components was included in the qualification test program. The switch housing is NEMA 4, which is designed to withstand direct spray impingement. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
75	II.a	Robertshaw; m/n SL412A1	LS 23-74 LS 23-75	Similarity	Specific materials information for the tested and installed components was obtained from the vendor. An analysis was performed which demonstrates that the differences in design and construction are not significant for qualification purposes. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
64	I.b	Square-D; m/n GHG551	PS-7193 PS-7192	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
65 66 67 68 69 70	II.a	Static-O-Ring; m/ns 12NAA4, 5NAA3X, 6NAA3, 12NK4	PS 10-101(A-D) PS 10-100(A-D) PS 10-119(A-D) PS 10-105(E-H) PS 2-128(A,B) PS 5-12(A-D)	Documentation; Quali- fied life/replace- ment schedule	These pressure switches have been re-evaluated using additional information provided by the manufacturer. The qualification file has been revised to resolve the documentation deficiency. A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.

7.0 SWITCHES: PRESSURE/LEVEL/LIMIT/FLOW/MICRO/TEMPERATURE (Cont'd)

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID	Deficiencies	Response/Resolution
81	II.a	Yarway; m/n 4418C	LIS 2-3-57(A,B) LIS 2-3-58(A,B) LIS 2-3-72(A,D)	Similarity; Aging; Qualified life/ replacement schedule; Aging Program; Criteria regarding aging simulation; Criteria regarding peak pressure	Similarity between tested and installed components was established by performing a detailed component analysis based on information provided by the manufacturer regarding the design and operation of these indicators. Post-accident pressure variations were determined to fall within the range of normal atmospheric fluctuations. A detailed materials analysis was also performed to determine the radiation damage threshold of constituent components: all thresholds were below the specified radiation dose for these components. A qualified life has been established for this equipment in accordance with the methodology outlined in Enclosure 1. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.
82 83	I.b	Yarway; m/ns 4418CE	LITS 2-3-59(A,B) LITS 2-3-73(A,B)	Similarity; Aging; Qualified life/ replacement schedule; Aging program; Criteria regarding aging simulation; criteria regarding peak pressure; Criteria regarding test sequence.	This equipment is scheduled for replacement during the February 1984 outage. JCOs were provided in the 11/81 SER Response.

8.0 TEMPERATURE ELEMENTS

NORTHERN STATES POWER
 Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
118	I.b	ThermoElectric; m/n CES 1625616T	TE 2995(F,G)	Documentation	This equipment has been replaced with fully qualified Pyco thermocouples.

9.0 TRANSMITTERS

NORTHERN STATES POWER
Monticello Nuclear Generating Plant

TER Item No.	NRC Cat.	Equipment Description	Plant ID Number	Deficiencies	Response/Resolution
80	I.b	Barton; m/n 296	DPT 10-91(A,B)	Documentation	These components have either already been replaced with fully qualified equipment or are scheduled to be replaced during the February 1984 outage. JCOs were provided for equipment which is still installed in the 11/81 SER Response.
89 90 91 94 95 96	I.b	General Electric; m/ns 551, 552, 553	FT 10-111(A,B) FT 10-109(A,B) FT 14-40(A,B) FT 23-82 LT 2-3-61 PT 6-53(A,B) PT-2994B	Documentation	These components have either already been replaced with fully qualified equipment or are scheduled to be replaced during the February 1984 outage. JCOs were provided for equipment which is still installed in the 11/81 SER Response.
92	I.b	Leeds and Northrup; m/n 191221(-) 0000001000100	FT-2942 FT-2943	Documentation	This equipment is scheduled for replacement during the February 1984 outage. A JCO was provided in the 11/81 SER Response.
93 147	II.a	Rosemount; m/ns 1153DA5, 1153AA17	LT-7338(A,B) PT-7251(A,B)	Documentation; Aging; Qualified life/ replacement schedule; Criteria regarding test failures/severe anomalies; Criteria regarding instrument accuracy.	Aging qualification was established in accordance with the methodology outlined in Enclosure 1. Failures demonstrated during testing occurred after exposure to environmental conditions more severe than those to which the installed equipment will be exposed. This data was used only to establish qualification for thermal aging and only up to the point of failure in the test. Rosemount testing of 1153 Series A transmitters demonstrated a maximum error of 6.95% of upper span during LOCA testing. The transmitters function only as display instrumentation of wide range torus water level and wide-range containment pressure, and are not used to initiate any manual or automatic actions. The Central File documents resolution of all cited deficiencies. This equipment is fully qualified.